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PSEG
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10 CFR 50.54(f)

LR-N14-0051

MAR 28 2014

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Salem Generating Station Units 1 and 2
Renewed Facility Operating License Nos. DPR-70 and DPR-75
NRC Docket Nos. 50-272 and 50-311

Subject: PSEG Nuclear LLC'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 – Salem Generating Station

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

2. NEI letter to NRC, "Proposed Path Forward for NTTF Recommendation 2.1: Seismic Reevaluations," dated April 9, 2013 (ADAMS Accession No. 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 (ADAMS Accession No. 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" (ADAMS Accession No. ML12333A170)

5. NRC Letter, "Endorsement of Electric Power Research Institute Final Draft Report 1025287, 'Seismic Evaluation Guidance'," dated February 15, 2013 (ADAMS Accession No. ML12319A074)

On March 12, 2012, the Nuclear Regulatory Commission (NRC) issued a request for information (Reference 1) pursuant to 10 CFR 50.54(f) to all power reactor licensees and holders of construction permits in active or deferred status. Enclosure 1 of the 10 CFR 50.54(f) letter 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 the request. The enclosed Salem Generating Station (SGS) Seismic Hazard and Screening Report is provided in response to the 10 CFR 50.54(f) letter.

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 the information needed to complete the reports. NEI proposed that descriptions of subsurface materials and properties and base case velocity profiles be submitted by September 12, 2013, with the remaining seismic hazard and screening information to be provided by March 31, 2014. By letter dated May 7, 2013 (Reference 3), the NRC agreed with the path forward proposed by NEI.

EPRI Report 1025287 (Reference 4) provides industry guidance and detailed information to be included in the Seismic Hazard Evaluation and Screening Report submittals. The NRC endorsed EPRI Report 1025287 via Reference 5. Enclosure 1 is the SGS Seismic Hazard Evaluation and Screening Report based on the NRC-endorsed guidance and schedule.

Based on the results of the screening evaluation provided in Section 4 of Enclosure 1, SGS Units 1 and 2 will perform a spent fuel pool evaluation and a relay chatter review. In addition, SGS Units 1 and 2 will implement the Expedited Seismic Evaluation Process as proposed by NEI in Reference 2 and endorsed by the NRC in Reference 3. There are regulatory commitments contained in this letter as identified in Enclosure 2.

If you have any questions or require additional information, please do not hesitate to contact Mr. Lee Marabella at 856-339-1208.

10 CFR 50.54(f)

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 3/28/2014
(Date)

Sincerely,

John F. Perry
Site Vice President
Salem Generating Station

Enclosures:

1. Salem Generating Station - Seismic Hazard and Screening Report
2. Summary of Commitments

cc: Mr. E. Leeds, Director of Office of Nuclear Reactor Regulation
Mr. W. Dean, Administrator, Region I, NRC
Mr. J. Hughey, Project Manager, NRC
NRC Senior Resident Inspector, Salem
Mr. P. Mulligan, Manager IV, NJBNE
Salem Commitment Tracking Coordinator
PSEG Corporate Commitment Coordinator

LR-N14-0051

Enclosure 1

Salem Generating Station - Seismic Hazard and Screening Report

Salem Generating Station Units 1 and 2
PSEG Nuclear LLC

MPR
ASSOCIATES INC.
ENGINEERS

Salem Generating Station – Seismic Hazard and Screening Report

0108-1310-REPT-01

March 26, 2014
Revision 0

QUALITY ASSURANCE DOCUMENT
This document has been prepared, reviewed, and approved in accordance with the Quality Assurance requirements of 10CFR50 Appendix B and/or ASME NQA-1, as specified in the MPR Nuclear Quality Assurance Program.

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Salem Generating Station – Seismic Hazard and Screening Report

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 [1] that requests information to assure that these recommendations are addressed by all U.S. nuclear power plants. The 50.54(f) letter [1] 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 [1] pertaining to NTTF Recommendation 2.1 for the Salem Generating Station (SGS) Units 1 and 2, located in Lower Alloways Creek Township, Salem County, New Jersey. In providing this information, PSEG Nuclear 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* [2]. The Augmented Approach, *Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic* [3], 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 SGS Safe Shutdown Earthquake Ground Motion (SSE) is developed based on the original geologic and seismic citing investigations. The SSE is used for the design of seismic Category I systems, structures and components and meets General Design Criterion 2 in Appendix A to 10 CFR Part 50.

In response to the 50.54(f) letter [1] and following the guidance provided in the SPID [2] , a seismic hazard reevaluation for SGS was performed. For screening purposes, a Ground Motion Response Spectrum (GMRS) was developed.

2.0 Seismic Hazard Reevaluation

Except where otherwise noted, all information provided in Section 2 is taken from [4].

2.1 Regional and Local Geology

The regional and local geology of the Salem site is provided in Section 2.5 of the SGS UFSAR [5]. Information from the UFSAR is summarized below.

The Salem site is located on the southern part of Artificial Island on the east bank of the Delaware River in Lower Alloways Creek Township, Salem County, New Jersey. SGS is located approximately 19 miles south of Wilmington, Delaware. The site lies within the Atlantic Coastal Plain Physiographic Province, about 18 miles southeast of the Piedmont Physiographic Province. The Fall Zone marks the contact of the low lying, gently undulating terrain of the Coastal Plain and the higher, more rugged terrain of the Piedmont Province. The site structures are founded on the Paleocene-Eocene Vincentown Formation, a competent, cemented, granular soil. Below the Vincentown are about 1800 feet of increasingly older sediments.

The Vincentown Formation was determined to be the closest stratum to the ground surface suitable for foundation support. In the Salem Station area the Vincentown is located about 70 feet below grade. The bottom of the base mats of the major Category I structures are located 22 to 46 feet below grade. A lean concrete fill was placed between the Vincentown and the base of the Category I structures.

The site is underlain by about 1800 feet of Cretaceous, Tertiary, and Quaternary-aged sediments. Conditions encountered at the site are completely consistent with the known regional picture. No known faults exist within the basement rock or sedimentary deposits in the vicinity of the site.

Earthquake activity in historic time within 200 miles of the plant site has been moderate. Sources of major earthquakes in the central and eastern United States (CEUS) are distant, and have not had an appreciable effect at the site. The original investigation of historical seismic activity in the region indicated that a design intensity of VII (Modified Mercalli Scale) at an epicentral distance of about 15 to 20 miles is adequately conservative for the site. PSEG determined that Intensity VII corresponds to a peak ground acceleration of 0.10 g, which was increased to 0.20 g for the design basis SSE.

2.2 Probabilistic Seismic Hazard Analysis

2.2.1 Probabilistic Seismic Hazard Analysis Results

In accordance with the 50.54(f) letter [1] and following the guidance in the SPID [2], 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 [6] together with the updated EPRI Ground-Motion Model (GMM) for the CEUS [7]. For the PSHA, a lower-bound moment magnitude of 5.0 was used, as specified in the 50.54(f) letter [1].

For the PSHA, the CEUS-SSC background seismic sources out to a distance of 400 miles (640 km) around SGS were included. This distance exceeds the 200 mile (320 km) recommendation contained in Regulatory Guide 1.208 [8] and was chosen for completeness. Background sources included in this site analysis are the following:

1. Atlantic Highly Extended Crust (AHEX)
2. Extended Continental Crust—Atlantic Margin (ECC_AM)
3. Mesozoic and younger extended prior – narrow (MESE-N)
4. Mesozoic and younger extended prior – wide (MESE-W)
5. Midcontinent-Craton alternative A (MIDC_A)
6. Midcontinent-Craton alternative B (MIDC_B)
7. Midcontinent-Craton alternative C (MIDC_C)
8. Midcontinent-Craton alternative D (MIDC_D)
9. Northern Appalachians (NAP)
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. St. Lawrence Rift, including the Ottawa and Saguenay grabens (SLR)
15. Study region (STUDY_R)

For sources of large magnitude earthquakes, designated Repeated Large Magnitude Earthquake (RLME) sources in [6], the following sources lie within 1,000 km of the site and were included in the analysis:

1. Charleston
2. Charlevoix
3. Wabash Valley

For each of the above background and RLME sources, the mid-continent version of the updated CEUS EPRI GMM [7] was used.

2.2.2 Base Rock Seismic Hazard Curves

Consistent with the SPID [2], 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 Section 3 at the SSE control point elevation.

2.3 Site Response Evaluation

Following the guidance contained in Seismic Enclosure 1 of the 50.54(f) letter [1] and in the SPID [2] for nuclear power plant sites that are not sited on hard rock (defined as 2.83 km/sec), a site response analysis was performed for SGS.

2.3.1 Description of Subsurface Material

The SGS site is located in the eastern US on the Delaware River within the Coastal Plain physiographic province. The Coastal Plain is underlain by a thick wedge of unconsolidated sediment ranging from Cretaceous to recent in age. Bedrock is estimated to be at a depth of about 1,800 ft [9].

The site region includes parts of several other physiographic provinces: the Continental Rise, Continental Slope, and Continental Shelf (from east to west), all located in the eastern portions of the site region, and (to the west) the Piedmont, New England, Blue Ridge, Valley and Ridge and Appalachian Plateau provinces [9].

The Coastal Plain province is characterized by low-lying, gently rolling terrain developed on sequences of deltaic, shallow, marine and continental shelf clastics consisting primarily of unconsolidated to semi-consolidated gravels, sands, silts, and clays that dip gently oceanward. The surface has been modified by erosional and depositional landforms associated with several transgressional and regressional marine cycles. The site stratigraphy described below is based on recent work completed for the PSEG Early Site Permit Application (ESPA) [10] for potential future construction at the SGS site [9]:

1) Basement Complex (Section 2.5.1.2.2.2 of [10])

    a) Control of the nature of basement lithologies that underlie the ESPA Site is provided by PSEG No. 6 production well located approximately 0.6 mi.from the site center. The log for this well reports residual clay, which is interpreted to be Wissahickon schist from a sidewall core at depth 1800 ft.

2) Coastal Plain Stratigraphic Sequences underlying the Site (Section 2.5.1.2.2.2 of [10])

    a) Lower Cretaceous Strata

i. Potomac Group (Formation): Potomac Group (Formation) strata were sampled in one deep boring within the ESP site (NB-1), and in another deep boring (PSEG Well No. 6) located near the new plant location. The lithologies in these samples consisted of hard plastic, red, gray, and white mottled clay. Boring NB-1 within the ESP site encountered Potomac Group (Formation) at elevation 454 ft. Based on the information provided by Benson from the PSEG No. 6 production well, the Potomac Group (Formation) strata are approximately 1300 ft thick beneath the site.

b) Upper Cretaceous Strata

i. Magothy Formation: The Magothy Formation unconformably overlies the Potomac Group (Formation). The Magothy Formation consists primarily of interbedded gray to dark gray, locally mottled, silts and clays containing trace amounts of lignite and carbonaceous material. The silts and clays were interbedded with sands that contain variable amounts of silt and clay. The Magothy Formation is 52 ft thick beneath the site location.

ii. Merchantville Formation: The Merchantville Formation is composed primarily of dark greenish-black glauconitic silts and clays with variable amounts of sand and mica. These characteristics are representative of those described regionally for this formation. Based on the deep boring NB1, the Merchantville Formation is 30 ft thick beneath the site.

iii. Woodbury Formation: The Woodbury Formation consists of black, micaceous and highly plastic clay. This formation is distinguished from the overlying Englishtown by increased clay and mica content. Based on boring NB-1, the Woodbury Formation is 36 ft thick at the site.

iv. Englishtown Formation: The Englishtown Formation consists of dark gray to black sandy clay, to clayey sand with shell fragments, grading to black silt and clay with trace amounts of glauconite and mica. These characteristics are representative for those reported regionally. The Englishtown is 44 ft thick beneath the site.

v. Marshalltown Formation: The Marshalltown Formation consists of glauconitic, silty, and clayey fine sand. The presence of significant amounts of glauconite and fine-grained nature of the clastic component are characteristic of this unit regionally. Based on borings NB-1 and NB-2, the Marshalltown Formation is 25 ft thick beneath the site.

vi. Wenonah Formation: The Wenonah Formation exhibits a gradational contact with the underlying Marshalltown Formation. The Wenonah Formation consists of sandy clay and clayey sand. Average thickness in borings from the site and nearby areas is 15 ft.

vii. Mount Laurel Formation: The Mount Laurel Formation consists of a dense to very dense brownish-gray to dark green, fine to coarse-grained sand with variable amounts of silt and clay. This unit appears to exhibit a coarsening upward sequence in that glauconite content, in addition to grain size and fine content decrease with depth. Based on borings NB-1, NB-2 and NB-8, at the site, the Mount Laurel Formation ranges from 102 to 105 ft thick.

viii. Navesink Formation: The Navesink Formation consists of fossiliferous, dark green to greenish-black, extremely glauconitic sand. Fossils consisted primarily of pelecypod fragments. Based on all eight of the borings in the site location, the Navesink Formation ranges from 23 to 26 ft thick.

c) Lower Tertiary Strata (Paleocene)

i. Hornerstown Formation: The Hornerstown Formation primarily consists of greenish-gray, to dark green, to greenish-black glauconitic sand with some indurated intervals. Borings at the site location give thickness of 16 to 21 feet.

ii. Vincentown Formation: The Vincentown Formation is a greenish-gray, finegrained to medium-grained silty sand with some zones of clayey sand. Glauconite is commonly present. This unit contains cemented zones from 0.1 to 3.0 ft thick. The Vincentown thickness is highly variable at the ESP site, with thickness of ranging from 35 to 79 ft thick. This variability is due in part to the fact that the top of the Vincentown Formation is a scour surface.

iii. The stratigraphic units overlying the Vincentown Formation are of low strength and are deemed unsuitable to serve as competent layers based on their physical properties. The Vincentown Formation serves as the competent layer for the SGS foundation. Approximately 70 ft of material is above the Vincentown Formation and consists of the following:

- Artificial Fill (mechanically placed)
- Hydraulic Fill
- Alluvium
- Kirkwood Formation

Table 2-1 shows the geotechnical properties for SGS.

TABLE 2-1 [11]
SUMMARY OF GEOTECHNICAL PROFILE DATA FOR SGS

Layer ID number and Formation	Depth Range (feet)	Soil/Rock Description	Unit Weight (pcf)	Shear Wave Velocity (fps)	Compressional Wave Velocity (fps)	Poisson's ratio
0	0-71	Hydraulic fill, alluvium and Tertiary sands (this layer is included for completeness only; it is not present below safety-related structures)	100-137	500	-	-
-	71	Bearing surface at 71' depth. Lean concrete fill placed above this point up to base of containment building mats.	-	-	-	-
1 Vincentown Hornerstown Navesink	71-163	Tertiary dense sands and Cretaceous dense silty and clayey sands	121	2250	-	-
2 Mt Laurel	163-181	Cretaceous dense sand	131	3920	-	-
3 Mt Laurel	181-203	Cretaceous dense sand	131	2490		
4 Mt Laurel	203-237	Cretaceous dense sand	131	3020	-	-
5 Mt Laurel Wenonah Marshaltown	237-299	Cretaceous sandy clay and silty sand, very stiff to hard/dense	128	2490	-	-
6 Marshaltown Englishtown Woodbury	299-383	Cretaceous sandy clay and clayey sand, very stiff to hard	125	1710	-	-
7 Woodbury Merchantville	383-409	Cretaceous silt and clay, hard	130	2290	-	-
8 Merchantville Magothy	409-434	Cretaceous clay and silt, hard	130	1780	-	-
9 Magothy Potomac	434-516	Cretaceous sand with clay and silt, dense	130	2490	-	-
10 Potomac	516-881	Cretaceous Potomac Formation, Upper Zone	135	2200	6200	0.42
11 Potomac	881-1311	Cretaceous Potomac Formation, Middle Zone	135	2630	6200	0.42
12 Potomac	1311-1761	Cretaceous Potomac Formation, Lower Zone	135	3060	6200	0.42
13 Potomac	1761	Seismic Basement, Crystalline Schist	150	11,000	20,450	0.30

2.3.2 Development of Base Case Profiles and Nonlinear Material Properties

2.3.2.1 Shear Wave Velocity Profile

Table 2-1 shows the recommended shear-wave velocities and unit weights along with depth ranges and corresponding stratigraphy. As indicated in [11], the SSE Control Point is located at a depth of 71 ft below grade at the top of the Vincentown Formation (Table 2-1). Mean base-case shear-wave velocities and unit weights were taken from Table 2-1 to Precambrian basement at a depth of about 1,760 ft below grade. The geology and material properties listed in Table 2-1 were taken from the nearby (about 3,000 ft) PSEG ESPA Site and reflect direct shear-wave measurements to a depth of the top of the Potomac Formation (Table 2-1). Below that depth, the shear-wave velocities were based on compressional-wave refraction surveys and assumed Poisson ratios, all at the ESPA site.

To accommodate epistemic uncertainty in shear-wave velocities two scale factors were used: 1.25 to reflect nearby measured shear-wave velocities above the Potomac Formation and 1.57 for the Potomac formation and below, reflecting assumed shear-wave velocities. Profiles extended to a depth (below the SSE control point) of 1,690 ft, randomized ±505 ft. The base-case profiles (P1, P2, and P3) are shown in Figure 2-1 and listed in Table 2-2. 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.

The scale factors of 1.25 and 1.57 reflect a σ_{ln} of about 0.20 and 0.35, based on the SPID [2] 10th and 90th fractiles which implies a 1.28 scale factor on σ_{μ}.

Table 2-2
SGS Layer Thicknesses, Depths, and Shear-wave Velocities (Vs) for 3 Profiles

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	2250		0	1800		0	2812
10.0	10.0	2250	10.0	10.0	1800	10.0	10.0	2812
10.0	20.0	2250	10.0	20.0	1800	10.0	20.0	2812
10.0	30.0	2250	10.0	30.0	1800	10.0	30.0	2812
10.0	40.0	2250	10.0	40.0	1800	10.0	40.0	2812
10.0	50.0	2250	10.0	50.0	1800	10.0	50.0	2812
10.0	60.0	2250	10.0	60.0	1800	10.0	60.0	2812
10.0	70.0	2250	10.0	70.0	1800	10.0	70.0	2812
10.0	80.0	2250	10.0	80.0	1800	10.0	80.0	2812
12.0	92.0	2250	12.0	92.0	1800	12.0	92.0	2812
9.0	101.0	3920	9.0	101.0	3136	9.0	101.0	4900

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)
9.0	110.0	3920	9.0	110.0	3136	9.0	110.0	4900
10.0	120.0	2490	10.0	120.0	1992	10.0	120.0	3112
10.0	130.0	2490	10.0	130.0	1992	10.0	130.0	3112
2.0	132.0	2490	2.0	132.0	1992	2.0	132.0	3112
11.3	143.3	3020	11.3	143.3	2416	11.3	143.3	3775
11.3	154.7	3020	11.3	154.7	2416	11.3	154.7	3775
11.3	166.0	3020	11.3	166.0	2416	11.3	166.0	3775
10.3	176.3	2490	10.3	176.3	1992	10.3	176.3	3112
10.3	186.7	2490	10.3	186.7	1992	10.3	186.7	3112
10.3	197.0	2490	10.3	197.0	1992	10.3	197.0	3112
10.3	207.3	2490	10.3	207.3	1992	10.3	207.3	3112
10.3	217.7	2490	10.3	217.7	1992	10.3	217.7	3112
10.3	228.0	2490	10.3	228.0	1992	10.3	228.0	3112
7.3	235.3	1710	7.3	235.3	1368	7.3	235.3	2137
7.3	242.6	1710	7.3	242.6	1368	7.3	242.6	2137
7.3	250.0	1710	7.3	250.0	1368	7.3	250.0	2137
7.7	257.7	1710	7.7	257.7	1368	7.7	257.7	2137
7.7	265.5	1710	7.7	265.5	1368	7.7	265.5	2137
7.7	273.2	1710	7.7	273.2	1368	7.7	273.2	2137
7.7	281.0	1710	7.7	281.0	1368	7.7	281.0	2137
7.7	288.7	1710	7.7	288.7	1368	7.7	288.7	2137
7.7	296.5	1710	7.7	296.5	1368	7.7	296.5	2137
7.7	304.2	1710	7.7	304.2	1368	7.7	304.2	2137
7.7	312.0	1710	7.7	312.0	1368	7.7	312.0	2137
8.7	320.6	2290	8.7	320.6	1832	8.7	320.6	2862
8.7	329.3	2290	8.7	329.3	1832	8.7	329.3	2862
8.7	338.0	2290	8.7	338.0	1832	8.7	338.0	2862
8.3	346.3	1780	8.3	346.3	1424	8.3	346.3	2225
8.3	354.6	1780	8.3	354.6	1424	8.3	354.6	2225
8.3	363.0	1780	8.3	363.0	1424	8.3	363.0	2225
11.7	374.7	2490	11.7	374.7	1992	11.7	374.7	3112
11.7	386.4	2490	11.7	386.4	1992	11.7	386.4	3112
11.7	398.1	2490	11.7	398.1	1992	11.7	398.1	3112
11.7	409.8	2490	11.7	409.8	1992	11.7	409.8	3112
11.7	421.5	2490	11.7	421.5	1992	11.7	421.5	3112
11.7	433.2	2490	11.7	433.2	1992	11.7	433.2	3112
11.7	445.0	2490	11.7	445.0	1992	11.7	445.0	3112
11.0	456.0	2200	11.0	456.0	1408	11.0	456.0	3454
11.0	467.0	2200	11.0	467.0	1408	11.0	467.0	3454
11.0	478.0	2200	11.0	478.0	1408	11.0	478.0	3454
11.0	489.0	2200	11.0	489.0	1408	11.0	489.0	3454

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)
11.0	500.0	2200	11.0	500.0	1408	11.0	500.0	3454
31.0	531.0	2200	31.0	531.0	1408	31.0	531.0	3454
31.0	562.0	2200	31.0	562.0	1408	31.0	562.0	3454
31.0	593.0	2200	31.0	593.0	1408	31.0	593.0	3454
31.0	623.9	2200	31.0	623.9	1408	31.0	623.9	3454
31.0	654.9	2200	31.0	654.9	1408	31.0	654.9	3454
31.0	685.9	2200	31.0	685.9	1408	31.0	685.9	3454
31.0	716.9	2200	31.0	716.9	1408	31.0	716.9	3454
31.0	747.9	2200	31.0	747.9	1408	31.0	747.9	3454
31.0	778.9	2200	31.0	778.9	1408	31.0	778.9	3454
31.0	809.9	2200	31.0	809.9	1408	31.0	809.9	3454
43.0	852.9	2630	43.0	852.9	1683	43.0	852.9	4129
43.0	895.9	2630	43.0	895.9	1683	43.0	895.9	4129
43.0	938.9	2630	43.0	938.9	1683	43.0	938.9	4129
43.0	981.9	2630	43.0	981.9	1683	43.0	981.9	4129
43.0	1024.9	2630	43.0	1024.9	1683	43.0	1024.9	4129
43.0	1067.9	2630	43.0	1067.9	1683	43.0	1067.9	4129
43.0	1110.9	2630	43.0	1110.9	1683	43.0	1110.9	4129
43.0	1153.9	2630	43.0	1153.9	1683	43.0	1153.9	4129
43.0	1196.9	2630	43.0	1196.9	1683	43.0	1196.9	4129
43.0	1239.9	2630	43.0	1239.9	1683	43.0	1239.9	4129
45.0	1284.9	3060	45.0	1284.9	1958	45.0	1284.9	4804
45.0	1329.9	3060	45.0	1329.9	1958	45.0	1329.9	4804
45.0	1374.9	3060	45.0	1374.9	1958	45.0	1374.9	4804
45.0	1419.9	3060	45.0	1419.9	1958	45.0	1419.9	4804
45.0	1464.9	3060	45.0	1464.9	1958	45.0	1464.9	4804
45.0	1509.9	3060	45.0	1509.9	1958	45.0	1509.9	4804
45.0	1554.9	3060	45.0	1554.9	1958	45.0	1554.9	4804
45.0	1599.9	3060	45.0	1599.9	1958	45.0	1599.9	4804
45.0	1644.9	3060	45.0	1644.9	1958	45.0	1644.9	4804
45.0	1689.9	3060	45.0	1689.9	1958	45.0	1689.9	4804
3280.8	4970.7	9285	3280.8	4970.7	9285	3280.8	4970.7	9285

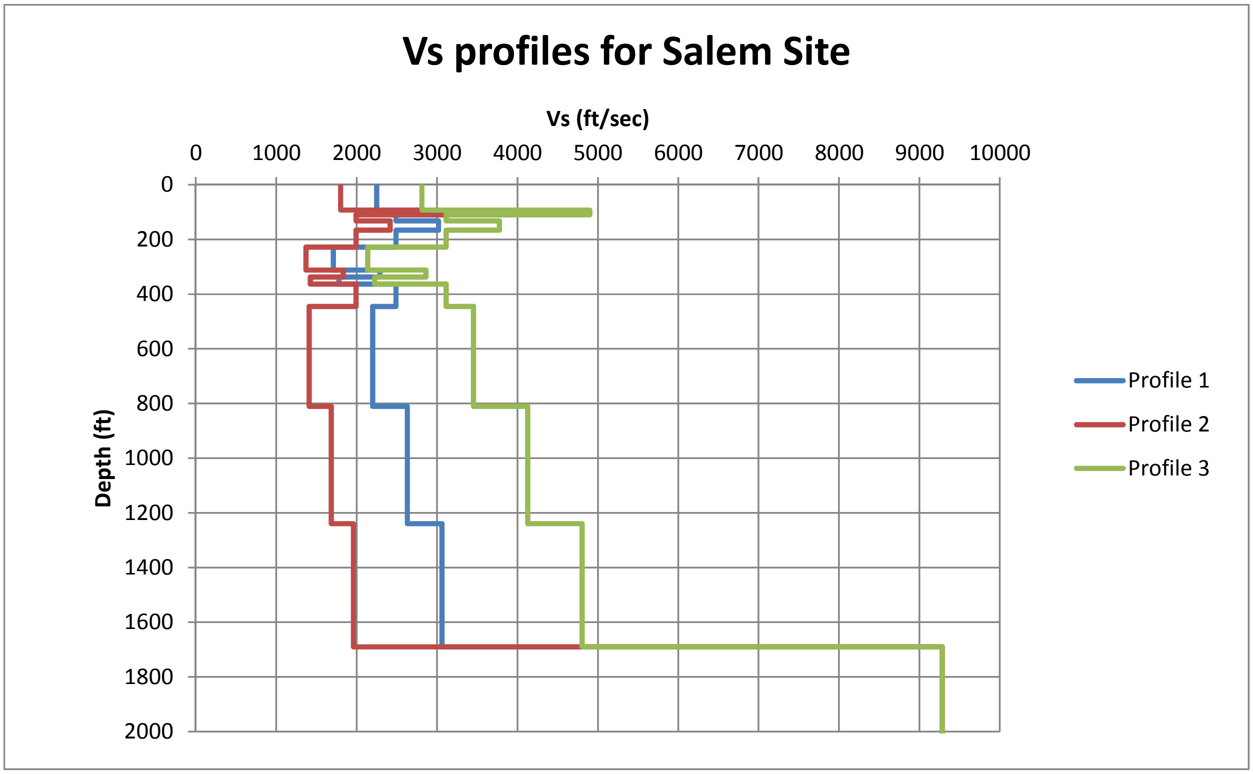


Figure 2-1. Shear-wave velocity profiles for Salem NPP site.

2.3.2.2 Shear Modulus and Damping Curves

No site-specific nonlinear dynamic material properties were available for the soils at the SGS site. The firm soil material over the upper 500 ft was assumed to have behavior that could be modeled with either EPRI cohesionless soil or Peninsular Range G/G_{max} and hysteretic damping curves [2]. Consistent with the SPID [2], the EPRI soil curves (model M1) were considered to be appropriate to represent the more nonlinear response likely to occur in the materials at this site. The Peninsular Range (PR) curves [2] for soils (model M2) were assumed to represent an equally plausible alternative more linear response across loading level.

2.3.2.3 Kappa

Kappa is profile damping contributed by both intrinsic hysteretic damping as well as scattering due to wave propagation in heterogeneous material. Base-case kappa estimates were determined using Section B-5.1.3.1 of the SPID [2] for sites with less than 3,000 ft of soil. For soil sites with depths less than 3,000 ft to hard rock, a mean base-case kappa may be estimated based on total soil thickness with the addition of the hard basement rock value of 0.006 s conditioned with an upper bound of 0.040 s [2]. For the SGS site, with about 1,700 ft of soil the total kappa value was 0.037 s (Table 2-3). Epistemic uncertainty in profile damping (kappa) was considered to be accommodated at design loading levels by the two sets of G/G_{max} and hysteretic damping curves.

Table 2-3
Kappa Values and Weights Used for Site Response Analyses

Velocity Profile	Kappa (s)
P1	0.037
P2	0.037
P3	0.037
Velocity Profile	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 Profile*s

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 SGS site, random shear wave velocity profiles were developed from the base case profiles shown in Figure 2-1. Consistent with the discussion in Appendix B of the SPID [2], 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 [12] 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 [2], 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.

2.3.4 Input Spectra

Consistent with the guidance in Appendix B of the SPID [2], 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 SGS site were the same as those identified in Tables B-4, B-5, B-6 and B-7 of the SPID [2] as appropriate for typical CEUS sites.

2.3.5 Methodology

To perform the site response analyses for the SGS site, 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 [2]. The guidance contained in Appendix B of the SPID [2] 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 SGS site.

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 [2] a minimum median amplification value of 0.5 was employed in the present analysis. Figure 2-2 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 soil 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 SGS soil site, Figure 2-3 shows the corresponding amplification factors developed with Peninsular Range G/G_{max} and hysteretic damping curves for soil (model M2). Figures 2-2 and Figure 2-3 respectively show only a relatively minor difference for the 0.5g loading level and below. Above about the 0.5 g loading level, the differences increase mainly in frequencies above 10 Hz to 20 Hz.

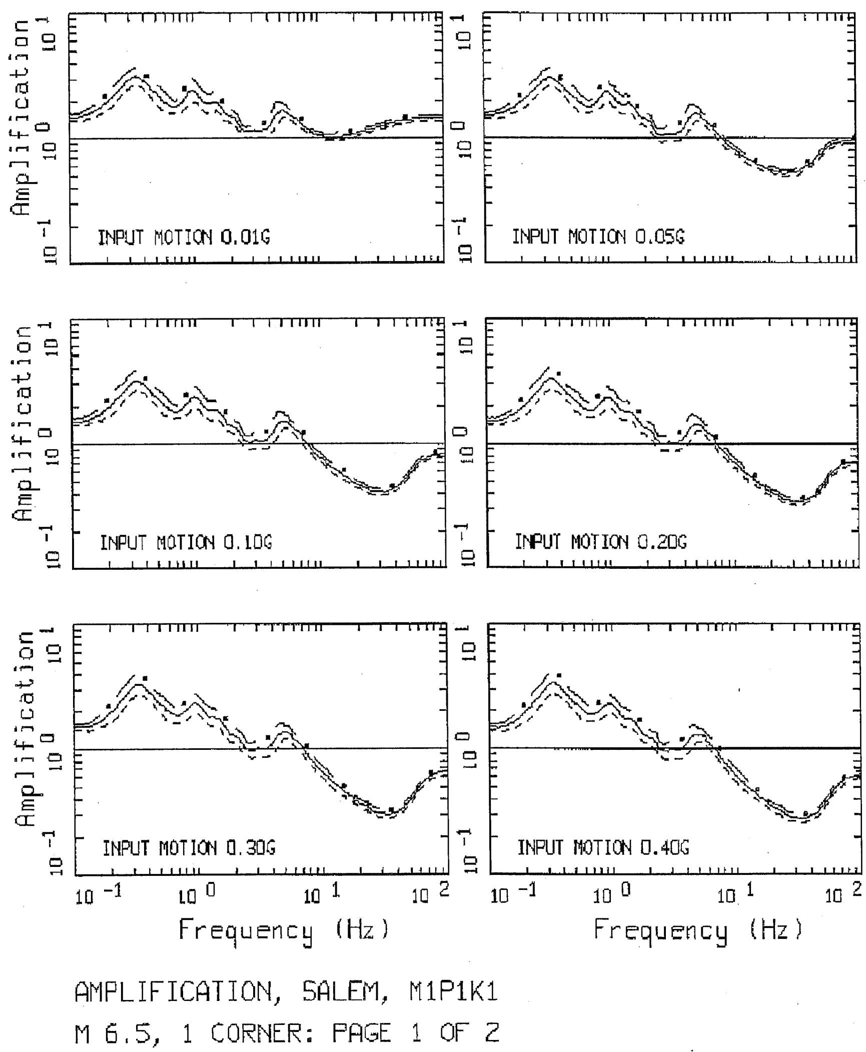
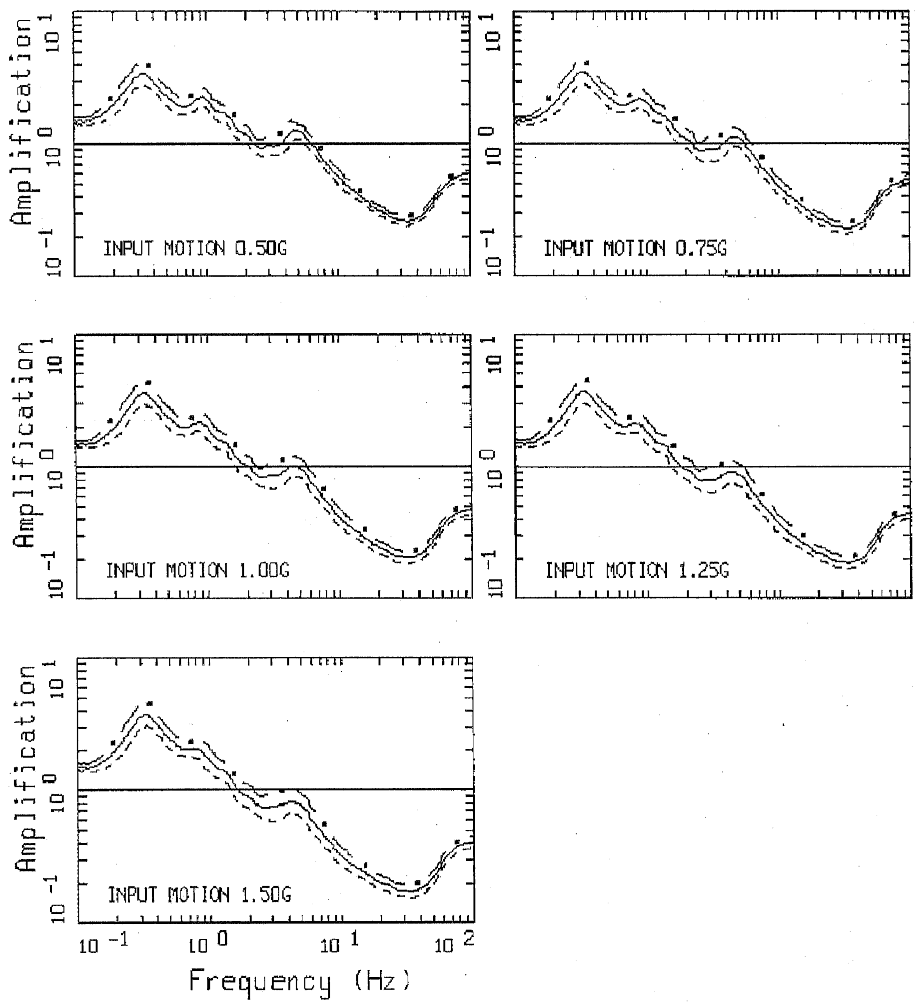


Figure 2-2. Amplification factors (5% damping pseudo absolute acceleration spectra) developed for the mean base-case profile (P1), EPRI soil and rock modulus reduction and hysteretic damping curves (model M1), and base-case kappa at eleven loading levels of hard rock median peak acceleration values from 0.01g to 1.50g. **M** 6.5 and single-corner source model [2].



AMPLIFICATION, SALEM, M1P1K1
M 6.5, 1 CORNER: PAGE 2 OF 2

Figure 2-2. (cont.)

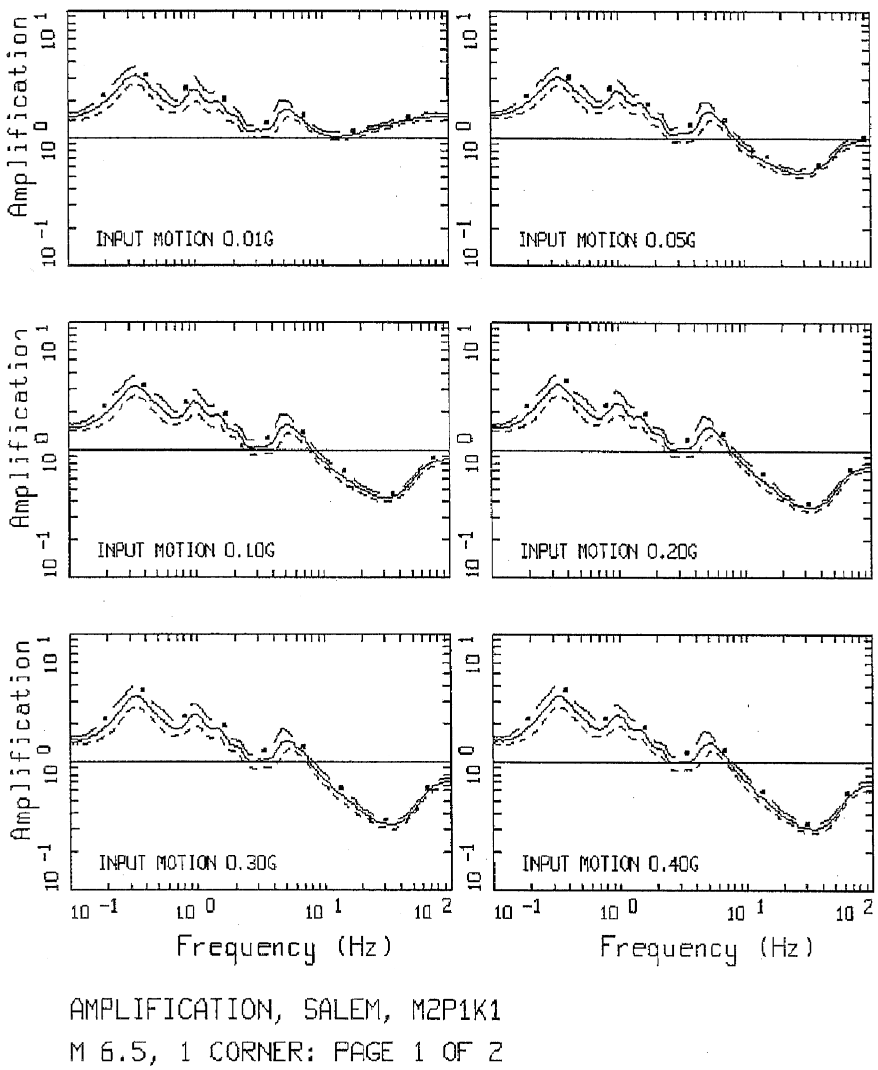


Figure 2-3. Amplification factors (5% damping pseudo absolute acceleration spectra) developed for the mean base-case profile (P1), Peninsular Range modulus reduction and hysteretic damping curves for soil and linear site response for rock (model M2), and base-case kappa at eleven loading levels of hard rock median peak acceleration values from 0.01g to 1.50g. **M** 6.5 and single-corner source model [2].

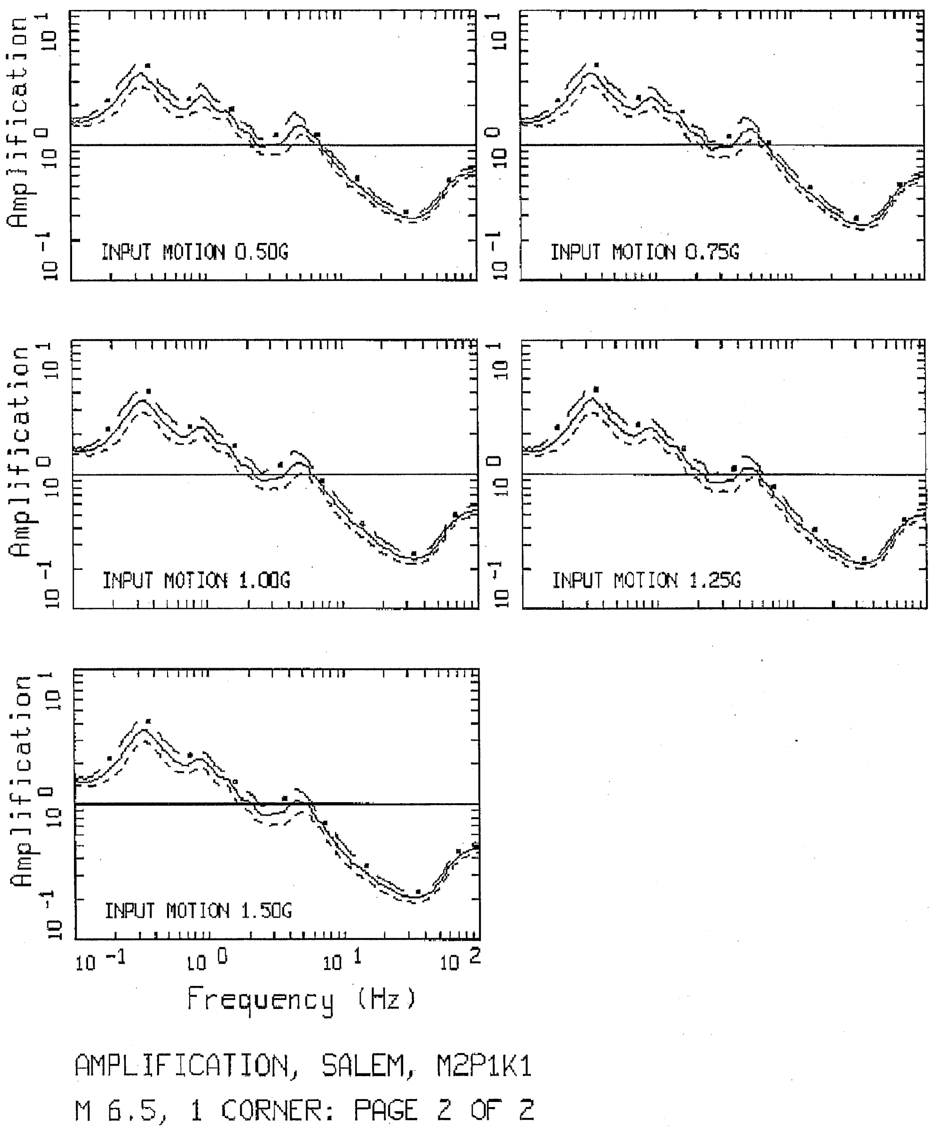


Figure 2-3. (cont.)

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 [2]. 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 SGS are shown in Figure 2-4 for the seven oscillator frequencies for which the ground motion model is defined. Tabulated values of control point hazard curves are provided in Appendix A.

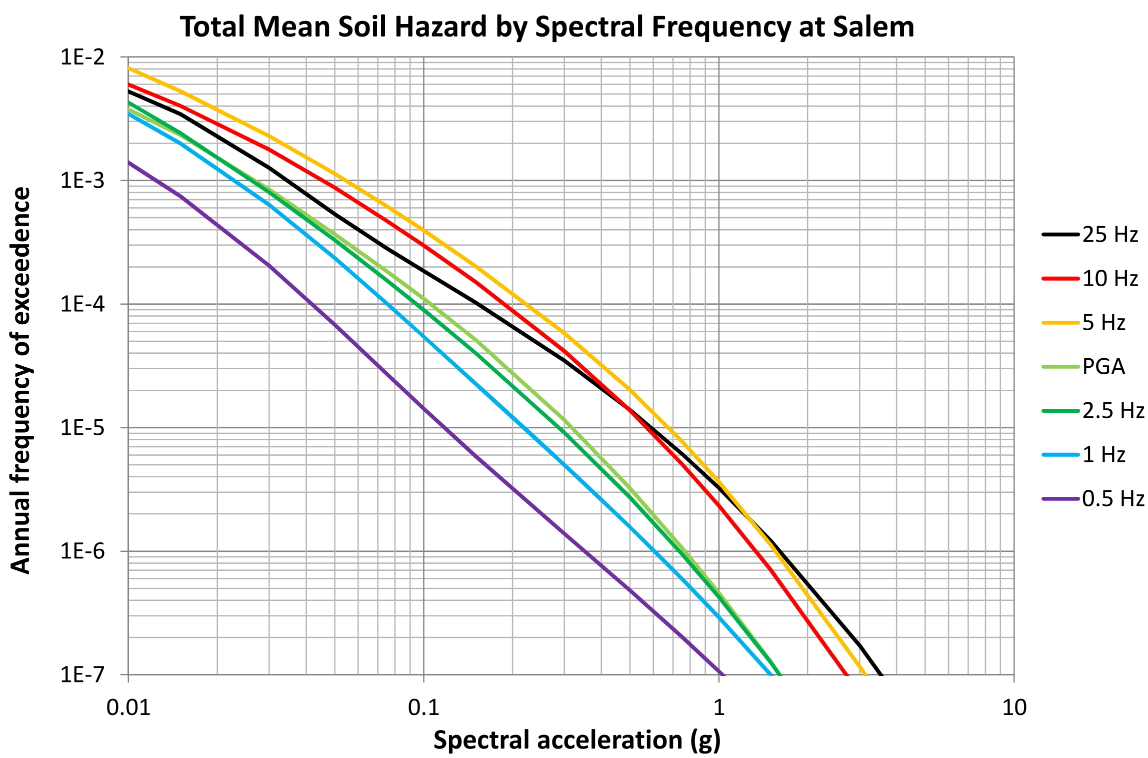


Figure 2-4. Control Point Mean Hazard Curves For Spectral Frequencies of 0.5, 1, 2.5, 5, 10, 25 and 100 Hz at SGS

2.4 Ground Motion Response Spectrum

The control point hazard curves described above have been used to develop uniform hazard response spectra (UHRS) and the ground motion response spectrum (GMRS). The UHRS were obtained through linear interpolation in log-log space to estimate the spectral acceleration at each oscillator 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 [8]. Figure 2-5 and Table 2-4 show the 5%-damped UHRS and GMRS spectral accelerations at the control point.

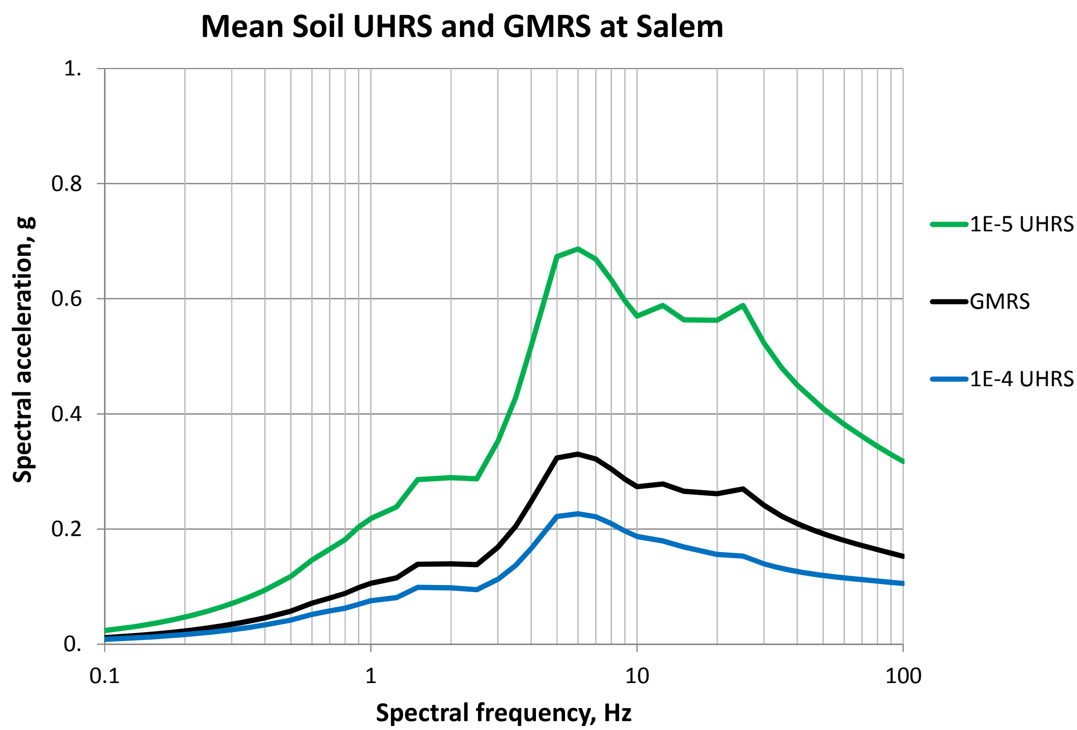


Figure 2-5. Plots of 10^{-4} and 10^{-5} Uniform Hazard Spectra and GMRS at Control Point for SGS (5%-damped response spectra).

Table 2-4. UHRS and GMRS for Salem.

Freq. (Hz)	10^{-4} UHRS (g)	10^{-5} UHRS (g)	GMRS (g)
100	0.105	0.317	0.153
90	0.107	0.330	0.158
80	0.109	0.344	0.164
70	0.112	0.361	0.171
60	0.115	0.382	0.180
50	0.119	0.409	0.192
40	0.126	0.450	0.209
35	0.132	0.480	0.222
30	0.140	0.523	0.241
25	0.153	0.588	0.270
20	0.156	0.563	0.261
15	0.169	0.563	0.266
12.5	0.179	0.588	0.278
10	0.187	0.570	0.274
9	0.196	0.596	0.287
8	0.0209	0.633	0.304
7	0.221	0.669	0.322
6	0.227	0.686	0.330
5	0.222	0.673	0.324
4	0.166	0.518	0.248
3.5	0.137	0.429	0.205
3	0.113	0.353	0.168
2.5	0.0946	0.287	0.138
2	0.0979	0.289	0.140
1.5	0.0986	0.286	0.139
1.25	0.0810	0.239	0.115
1	0.0754	0.218	0.106
0.9	0.0693	0.203	0.0984
0.8	0.0623	0.182	0.0880
0.7	0.0576	0.165	0.0801
0.6	0.0514	0.146	0.0711
0.5	0.0418	0.118	0.0574
0.4	0.0334	0.0941	0.0459
0.35	0.0293	0.0823	0.0402
0.3	0.0251	0.0706	0.0344
0.25	0.0209	0.0588	0.0287
0.2	0.0167	0.0470	0.0229
0.15	0.0125	0.0353	0.0172
0.125	0.0104	0.0294	0.0143
0.1	0.00836	0.0235	0.0115

3.0 Plant Design Basis and Beyond Design Basis Evaluation Ground Motion

The design basis horizontal SSE for SGS is identified in the Updated Final Safety Analysis Report (UFSAR) [5] Figure 2.5-12. The SGS Seismic Category I structures are evaluated using dynamic analysis with a time history input that is conservative relative to the design basis SSE response spectrum (see UFSAR [5] Figure 3.7-2). The SSE response spectra are discussed further below. Response spectra at different floors in the structures were derived from the conservative time history and used in the design of Seismic Category I structures, mechanical and electrical equipment, piping, and their supports.

An evaluation for beyond design basis (BDB) ground motions was performed in the Individual Plant Examination of External Events (IPEEE). The IPEEE capacity response spectrum is included below for screening purposes.

3.1 SSE Description of Spectral Shape

The following discussion of the SSE spectral shape is taken from Section 2.5.2 of the UFSAR [5]. The SSE for Salem was developed through an evaluation of the maximum earthquake potential for the region surrounding the site. Considering the historic seismicity of the site region, PSEG Nuclear determined that the maximum potential earthquake might be either an Intensity VII (Modified Mercalli Scale) event along the Fall Zone at its closest approach to the site or an Intensity VI event very near the site. Because of the uncertainties involved in associating regional activity with specific geologic structures, the maximum potential earthquake was specified as being equivalent to the Intensity VII 1871 Wilmington, Delaware earthquake occurring near the site.

The SSE is defined in terms of a PGA and a design response spectrum. Considering a site intensity of VII, a PGA of 0.10 g at the foundation level was estimated. For additional conservatism, this peak ground acceleration was increased to 0.20 g at the foundation level for the SSE. The 5% damped horizontal SSE is shown in Table 3-1.

Table 3-1. SSE for Salem (UFSAR [5], Figure 2.5-12)

Freq. (Hz)	SA (g)
100	0.200
40	0.200
30	0.200
18.0	0.200
15.4	0.204
10.5	0.224
8.9	0.240
5.5	0.308
4.7	0.322
3.8	0.324
2.7	0.299
2.1	0.270
1.4	0.211
1.2	0.190
1.0	0.167
0.31	0.062
0.21	0.039
0.17	0.028
0.14	0.021
0.01	0.0002

3.2 Control Point Elevation

The SSE control point elevation is defined at the foundation level which is at a depth of 71 ft below grade ([11] and Section 2.5.2.10 of [5]).

3.3 IPEEE Description and Capacity Response Spectrum

The Individual Plant Examination of External Events (IPEEE) for SGS was performed using methods identified in NUREG-1407 [13]. A Seismic Probabilistic Risk Assessment (SPRA) was taken to identify any potential seismic vulnerabilities at Salem. The SPRA techniques include consideration of a Seismic Hazard Analysis, a Seismic Fragility Assignment, a Seismic Systems Analysis, and quantification of the seismically induced Core Damage Frequency (CDF). Additional IPEEE related seismic analyses at SGS evaluate other seismic vulnerabilities through the evaluation of human interactions and recovery actions under seismic conditions, relay chatter during a seismic event, soil seismic liquefaction and slope stability effects, and containment seismic performance.

The IPEEE Adequacy Determination according to SPID [2] Section 3.3.1 is included as Appendix B. The results of the review have shown that the IPEEE is adequate to support screening of the updated seismic hazard for SGS. The review also concluded that the risk insights obtained from the IPEEE are still valid under the current plant configuration.

The Full Scope IPEEE detailed review of relay chatter required in SPID [2] Section 3.3.1 has not been completed. PSEG Nuclear intends to complete the relay chatter review consistent with NEI letter to NRC dated October 3, 2013 [41] on the same schedule as the High Frequency Confirmation as proposed in the NEI letter to NRC dated April 9, 2013 [42] and accepted in NRC's response dated May 7, 2013 [15].

The SPRA was performed based on the seismic hazard curve developed by Lawrence Livermore National Laboratory. The total seismic core damage frequency (CDF) for SGS was 9.5×10^{-6} per year. The plant-level high confidence of low probability of failure (HCLPF) for SGS calculated from the IPEEE determined CDF is 0.27 g. The IPEEE Probabilistic Seismic Response Analyses for SGS structures was performed using the EPRI Uniform Hazard Spectrum shape. Accordingly, the 5% damped horizontal IPEEE HCLPF spectrum (IHS) is estimated using the EPRI spectral shape anchored at the plant level HCLPF. The IHS for SGS is shown in Table 3-2. The SSE and IHS are shown in Figure 3-1.

Table 3-2. SGS IHS (See Appendix B)

Freq (Hz)	IHS
1.0	0.072
2.5	0.23
5.0	0.37
10	0.47
25	0.39
40	0.27
100	0.27

3.4 Conservative SSE Time History

The SGS Category I structures are evaluated using dynamic analysis with a time history input that is conservative relative to the design basis SSE response spectrum (UFSAR [5] Figure 3.7-2). Response spectra at different floors in the structures were derived from the conservative time history and used in the design of Category I (seismic) mechanical and electrical equipment, piping, and their supports (UFSAR [5] Section 3.7.2.5). The 5% damped response spectrum determined from the SSE time history is shown on Figure 3-1. The 5% damped SSE and IHS response spectra are also included.

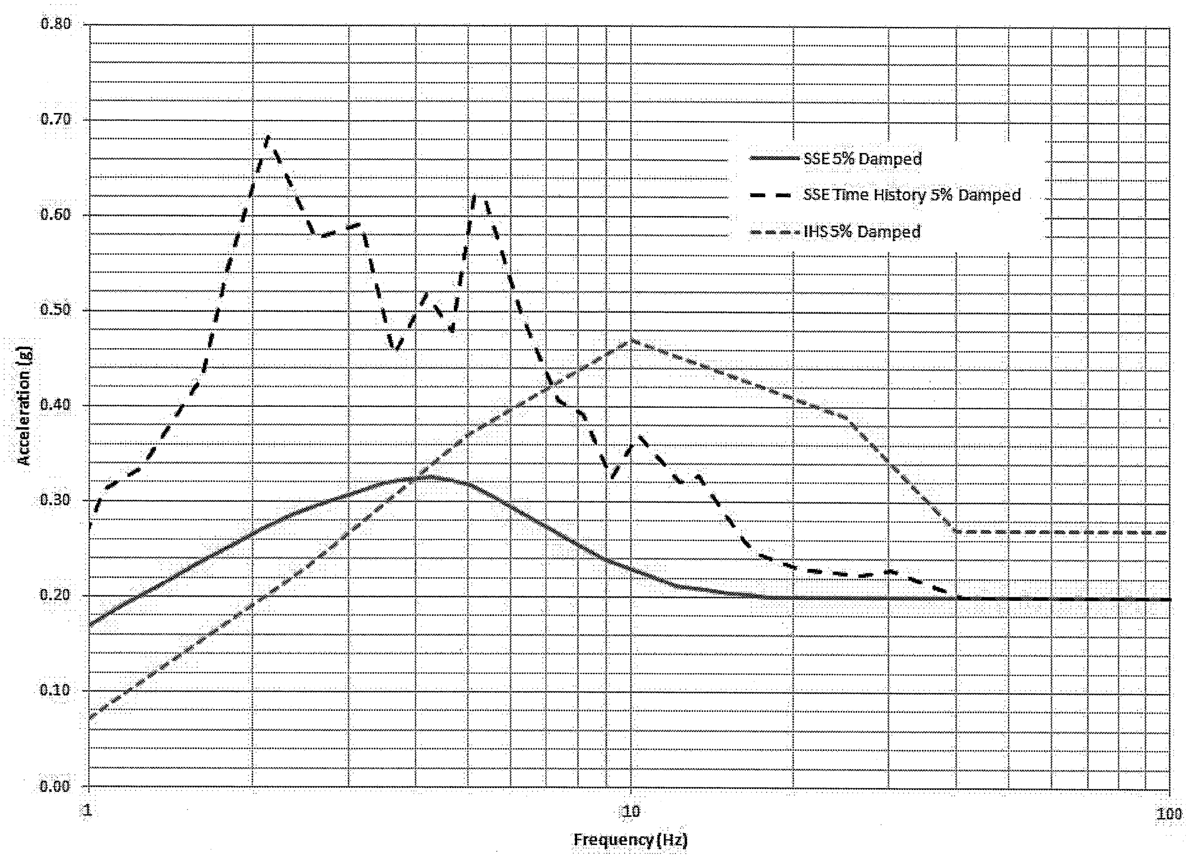


Figure 3-1. Comparison of SSE and IHS (5%-damped response spectra).

4.0 Screening Evaluation

In accordance with SPID [2] 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 design basis SSE above about 5 Hz. In the 1 to 10 Hz part of the response spectrum, the IHS exceeds the GMRS for all frequencies above 1.5 Hz. Per SPID [2] Section 3.2.1.1, since the GMRS does not exceed the Low Hazard Threshold (LHT) of 0.4 g, low frequency exceedances below 2.5 Hz do not require a full risk evaluation. Instead, it is sufficient to identify all safety significant structures, systems, and components at frequencies below the exceedance frequency and demonstrate their capacities are above the GMRS. All low frequency components demonstrate capacity above the GMRS based on the fact that they satisfy the design basis SSE response spectrum which is significantly higher than the GMRS at all frequencies below 2.5 Hz. Furthermore, in the 1 to 10 Hz part of the response spectrum, the conservative design time history SSE response spectrum used for design of Category I structures, systems, and components exceeds the GMRS. Therefore, a risk evaluation is not warranted and will not be performed.

4.2 High Frequency Screening (Greater than 10 Hz)

Above 10 Hz, the IHS exceeds the GMRS. However, the Full Scope IPEEE detailed review of relay chatter required in SPID [2] Section 3.3.1 has not been completed. PSEG Nuclear intends to complete the relay chatter review consistent with NEI letter to NRC dated October 3, 2013 [41] on the same schedule as the High Frequency Confirmation as proposed in the NEI letter to NRC dated April 9, 2013 [42] and accepted in NRC's response dated May 7, 2013 [15]. High Frequency Confirmation per SPID [2] Section 3.4 will only be performed if the relay chatter review is not successful in demonstrating relay adequacy based on the GMRS.

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, Salem screens in for a spent fuel pool evaluation.

5.0 Interim Actions

Section 4.1 and 4.2 above indicate that no further seismic risk evaluations are warranted. However since the GMRS exceeds the design basis SSE between 5 and 10 Hz, the expedited seismic evaluation process (ESEP) described in [3] will be performed as proposed in a letter to NRC dated April 9, 2013 [14] and agreed to by the NRC in a letter dated May 7, 2013 [15].

Consistent with NRC letter dated February 20, 2014 [16], the seismic hazard reevaluations presented herein are distinct from the current design and licensing bases of SGS. Therefore, the results do not call into question the operability or functionality of Systems, Structures, 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, NEI letter dated March 12, 2014 [17], 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 provided in [18]:

> 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.

SGS is included in the [17] risk estimates. Using the methodology described in the NEI letter [17], all plants were shown to be below 10^{-4}/year; thus, the above conclusions apply.

PSEG Nuclear recently performed seismic walkdowns at Salem in accordance with NTTF Recommendation 2.3. The potentially adverse seismic conditions identified during the walkdowns have been assessed and judged not to involve a significant adverse seismic condition and thus, immediate actions were not required. These issues were entered into the Corrective Action Program and have since been either been corrected, determined to be acceptable as-is, or identified for condition-based monitoring until corrective actions are completed as part of Service Water Intake Structure material condition improvements [19].

6.0 Conclusions

In accordance with the 50.54(f) request for information [1], a seismic hazard and screening evaluation was performed for SGS. A GMRS was developed solely for purpose of screening for additional evaluations in accordance with the SPID [2]. Based on the results of the screening evaluation, the plant screens in for a Spent Fuel Pool evaluation.

In addition, a Full Scope IPEEE detailed review of relay chatter required in SPID [2] Section 3.3.1 has not been completed. PSEG Nuclear intends to complete the relay chatter review consistent with NEI letter to NRC dated October 3, 2013 [41] on the same schedule as the High Frequency Confirmation as proposed in the NEI letter to NRC dated April 9, 2013 [42] and accepted in NRC's response dated May 7, 2013 [15].

7.0 References

1. U.S. Nuclear Regulatory Commission Letter from E. Leeds and M. Johnson 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 (ML12053A340).

2. 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," Electric Power Research Institute, 2013.

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

4. Lettis Consultants International, "Salem Seismic Hazard and Screening Report," November 26, 2013.

5. Salem Generating Station Unit 1 and Unit 2, Updated Final Safety Analysis Report (UFSAR), Revision 26.

6. U.S. Nuclear Regulatory Commission Report, NUREG-2115; EPRI Report 1021097, 6 Volumes; DOE Report# DOE/NE-0140; “Central and Eastern United States Seismic Source Characterization for Nuclear Facilities,” 2012 (ML12048A776).

7. EPRI Report 3002000717, “EPRI (2004, 2006) Ground-Motion Model (GMM) Review Project,” Electric Power Research Institute, 2013.

8. U.S. Nuclear Regulatory Commission Regulatory Guide 1.208, “A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion,” 2007.

9. PSEG Nuclear Letter from J. Mallon to J. Hamel (EPRI), “PSEG Nuclear LLC’s Response to EPRI’s Data Request for Site Amplification Calculations,” July 31, 2012.

10. PSEG Nuclear Early Site Permit Application, Site Safety Analysis Report, Revision 2 (ML13098A775).

11. PSEG Nuclear Letter from J. Perry to U.S. Nuclear Regulatory Commission, “Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding the Seismic Aspects of Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident - Base Case Velocity Profiles With Supporting Subsurface Materials and Properties,” September 10, 2013 (ML13253A391).

12. Pacific Engineering and Analysis Report 94PJ20, “Description and Validation of the Stochastic Ground Motion Model,” 1996; Appendix C, “Probabilistic Models of Site Velocity Profiles for Generic and Site-Specific Ground-Motion Amplification Studies,” by W. Toro.

13. U.S. Nuclear Regulatory Commission Report, NUREG-1407, “Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities,” 1991 (ML063550238).

14. Nuclear Energy Institute Letter from A. Pietrangelo to D. Skeen (USNRC), “Proposed Path Forward for NTTF Recommendation 2.1: Seismic Reevaluations,” April 9, 2013 (ML13101A379).

15. U.S. Nuclear Regulatory Commission Letter from 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 Reevaluations,” May 7, 2013 (ML13106A331).

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Appendix A – Additional Seismic Hazard Curve Data

Tables A-1 through A-7 provide the mean values and fractile values for the seismic hazard curves at each frequency provided in Figure 2-4.

Table A-8 provides the amplification functions for Salem.

Tables A-9 and A-10 are tabular versions of the typical amplification factors provided in Figures 2-2 and 2-3. Values are provided for two input motion levels at approximately 10^{-4} and 10^{-5} mean annual frequency exceedances. These factors are unverified and are provided for information only. The figures should be considered the governing information.

All information provided in this Appendix is from [4] unless otherwise noted.

Table A-1. Mean and fractile seismic hazard curves for 100 Hz at Salem

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	3.29E-02	1.53E-02	2.60E-02	3.37E-02	4.01E-02	4.50E-02
0.001	2.41E-02	9.93E-03	1.77E-02	2.39E-02	3.14E-02	3.57E-02
0.005	7.33E-03	3.23E-03	4.63E-03	6.73E-03	9.51E-03	1.44E-02
0.01	3.76E-03	1.53E-03	2.16E-03	3.37E-03	4.98E-03	8.12E-03
0.015	2.33E-03	8.47E-04	1.23E-03	2.01E-03	3.19E-03	5.50E-03
0.03	8.46E-04	2.13E-04	3.33E-04	6.45E-04	1.29E-03	2.32E-03
0.05	3.67E-04	6.09E-05	1.15E-04	2.53E-04	5.91E-04	1.11E-03
0.075	1.84E-04	2.39E-05	5.12E-05	1.21E-04	3.01E-04	5.83E-04
0.1	1.11E-04	1.21E-05	2.96E-05	7.13E-05	1.79E-04	3.52E-04
0.15	5.13E-05	4.70E-06	1.32E-05	3.33E-05	8.23E-05	1.64E-04
0.3	1.15E-05	7.45E-07	2.64E-06	7.45E-06	1.84E-05	3.68E-05
0.5	3.24E-06	1.51E-07	6.26E-07	2.04E-06	5.27E-06	1.05E-05
0.75	1.07E-06	3.28E-08	1.62E-07	6.36E-07	1.74E-06	3.68E-06
1.	4.58E-07	9.93E-09	5.35E-08	2.53E-07	7.55E-07	1.69E-06
1.5	1.27E-07	1.51E-09	8.98E-09	5.91E-08	2.04E-07	5.20E-07
3.	1.02E-08	1.01E-10	2.88E-10	2.96E-09	1.44E-08	5.12E-08
5.	1.15E-09	6.09E-11	1.01E-10	2.64E-10	1.46E-09	6.54E-09
7.5	1.66E-10	5.05E-11	6.09E-11	1.01E-10	2.57E-10	1.04E-09
10.	3.80E-11	5.05E-11	5.05E-11	1.01E-10	1.13E-10	2.84E-10

Table A-2. Mean and fractile seismic hazard curves for 25 Hz at Salem

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	3.41E-02	1.84E-02	2.84E-02	3.42E-02	4.07E-02	4.50E-02
0.001	2.58E-02	1.25E-02	1.98E-02	2.53E-02	3.23E-02	3.79E-02
0.005	9.23E-03	4.37E-03	6.09E-03	8.60E-03	1.18E-02	1.77E-02
0.01	5.26E-03	2.39E-03	3.28E-03	4.83E-03	6.83E-03	1.05E-02
0.015	3.45E-03	1.46E-03	2.04E-03	3.14E-03	4.63E-03	7.03E-03
0.03	1.26E-03	4.43E-04	6.45E-04	1.10E-03	1.79E-03	2.76E-03
0.05	5.34E-04	1.40E-04	2.25E-04	4.31E-04	8.23E-04	1.29E-03
0.075	2.82E-04	5.66E-05	1.05E-04	2.22E-04	4.50E-04	7.45E-04
0.1	1.85E-04	3.01E-05	6.64E-05	1.42E-04	3.01E-04	4.98E-04
0.15	1.03E-04	1.46E-05	3.57E-05	7.89E-05	1.69E-04	2.80E-04
0.3	3.48E-05	3.90E-06	1.18E-05	2.72E-05	5.66E-05	9.24E-05
0.5	1.39E-05	1.18E-06	4.43E-06	1.07E-05	2.29E-05	3.79E-05
0.75	6.12E-06	4.56E-07	1.84E-06	4.63E-06	1.01E-05	1.69E-05
1.	3.24E-06	2.22E-07	9.11E-07	2.42E-06	5.35E-06	9.11E-06
1.5	1.21E-06	6.83E-08	2.92E-07	8.72E-07	2.04E-06	3.57E-06
3.	1.72E-07	5.66E-09	2.64E-08	1.07E-07	3.01E-07	5.75E-07
5.	3.16E-08	7.13E-10	3.01E-09	1.60E-08	5.42E-08	1.20E-07
7.5	6.96E-09	1.46E-10	4.77E-10	2.80E-09	1.16E-08	2.92E-08
10.	2.17E-09	1.01E-10	1.60E-10	7.66E-10	3.57E-09	9.51E-09

Table A-3. Mean and fractile seismic hazard curves for 10 Hz at Salem

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	3.81E-02	2.76E-02	3.28E-02	3.84E-02	4.37E-02	4.77E-02
0.001	3.04E-02	1.90E-02	2.46E-02	3.01E-02	3.63E-02	4.07E-02
0.005	1.09E-02	5.66E-03	7.66E-03	1.05E-02	1.40E-02	1.82E-02
0.01	5.98E-03	3.01E-03	3.95E-03	5.66E-03	7.66E-03	1.05E-02
0.015	4.01E-03	1.92E-03	2.57E-03	3.79E-03	5.27E-03	7.23E-03
0.03	1.78E-03	7.45E-04	1.04E-03	1.64E-03	2.46E-03	3.37E-03
0.05	8.75E-04	3.05E-04	4.43E-04	7.66E-04	1.27E-03	1.79E-03
0.075	4.71E-04	1.32E-04	2.10E-04	3.95E-04	7.23E-04	1.07E-03
0.1	2.98E-04	7.03E-05	1.20E-04	2.42E-04	4.70E-04	7.13E-04
0.15	1.51E-04	2.76E-05	5.35E-05	1.18E-04	2.46E-04	3.84E-04
0.3	4.16E-05	5.05E-06	1.27E-05	3.14E-05	6.93E-05	1.15E-04
0.5	1.38E-05	1.21E-06	3.73E-06	9.93E-06	2.32E-05	3.95E-05
0.75	5.08E-06	3.33E-07	1.20E-06	3.47E-06	8.60E-06	1.51E-05
1.	2.33E-06	1.25E-07	4.90E-07	1.55E-06	4.01E-06	7.13E-06
1.5	7.05E-07	2.80E-08	1.18E-07	4.25E-07	1.23E-06	2.29E-06
3.	7.05E-08	1.21E-09	6.09E-09	3.42E-08	1.23E-07	2.68E-07
5.	1.07E-08	1.32E-10	4.98E-10	3.79E-09	1.77E-08	4.56E-08
7.5	2.11E-09	8.85E-11	1.13E-10	5.50E-10	3.23E-09	9.79E-09
10.	6.18E-10	6.09E-11	9.11E-11	1.72E-10	9.24E-10	3.01E-09

Table A-4. Mean and fractile seismic hazard curves for 5 Hz at Salem

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	4.16E-02	3.28E-02	3.63E-02	4.13E-02	4.70E-02	5.12E-02
0.001	3.61E-02	2.46E-02	2.96E-02	3.63E-02	4.25E-02	4.70E-02
0.005	1.52E-02	7.45E-03	1.05E-02	1.49E-02	2.01E-02	2.35E-02
0.01	8.12E-03	3.79E-03	5.27E-03	7.77E-03	1.11E-02	1.34E-02
0.015	5.29E-03	2.46E-03	3.37E-03	5.05E-03	7.23E-03	8.98E-03
0.03	2.29E-03	1.01E-03	1.38E-03	2.16E-03	3.14E-03	4.07E-03
0.05	1.13E-03	4.43E-04	6.36E-04	1.05E-03	1.62E-03	2.13E-03
0.075	6.20E-04	2.13E-04	3.14E-04	5.58E-04	9.24E-04	1.25E-03
0.1	3.95E-04	1.18E-04	1.84E-04	3.42E-04	6.00E-04	8.47E-04
0.15	2.02E-04	4.90E-05	8.35E-05	1.69E-04	3.23E-04	4.70E-04
0.3	5.80E-05	9.37E-06	1.98E-05	4.56E-05	9.51E-05	1.51E-04
0.5	2.02E-05	2.35E-06	6.09E-06	1.53E-05	3.37E-05	5.50E-05
0.75	7.75E-06	6.73E-07	2.04E-06	5.58E-06	1.31E-05	2.22E-05
1.	3.64E-06	2.49E-07	8.12E-07	2.46E-06	6.17E-06	1.10E-05
1.5	1.12E-06	4.50E-08	1.74E-07	6.64E-07	1.98E-06	3.73E-06
3.	1.16E-07	1.25E-09	5.58E-09	4.37E-08	2.04E-07	4.63E-07
5.	1.85E-08	1.08E-10	2.92E-10	4.01E-09	2.92E-08	8.35E-08
7.5	3.90E-09	6.09E-11	1.01E-10	5.05E-10	5.27E-09	1.82E-08
10.	1.22E-09	5.05E-11	8.72E-11	1.51E-10	1.40E-09	5.58E-09

Table A-5. Mean and fractile seismic hazard curves for 2.5 Hz at Salem

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	3.82E-02	2.84E-02	3.23E-02	3.84E-02	4.43E-02	4.83E-02
0.001	3.05E-02	1.95E-02	2.35E-02	3.01E-02	3.79E-02	4.19E-02
0.005	9.73E-03	4.63E-03	6.36E-03	9.24E-03	1.32E-02	1.62E-02
0.01	4.28E-03	1.90E-03	2.64E-03	4.01E-03	5.91E-03	7.66E-03
0.015	2.42E-03	1.04E-03	1.44E-03	2.25E-03	3.37E-03	4.50E-03
0.03	8.01E-04	3.01E-04	4.37E-04	7.23E-04	1.16E-03	1.57E-03
0.05	3.29E-04	1.04E-04	1.60E-04	2.88E-04	4.98E-04	6.93E-04
0.075	1.56E-04	4.07E-05	6.64E-05	1.31E-04	2.42E-04	3.57E-04
0.1	8.98E-05	1.98E-05	3.42E-05	7.23E-05	1.44E-04	2.19E-04
0.15	4.01E-05	6.64E-06	1.32E-05	3.05E-05	6.54E-05	1.07E-04
0.3	9.11E-06	8.35E-07	2.25E-06	6.17E-06	1.53E-05	2.68E-05
0.5	2.72E-06	1.55E-07	5.12E-07	1.69E-06	4.70E-06	8.85E-06
0.75	9.48E-07	3.33E-08	1.31E-07	5.27E-07	1.64E-06	3.28E-06
1.	4.25E-07	9.79E-09	4.37E-08	2.07E-07	7.34E-07	1.57E-06
1.5	1.25E-07	1.49E-09	7.77E-09	4.77E-08	2.16E-07	4.98E-07
3.	1.22E-08	1.07E-10	2.80E-10	2.39E-09	1.82E-08	5.50E-08
5.	1.78E-09	6.09E-11	9.24E-11	2.22E-10	2.07E-09	8.12E-09
7.5	3.39E-10	5.05E-11	6.09E-11	1.01E-10	3.47E-10	1.49E-09
10.	9.65E-11	5.05E-11	5.35E-11	1.01E-10	1.32E-10	4.56E-10

Table A-6. Mean and fractile seismic hazard curves for 1 Hz at Salem

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	3.28E-02	1.98E-02	2.53E-02	3.33E-02	4.01E-02	4.43E-02
0.001	2.44E-02	1.25E-02	1.72E-02	2.42E-02	3.14E-02	3.63E-02
0.005	7.47E-03	2.76E-03	4.37E-03	7.13E-03	1.05E-02	1.32E-02
0.01	3.46E-03	1.04E-03	1.74E-03	3.14E-03	5.12E-03	7.13E-03
0.015	1.99E-03	5.27E-04	8.98E-04	1.72E-03	3.05E-03	4.50E-03
0.03	6.34E-04	1.36E-04	2.46E-04	5.05E-04	9.93E-04	1.62E-03
0.05	2.36E-04	4.37E-05	8.00E-05	1.77E-04	3.84E-04	6.36E-04
0.075	1.01E-04	1.60E-05	3.05E-05	7.23E-05	1.69E-04	2.84E-04
0.1	5.46E-05	7.55E-06	1.49E-05	3.73E-05	9.24E-05	1.60E-04
0.15	2.27E-05	2.46E-06	5.27E-06	1.42E-05	3.90E-05	7.13E-05
0.3	4.98E-06	2.96E-07	8.00E-07	2.64E-06	8.35E-06	1.79E-05
0.5	1.57E-06	5.12E-08	1.77E-07	7.03E-07	2.60E-06	6.09E-06
0.75	5.98E-07	1.11E-08	4.50E-08	2.29E-07	9.65E-07	2.49E-06
1.	2.91E-07	3.47E-09	1.60E-08	9.51E-08	4.56E-07	1.27E-06
1.5	9.98E-08	6.09E-10	3.19E-09	2.39E-08	1.46E-07	4.50E-07
3.	1.34E-08	1.01E-10	1.90E-10	1.51E-09	1.51E-08	6.26E-08
5.	2.57E-09	6.09E-11	9.11E-11	1.95E-10	2.10E-09	1.16E-08
7.5	6.20E-10	5.05E-11	6.09E-11	1.01E-10	4.07E-10	2.53E-09
10.	2.10E-10	5.05E-11	5.91E-11	1.01E-10	1.55E-10	8.12E-10

Table A-7. Mean and fractile seismic hazard curves for 0.5 Hz at Salem

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	2.01E-02	1.13E-02	1.55E-02	1.98E-02	2.46E-02	2.88E-02
0.001	1.29E-02	6.64E-03	9.37E-03	1.27E-02	1.64E-02	2.01E-02
0.005	3.32E-03	9.37E-04	1.60E-03	3.01E-03	4.98E-03	6.83E-03
0.01	1.40E-03	2.68E-04	5.20E-04	1.15E-03	2.29E-03	3.47E-03
0.015	7.45E-04	1.16E-04	2.39E-04	5.58E-04	1.27E-03	2.04E-03
0.03	2.04E-04	2.32E-05	4.98E-05	1.32E-04	3.63E-04	6.26E-04
0.05	6.80E-05	6.17E-06	1.36E-05	3.90E-05	1.27E-04	2.22E-04
0.075	2.72E-05	1.98E-06	4.56E-06	1.40E-05	5.12E-05	9.51E-05
0.1	1.43E-05	8.47E-07	2.07E-06	6.73E-06	2.64E-05	5.27E-05
0.15	5.88E-06	2.39E-07	6.54E-07	2.42E-06	1.02E-05	2.42E-05
0.3	1.38E-06	2.07E-08	8.85E-08	4.25E-07	2.10E-06	6.54E-06
0.5	4.81E-07	2.72E-09	1.74E-08	1.10E-07	6.54E-07	2.46E-06
0.75	2.02E-07	5.12E-10	4.19E-09	3.37E-08	2.46E-07	1.07E-06
1.	1.07E-07	1.95E-10	1.42E-09	1.36E-08	1.16E-07	5.66E-07
1.5	4.12E-08	1.01E-10	3.19E-10	3.42E-09	3.73E-08	2.16E-07
3.	6.77E-09	6.09E-11	1.01E-10	2.84E-10	3.90E-09	3.33E-08
5.	1.49E-09	5.05E-11	6.09E-11	1.01E-10	6.00E-10	6.54E-09
7.5	3.95E-10	5.05E-11	5.75E-11	1.01E-10	1.62E-10	1.49E-09
10.	1.42E-10	5.05E-11	5.05E-11	1.01E-10	1.02E-10	5.20E-10

Table A-8. Amplification Functions for Salem

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	1.27E+00	6.64E-02	1.30E-02	9.93E-01	6.68E-02	1.90E-02	9.68E-01	7.68E-02	2.09E-02	1.48E+00	1.57E-01
4.95E-02	8.76E-01	7.33E-02	1.02E-01	5.00E-01	7.79E-02	9.99E-02	8.09E-01	9.77E-02	8.24E-02	1.42E+00	1.58E-01
9.64E-02	7.58E-01	7.52E-02	2.13E-01	5.00E-01	8.18E-02	1.85E-01	7.67E-01	1.03E-01	1.44E-01	1.38E+00	1.56E-01
1.94E-01	6.66E-01	7.83E-02	4.43E-01	5.00E-01	8.67E-02	3.56E-01	7.17E-01	1.09E-01	2.65E-01	1.33E+00	1.53E-01
2.92E-01	6.17E-01	8.14E-02	6.76E-01	5.00E-01	9.09E-02	5.23E-01	6.79E-01	1.15E-01	3.84E-01	1.28E+00	1.53E-01
3.91E-01	5.84E-01	8.44E-02	9.09E-01	5.00E-01	9.41E-02	6.90E-01	6.47E-01	1.19E-01	5.02E-01	1.24E+00	1.54E-01
4.93E-01	5.58E-01	8.66E-02	1.15E+00	5.00E-01	9.66E-02	8.61E-01	6.19E-01	1.23E-01	6.22E-01	1.20E+00	1.56E-01
7.41E-01	5.09E-01	9.29E-02	1.73E+00	5.00E-01	1.03E-01	1.27E+00	5.59E-01	1.32E-01	9.13E-01	1.12E+00	1.62E-01
1.01E+00	5.00E-01	9.87E-02	2.36E+00	5.00E-01	1.09E-01	1.72E+00	5.07E-01	1.39E-01	1.22E+00	1.04E+00	1.70E-01
1.28E+00	5.00E-01	1.04E-01	3.01E+00	5.00E-01	1.13E-01	2.17E+00	5.00E-01	1.46E-01	1.54E+00	9.70E-01	1.79E-01
1.55E+00	5.00E-01	1.09E-01	3.63E+00	5.00E-01	1.19E-01	2.61E+00	5.00E-01	1.54E-01	1.85E+00	9.15E-01	1.89E-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	1.17E+00	1.19E-01	1.27E-02	2.07E+00	2.18E-01	8.25E-03	2.07E+00	1.67E-01			
7.05E-02	1.11E+00	1.25E-01	3.43E-02	2.03E+00	2.06E-01	1.96E-02	2.06E+00	1.62E-01			
1.18E-01	1.09E+00	1.28E-01	5.51E-02	2.01E+00	1.97E-01	3.02E-02	2.06E+00	1.61E-01			
2.12E-01	1.05E+00	1.33E-01	9.63E-02	1.99E+00	1.87E-01	5.11E-02	2.06E+00	1.61E-01			
3.04E-01	1.02E+00	1.37E-01	1.36E-01	1.96E+00	1.79E-01	7.10E-02	2.07E+00	1.62E-01			
3.94E-01	1.00E+00	1.40E-01	1.75E-01	1.94E+00	1.74E-01	9.06E-02	2.08E+00	1.63E-01			
4.86E-01	9.79E-01	1.43E-01	2.14E-01	1.91E+00	1.70E-01	1.10E-01	2.09E+00	1.65E-01			
7.09E-01	9.34E-01	1.51E-01	3.10E-01	1.86E+00	1.66E-01	1.58E-01	2.10E+00	1.66E-01			
9.47E-01	8.93E-01	1.59E-01	4.12E-01	1.82E+00	1.66E-01	2.09E-01	2.12E+00	1.66E-01			
1.19E+00	8.61E-01	1.66E-01	5.18E-01	1.79E+00	1.67E-01	2.62E-01	2.13E+00	1.66E-01			
1.43E+00	8.41E-01	1.70E-01	6.19E-01	1.75E+00	1.68E-01	3.12E-01	2.14E+00	1.67E-01			

Table A-9. Median AFs and Sigmas for Model 1, Profile 1, for 2 PGA Levels [20].

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	0.141	0.728	0.060	100.0	0.393	0.531	0.089
87.1	0.141	0.711	0.060	87.1	0.394	0.514	0.089
75.9	0.142	0.680	0.060	75.9	0.394	0.486	0.089
66.1	0.142	0.625	0.060	66.1	0.394	0.436	0.089
57.5	0.142	0.536	0.061	57.5	0.395	0.363	0.089
50.1	0.143	0.448	0.061	50.1	0.396	0.298	0.090
43.7	0.144	0.382	0.061	43.7	0.397	0.253	0.090
38.0	0.145	0.351	0.062	38.0	0.399	0.234	0.091
33.1	0.148	0.337	0.063	33.1	0.402	0.227	0.093
28.8	0.152	0.345	0.063	28.8	0.408	0.233	0.096
25.1	0.157	0.354	0.068	25.1	0.415	0.239	0.099
21.9	0.164	0.388	0.064	21.9	0.428	0.263	0.102
19.1	0.174	0.418	0.067	19.1	0.443	0.281	0.107
16.6	0.188	0.470	0.081	16.6	0.468	0.312	0.113
14.5	0.199	0.520	0.097	14.5	0.493	0.349	0.124
12.6	0.214	0.575	0.098	12.6	0.526	0.387	0.140
11.0	0.236	0.649	0.087	11.0	0.571	0.435	0.145
9.5	0.257	0.739	0.098	9.5	0.629	0.506	0.150
8.3	0.276	0.860	0.101	8.3	0.683	0.602	0.147
7.2	0.308	1.025	0.106	7.2	0.749	0.709	0.151
6.3	0.342	1.209	0.108	6.3	0.842	0.856	0.164
5.5	0.375	1.390	0.107	5.5	0.962	1.032	0.157
4.8	0.373	1.410	0.176	4.8	1.012	1.117	0.158
4.2	0.314	1.225	0.213	4.2	0.950	1.087	0.213
3.6	0.262	1.050	0.159	3.6	0.801	0.947	0.208
3.2	0.243	1.034	0.144	3.2	0.715	0.903	0.197
2.8	0.223	1.001	0.131	2.8	0.666	0.891	0.178
2.4	0.222	1.077	0.130	2.4	0.628	0.915	0.147
2.1	0.239	1.275	0.128	2.1	0.659	1.059	0.167
1.8	0.231	1.378	0.135	1.8	0.636	1.149	0.155
1.6	0.245	1.684	0.162	1.6	0.657	1.377	0.195
1.4	0.227	1.814	0.165	1.4	0.658	1.609	0.142
1.2	0.208	1.894	0.169	1.2	0.611	1.707	0.168
1.0	0.219	2.205	0.170	1.0	0.624	1.943	0.177
0.91	0.208	2.297	0.202	0.91	0.637	2.195	0.163
0.79	0.162	1.980	0.172	0.79	0.544	2.089	0.155
0.69	0.133	1.828	0.118	0.69	0.447	1.943	0.135
0.60	0.123	1.938	0.165	0.60	0.399	2.005	0.148
0.52	0.118	2.179	0.179	0.52	0.376	2.233	0.180
0.46	0.112	2.477	0.148	0.46	0.355	2.547	0.164
0.10	0.003	1.538	0.056	0.10	0.009	1.536	0.056

Table A-10. Median AFs and Sigmas for Model 2, Profile 1, for 2 PGA levels [20].

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	0.148	0.761	0.071	100.0	0.437	0.590	0.079
87.1	0.148	0.743	0.071	87.1	0.437	0.571	0.079
75.9	0.148	0.711	0.071	75.9	0.438	0.540	0.079
66.1	0.148	0.654	0.071	66.1	0.438	0.485	0.079
57.5	0.149	0.561	0.071	57.5	0.439	0.403	0.080
50.1	0.150	0.469	0.072	50.1	0.441	0.332	0.080
43.7	0.151	0.400	0.072	43.7	0.443	0.282	0.081
38.0	0.153	0.369	0.073	38.0	0.447	0.262	0.082
33.1	0.156	0.356	0.074	33.1	0.453	0.255	0.084
28.8	0.160	0.365	0.075	28.8	0.462	0.264	0.085
25.1	0.167	0.376	0.082	25.1	0.475	0.274	0.089
21.9	0.175	0.414	0.069	21.9	0.494	0.304	0.089
19.1	0.187	0.449	0.083	19.1	0.519	0.328	0.093
16.6	0.204	0.508	0.098	16.6	0.555	0.371	0.103
14.5	0.215	0.562	0.117	14.5	0.590	0.417	0.113
12.6	0.230	0.615	0.098	12.6	0.636	0.468	0.125
11.0	0.256	0.702	0.098	11.0	0.691	0.527	0.120
9.5	0.277	0.796	0.105	9.5	0.771	0.621	0.127
8.3	0.296	0.924	0.094	8.3	0.828	0.729	0.113
7.2	0.333	1.108	0.087	7.2	0.917	0.869	0.090
6.3	0.366	1.296	0.098	6.3	1.027	1.044	0.111
5.5	0.398	1.475	0.122	5.5	1.138	1.220	0.105
4.8	0.391	1.478	0.218	4.8	1.159	1.279	0.184
4.2	0.321	1.254	0.231	4.2	1.029	1.178	0.233
3.6	0.268	1.075	0.163	3.6	0.843	0.996	0.191
3.2	0.248	1.056	0.144	3.2	0.768	0.970	0.154
2.8	0.230	1.029	0.130	2.8	0.707	0.945	0.152
2.4	0.230	1.116	0.133	2.4	0.673	0.979	0.138
2.1	0.249	1.330	0.115	2.1	0.718	1.155	0.136
1.8	0.239	1.427	0.136	1.8	0.688	1.244	0.140
1.6	0.254	1.747	0.148	1.6	0.722	1.512	0.154
1.4	0.230	1.839	0.170	1.4	0.703	1.720	0.136
1.2	0.211	1.918	0.176	1.2	0.641	1.790	0.155
1.0	0.223	2.241	0.182	1.0	0.666	2.074	0.157
0.91	0.207	2.287	0.215	0.91	0.661	2.277	0.186
0.79	0.158	1.932	0.156	0.79	0.530	2.034	0.172
0.69	0.131	1.794	0.112	0.69	0.428	1.861	0.122
0.60	0.122	1.915	0.169	0.60	0.388	1.949	0.163
0.52	0.117	2.160	0.182	0.52	0.368	2.186	0.186
0.46	0.111	2.461	0.151	0.46	0.349	2.504	0.166
0.10	0.003	1.533	0.056	0.10	0.009	1.523	0.057

Appendix B – IPEEE Adequacy Review

B.1 Background and Purpose

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 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 [1] to all U.S. nuclear power plants that requests information to assure that these recommendations are addressed. NTTF Recommendation 2.1: Seismic requests information related to performing a seismic risk evaluation.

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," [2] provides guidance for responding to the NRC's request for information. One of the methods in this report that can be used to demonstrate that plants are seismically adequate is to show that the seismic risk assessments performed as part of the Individual Plant Examination for External Events (IPEEE) for Severe Accident Vulnerabilities (Generic Letter 88-20, Supplement 4) [21] result in a High Confidence of Low Probability of Failure (HCLPF) seismic capacity response spectrum (IPEEE HCLPF Spectrum - IHS) that is higher than the new Ground Motion Response Spectrum (GMRS). Plants for which the IPEEE results meet certain criteria can be "screened out" using this method so that it is not necessary to perform new seismic risk analyses.

The purpose of this appendix is to show that the Salem IPEEE results from the report submitted to the USNRC for the Salem Nuclear Generating Station [22] together with the responses to Requests for Additional Information [23] meet the IPEEE adequacy criteria in the SPID [2], Section 3.3.1.

Salem is a two unit plant on the east bank of the Delaware River in Lower Alloways Creek Township, Salem County, New Jersey. The site is considered a soil site for the purposes of seismic evaluations. As discussed in Section 3.7.1.1 of the SGS UFSAR [5], the design basis horizontal Safe Shutdown Earthquake (SSE) is 0.20g and the Operating Basis Earthquake (OBE) is 0.10g. The vertical component of ground motion is taken as two-thirds of the horizontal spectrum.

For the IPEEE program, NUREG-1407 [13] assigned Salem as a Focused Scope plant with a Review Level Earthquake (RLE) of 0.30g for screening purposes. A Seismic Probabilistic Risk Assessment (SPRA) was performed for Salem to address the seismic portion of the IPEEE program; enhancements to the seismic PRA methodology were also implemented to address plant walkdowns, the potential for relay chatter, and the potential for soil failure.

Salem was also subject to resolution of Unresolved Safety Issue (USI) A-46 [24] to resolve certain seismic adequacy issues associated with older operating nuclear power plants. This seismic program focused primarily upon verifying that mechanical and electrical equipment needed to safely shutdown following an SSE had adequate seismic capacity to meet the plant licensing basis.

The topics covered in this IPEEE screening evaluation report parallel those described in [2] Section 3.3.1 namely, a discussion of the General Considerations for applying the IPEEE screening method, a description of how the Prerequisites are met for using the screening method, a discussion of the Adequacy Demonstration, and an overall Conclusion that the IPEEE results can be used for screening purposes.

B.2 General Considerations

The results from a Focused Scope IPEEE evaluation performed for Salem can be used to screen out of performing new seismic risk analyses in the NTTF 2.1 seismic program if four conditions are met. First, the Focused Scope IPEEE review must be enhanced to include a Full Scope detailed review of relay chatter for components such as electric relays and switches. Second, the Focused Scope IPEEE review must include a Full Scope evaluation of soil failures, such as liquefaction, slope stability, and settlement. Third, the plant-level HCLPF response spectrum, determined from the IPEEE evaluation, bounds the GMRS over two frequency ranges (1 – 10 Hz and greater than 10 Hz). And fourth, where modifications were required to achieve the IPEEE HCLPF, it is necessary to verify that the changes that were identified during the IPEEE program were implemented and remain in effect and subsequent plant modifications have not reduced the plant seismic capability. How each of these conditions is met is described in the following four subsections.

B.2.1 Relay Chatter

To satisfy the IPEEE program requirements for a Focused Scope plant, a relay evaluation, as defined in Section 3.2.4.2 of NUREG-1407 [13], was performed for Salem [25]. At the time of the evaluation, the A-46 relay evaluation had not been completed, so a review of low ruggedness relays[1] (LRRs) was performed. The LRRs were identified using guidance from [26].

This LRRs were evaluated to determine whether they could adversely impact safe shutdown of the plant and containment performance if they chattered. Those that could have a potential impact were evaluated further to determine whether they had sufficient seismic capacity to withstand a seismic demand up to the RLE of 0.30 g.

Of the approximately 100 LRRs identified in the plant, the relays installed in one application were replaced with higher seismic capacity relays. The rest of the LRRs were screened out from further consideration because (1) they were not associated with safe shutdown or

[1] See [26] Appendix E for identification of LRRs.

containment function components, (2) relay chatter was acceptable, or (3) they were determined to have capacity greater than the IPEEE RLE.

Therefore, the IPEEE Submittal Report [22] concludes in Section 3.1.5.4.3 that relay chatter is not significant to safe shutdown or containment performance after a seismic event at the Salem plant. This conclusion was confirmed in the Brookhaven National Laboratory report included as Attachment 1 to the NRC Staff Evaluation Report of the Salem IPEEE program [27] where it concluded:

> "This evaluation procedure [for relay chatter] appears to be acceptable as long as the identified LRR's have been, in fact, replaced by higher capacity relays." [pg 4]

As described in Section B.3.2 below, the modifications and other changes credited in the IPEEE analysis are shown to have been implemented.

In accordance with SPID [2] Section 3.3.1, Focused Scope margin submittals may be used after having been enhanced to bring the assessment in line with Full Scope assessments. Therefore, a Full Scope detailed review of relay chatter for components such as electric relays and switches is required.

B.2.2 Soil Liquefaction and Soil Failure

The second condition that must be met to screen out from performing new seismic risk analyses is to show that a Full Scope evaluation of soil failures, such as liquefaction, slope stability, and settlement, was completed as a part of the IPEEE program.

As described in Section 3.1.4.4.3 of the IPEEE submittal report [22], the potential for liquefaction, seismically induced settlement, and lateral spreading at the Salem site was evaluated by Woodward-Clyde Consultants using a probabilistic evaluation approach. This evaluation considered buildings at the power block and the Service Water Intake Structure. Soil at the SGS site consists of a layer of hydraulic fill at the surface and various sands beneath.

The computed probabilities of liquefaction and seismically induced foundation settlements are very small even at a peak ground acceleration as high as 0.60 g. Specific discussion on settlement and slope instability is provided in the Woodward-Clyde Report on soil liquefaction and slope stability [39] as follows:

- Total seismically induced settlements at the 84th percentile are less than 1/4-inch for the SSE level.
- The critical piping systems that may be susceptible to lateral spreading displacement at SGS are the service water supply piping, the service water discharge piping, and the service water electrical cables. Results of analyses performed to evaluate the earthquake-induced displacements of the compacted fill indicate that buried piping and

cables that are located within engineered backfill are likely to experience less than 0.5 ft of earthquake induced displacements for PGA less than 0.6 g.

- The site is generally level with no significant natural or constructed slopes beyond the shoreline. The shoreline consists of riprap slopes at the southern portion of the site, vegetated slopes between the Salem and Hope Creek Service Water Intake Structures (SWIS) and a bulkhead north of the Hope Creek SWIS. These site conditions indicate that flow failures, typically associated with steep slopes, do not appear to be a concern.

The HCLPF capacities (in terms of PGA) for liquefaction were estimated to be 0.60 g at the power block and 0.50 g at the Service Water Intake Structure. The HCLPF value for extensive liquefaction at the Vincentown formation, upon which the Seismic Category I structures are founded, is estimated to be in excess of 0.60 g PGA.

Although the hydraulic fill near the plant grade level may liquefy at an acceleration level much lower than 0.60 g and therefore could result in an increase of lateral pressure on the subgrade walls of the Seismic Category I structures, the seismic fragility evaluation of the structures included the effects of hydrostatic and hydrodynamic pressure as well as the static and dynamic lateral earthquake pressure acting on the below grade exterior walls. Therefore, liquefaction of the hydraulic fill does not have a significant impact on the seismic fragilities of Seismic Category I structures.

The soil failures evaluation described in the IPEEE submittal report is considered to adequately assess the potential for adverse soil failure effects on Salem Seismic Category I structures. Therefore, this second condition is considered to have been satisfactorily met so that the IPEEE results can be used to screen out from performing new seismic risk analyses in the NTTF 2.1 seismic program.

B.2.3 IPEEE HCLPF Spectrum

The third condition that must be met to be able to use the IPEEE results to screen out from performing new seismic risk analyses is to show that the plant-level IPEEE HCLPF spectrum (IHS) bounds the GMRS over two frequency ranges: 1 to 10 Hz and greater than 10 Hz.

The Salem IPEEE evaluation did not determine the plant-level high confidence of low probability of failure (HCLPF) acceleration value. The Salem IPEEE SPRA determined that the total seismic core damage frequency (CDF) was 9.5×10^{-6} per year based on the seismic hazard curve developed by Lawrence Livermore National Laboratory (LLNL). The plant-level HCLPF can be back calculated from the CDF and the LLNL seismic hazard curve assuming a fragility curve total uncertainty (β_c) of 0.4.[2] Based on this methodology, the Salem plant-level HCLPF is 0.27 g PGA [28]. The IPEEE Probabilistic Seismic Response analysis is developed using the

[2] This value is consistent with the approach outlined in EPRI 1003121 [29] and is conservative relative to the value recommended for a combined set of component fragilities by Dr. R. Kennedy in NEA/CSNI/R(99)28 [30]. This methodology is consistent with Method 1B used by the NRC Staff to determine the plant-level HCLPF values for plants performing IPEEE SPRAs [18] Appendix C.

EPRI Uniform Hazard Spectral Shape. Accordingly, the IPEEE HCLPF spectrum (IHS) is developed using the EPRI UHS spectral shape anchored at the plant level HCLPF value. The 5% damped horizontal IHS for SGS is shown in Table 3-2. The GMRS is provided in Table 2-4.

The IHS exceeds the GMRS for all frequencies above 1.5 Hz. Per SPID Section 3.2.1.1, since the GMRS does not exceed the Low Hazard Threshold (LHT) of 0.4 g, low frequency exceedances below 2.5 Hz do not require a plant to perform a full risk evaluation (see Figures 2-5 and 3-1). Instead, it is sufficient to identify all safety-significant SSCs at frequencies below the exceedance frequency and demonstrate their capacities are above the GMRS. All low frequency components demonstrate capacity above the GMRS based on the fact that they satisfy the design basis SSE response spectrum which is significantly higher than the GMRS at all frequencies below 2.5 Hz. Therefore, this third condition is considered to have been satisfactorily met so that the IPEEE results can be used to screen out from performing a full risk evaluation in the NTTF 2.1 seismic program.

B.2.4 Modifications to Achieve IPEEE HCLPF

The fourth condition that must be met to screen out from performing new seismic risk analyses in the NTTF 2.1 seismic program is to verify that the modifications required to achieve the IPEEE HCLPF were implemented and remain in effect. Confirmation of this condition is described in Section B.3.2.

B.3 Prerequisites

Section 3.3.1 of the SPID [2] identifies the following items that must be confirmed in order to use the IPEEE analysis to demonstrate that the IPEEE results can be used for comparison with the GMRS:

1. Confirm that commitments made during the IPEEE program have been met.
2. Confirm that all of the modifications and other changes credited in the IPEEE program are still in place.
3. Confirm that any identified deficiencies or weaknesses to NUREG-1407 in the plant-specific NRC Safety Evaluation Report (SER) are properly justified to ensure that the IPEEE conclusions remain valid.
4. Confirm that major plant modifications implemented after the IPEEE program was completed have not degraded or adversely impacted the conclusions reached in the IPEEE program.

Confirmation of each of these prerequisites is summarized below.

B.3.1 IPEEE Commitments

No commitments were made as a result of the seismic IPEEE evaluations.

B.3.2 Modifications Credited in IPEEE Analyses

The enhancements described below in Section B.3.3 were credited in the risk quantification performed in the IPEEE program, as noted on page 8 of the Brookhaven National Laboratory report included as Attachment 1 to the NRC Staff Evaluation Report of the Salem IPEEE program [27].

B.3.3 Weaknesses Identified in IPEEE SER

The IPEEE Submittal Report [22] states on page 2-4 that the Nuclear Energy Institute (NEI) Report 91-04, "Severe Accident Issue Closure Guideline," Revision 1, was used to identify vulnerabilities for further evaluation. Based on those guidelines, no potential vulnerabilities associated with seismic, fire, or other external events were identified. Therefore, as reported on page 2 of the cover letter that transmitted NRC IPEEE SER [27] to PSE&G, no improvements related to external events were considered necessary. Nevertheless, even though no significant deficiencies or weaknesses were identified, PSEG chose to implement a number of enhancements related to the seismic area as a result of the IPEEE seismic walkdowns. The seismic related enhancements include the following:

- Reinforcement of an 8-foot masonry wall in the 4kV switchgear room. AR 951020095 [31] was issued with the recommendation that the seismic capacity of the wall be increased. Subsequent evaluation determined that seismic interaction between the masonry wall and safety-related equipment was not credible, based on guidance from EPRI NP-6041-SL [32]. Accordingly, PSEG Nuclear did not modify the masonry wall.

- Replacement of certain low ruggedness relays with higher seismic capacity relays (e.g., 4kV Phase A/B/C diesel generator differential relays). Completion of this enhancement is verified on page 3-31 of the IPEEE submittal [22].

B.3.4 Major Plant Modifications since IPEEE

A review of major plant modifications [33] performed since completion of the Salem IPEEE program did not identify any adverse impact on the conclusions of the IPEEE. To complete this review, all modifications since January 1996 were screened for relevance to the seismic damage sequences cited in the IPEEE report as having a significant contribution to CDF. Screening was performed via keyword search and also using a line-by-line manual scan of modification descriptions. Details of relevant modifications were then reviewed to assess the potential for adverse impact on the IPEEE results.

Based on the above four confirmation statements, Salem meets the prerequisite requirements to proceed to perform an IPEEE screening evaluation.

B.4 Adequacy Demonstration

Section 3.3.1 of the SPID [2] identifies information that should be included in the submittal report to the NRC when the IPEEE screening method is used. This information addresses the major technical considerations associated with the adequacy of the IPEEE analyses, documentation, and peer review to support use of the IPEEE results for screening purposes.

As noted by NRC staff on page 6 of the Staff Evaluation Report of the Salem IPEEE program [27], the IPEEE program is considered complete and reasonable such that the most likely severe accidents and vulnerabilities can be identified.

> "On the basis of the overall review findings, the staff concludes that: (1) the licensee's IPEEE is complete with regard to the information requested by Supplement 4 to GL 88-20 (and associated guidance in NUREG-1407), and (2) the IPEEE results are reasonable given the Salem design, operation, and history. Therefore, the staff concludes that the licensee's IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities, and therefore, that the Salem IPEEE has met the intent of Supplement 4 to GL 88-20 and the resolution of specific generic and unresolved safety issues discussed in this SER."

To confirm that the Salem IPEEE also adequately addresses the major technical considerations to support use of the IPEEE screening method for NTTF 2.1, nine areas of the IPEEE program are described and evaluated in the following subsections. For each of these areas, the following discussion includes a discussion of (1) the methodology used, (2) whether the analysis was conducted in accordance with the guidance in NUREG-1407 [13], and (3) a statement, if applicable, as to whether the methodology and results are adequate for screening purposes.

B.4.1 Structural Models and Structural Response Analysis

The IPEEE Submittal Report [22] and its supporting reference documents describe in detail the structural models developed for the IPEEE program as well as the structural responses using those models. In summary, new probabilistic, soil-structure interaction (SSI) building models were developed for the containment building including internal structures, for the auxiliary building, and for the service water intake structure.

Probabilistic response analyses were performed for free field input motions selected to match the 10,000 year EPRI Uniform Hazard Spectrum (UHS) shape anchored to 3 times SSE (i.e., 0.60 g peak ground acceleration). An ensemble of time histories was generated such that their median response spectra matched the median 10,000 year EPRI UHS. Variability in the time histories corresponds to the peak-to-valley variability in real earthquake ground motion spectra. Thirty earthquake motions, three components each, were generated such that their median 5% damped spectra matched the EPRI UHS with coefficient of variation of 0.20. Variability in stiffness and damping of both structures and soil were also considered in these analyses.

The Soil Structure Interaction analysis [34] utilized the substructure approach; structural models for this approach are fixed-base and SSI effects are incorporated using foundation impedances and wave scattering functions. Structural models were developed for the reactor containment buildings and internal structures, the auxiliary building, and the service water intake structure. The modal damping ratios used for the building models were the upper bound damping values from NUREG/CR-0098 [40] corresponding to the at-yield values. The variability in soil and structure properties were incorporated in the probabilistic response analysis by performing a Latin Hypercube Simulation from lognormal probability distributions with the following coefficients of variations [22]:

Soil shear modulus: 0.35
Soil material damping: 0.50
Structural frequencies: 0.25
Structural modal damping: 0.35

The median and 84% non-exceedance probability (NEP) responses were calculated for each selected in-structure response. These included peak accelerations, maximum member forces, and floor response spectra at chosen elevations as needed for equipment fragility estimation.

The structural models and structural response analyses are consistent with the guidelines in NUREG-1407 [13] and SPID [2] Section 6.3.1. In particular, the new models developed for the IPEEE program accounted for variability of soil conditions.

The methodology and results of the IPEEE structural modeling are considered adequate for IPEEE screening purposes.

B.4.2 In-structure Demands and In-structure Response Spectrum (ISRS)

The probabilistic seismic response analysis report [34] describes in detail the methodology used to perform the probabilistic seismic response analyses for the IPEEE program. In summary, the new probabilistic floor response spectra developed for the IPEEE program followed the methodology developed under the Seismic Safety Margins Research Program (SSMRP), conducted in the early 1980s in the United States and applied to several seismic probabilistic risk assessment (PRA) and margin studies. Such an approach was in line with trends in the early 1990s toward an explicit treatment of uncertainties in various phases of the analysis procedure, e.g., specification of free-field ground motion and development of the structure model. This approach provides a complete description of the seismic environment for equipment mounted in structures and was used directly in the seismic PRA for Salem.

The in-structure seismic demands and in-structure response spectra are consistent with the guidelines in NUREG-1407 [13] and SPID [2] Section 6.3.1. In particular, new in-structure seismic demands and in-structure response spectra were developed for the IPEEE program.

The methodology and results for IPEEE in-structure demands and in-structure response spectra are considered adequate for IPEEE screening purposes.

B.4.3 Selection of Seismic Equipment List or Safe Shutdown Equipment List

The EQE seismic walkdown report [35] describes the details of the methods use to develop the seismic equipment list. In summary, more than 300 components were initially selected, as shown in Table 3-4 of the IPEEE Submittal Report [22]. The types of components in this list include:

- Critical components identified in the internal events PRA model
- Components needed for containment performance
- Components associated with such issues as seismic-induced fires and floods
- Passive components that could have significant conditional probabilities of seismic failure
- Components that could inadvertently change state during an earthquake and divert flow
- Instrumentation, racks, cabinets, transformers, switchgear, motor control centers, and panels that provide essential signals, power or control room indication
- Structures housing the components identified above

The selection of components for the seismic equipment list is consistent with the guidelines in NUREG-1407 [13] and EPRI NP-6041-SL [32].

The methodology and results for equipment selection are considered adequate for IPEEE screening purposes.

B.4.4 Screening of Components

The EQE seismic walkdown report [35] describes the details of the methods used to screen out from further evaluations components on the seismic equipment list. In summary, as described in the IPEEE submittal report [22], several screens were used to narrow the scope of seismic evaluations. The first screen was to narrow the scope of components to only those described above in Section B.4.3.

The second type of screen was based on plant walkdowns in which components with high seismic capacity were identified and screened out from further evaluations. This screen eliminated such components as valves, horizontal pumps and compressors, small instruments mounted on walls or ceilings, and distributed systems such as piping, cable trays, and HVAC ducts.

The third type of screen eliminated components with relatively high seismic capacity compared to realistic seismic demands based on the Salem probabilistic floor response spectra. Most structures and components on the seismic equipment list were screened out based on their high seismic capacity. The conservative screening criteria used for this screening were: 1) median acceleration capacity greater than 1.50 g, and 2) HCLPF greater than 0.50 g. These criteria are conservative when compared to the seismic margins RLE of 0.30 g and a design basis earthquake (SSE) of 0.20 g.

These screening methods narrowed the scope of components to about 100 items for further evaluations.

The methodology and results for screening of components from the seismic equipment list are reasonable and meet the intent of NUREG-1407 [13]. Therefore, this component screening is considered adequate for IPEEE screening purposes.

B.4.5 Walkdowns

The EQE seismic walkdown report [35] describes in detail the approach used in and results of the seismic walkdowns. In summary, walkdowns were performed to find as-designed, as-built, and as-operated seismic weaknesses of components in the plant. In particular, personnel who participated or supported the seismic walkdowns included individuals who were familiar with plant systems and operations, PRA methods, and structural analysis. A series of walkdowns were performed to further develop the seismic equipment list, to pre-screen components with high seismic capacity, to determine component failure modes, to identify spatial interactions, and to evaluate the likelihood of seismic induced fire and flooding. For each of the components on the seismic equipment list, a Seismic Evaluation Walkdown Sheet (SEWS) was prepared to record the walkdown findings.

The seismic walkdowns are consistent with the guidelines in NUREG-1407 [13]. In particular, as specified in Section 3.1.1.4 of NUREG-1407 [13], the intent of the guidelines in EPRI NP-6041-SL [32] is met. This was confirmed on page 3 of the Brookhaven National Laboratory report, included as Attachment 1 in the NRC Staff Evaluation Report of the Salem IPEEE program [27] in which it was concluded that the walkdown procedure was appropriate.

The methodology and results of the seismic walkdowns are considered adequate for IPEEE screening purposes.

B.4.6 Fragility Evaluations

The EQE seismic fragility analysis report [36] describes in detail the approach used in and results of determining the seismic fragilities of structures and equipment. A summary of the component fragilities and failure modes is included in the IPEEE submittal report [22]. Highlights from these documents are summarized below.

Seismic fragilities of structures and equipment were estimated using EPRI TR-103959 [37] and EPRI NP-6041-SL [32]. Seismic fragilities were developed in terms of the peak ground acceleration capacity of structures and equipment. As such, three fragility parameters were calculated for each screened-in component for its significant failure mode, namely the median ground acceleration capacity (A_m) and the logarithmic standard deviations associated with randomness (β_R), and uncertainty (β_U).

Seismic fragilities of important structures, tanks, and block walls were estimated for significant failure modes using a combination of the probabilistic response analyses as described in the EQE probabilistic seismic response analysis report [34] together with knowledge of the SSE design criteria utilized to build the plant.

Structures were deemed to fail when their inelastic deformation exceeds the level that interferes with the operability of the equipment housed inside or mounted on the structure. In some instances, structures were considered to fail when the sliding displacements exceeded the deformation capability of attached piping. Tanks were considered to fail when they lose their contents. Block walls were deemed to fail when they either collapse on adjacent components or suffer large deformations that may interfere with the functionality of attached equipment.

The fragility parameters for selected structures, tanks, and block walls are shown in Table B-1 below. This table also shows HCLPF capacities for these structures, which are generally about 0.50 g PGA or higher.

Table B-1 Seismic Fragility Parameters for Structures
(Adapted from Table 3-5 of Reference 22)

Structure	Failure Mode	A_m (g)	β_R	β_U	HCLPF (g)
Containment Building	Wall shear	6.5	0.20	0.33	2.8
Concrete Internal	Shear failure	2.3	0.20	0.38	0.88
Aux. Bldg	Flexure of Shearwall	1.68	0.25	0.32	0.66
Service Water Intake Structure	Sliding	2.40	0.35	0.35	0.76
Refueling Water Storage Tank	Wall Buckling	1.29	0.22	0.36	0.50
Aux. Feedwater Storage Tank	Wall Buckling	1.10	0.22	0.36	0.42
Battery Room Block Wall	Flexure Failure	1.35	0.27	0.35	0.49

Seismic fragilities of screened-in equipment from the seismic equipment list were estimated for significant failure modes. Failure modes considered in the fragility evaluation include elastic functional failures, brittle failures, and ductile failures. Elastic functional failures involve loss of intended function while the component is stressed below its yield point. Examples of this type of failure include: elastic buckling in tank walls or component supports; chatter and trip in electrical components; excessive blade deflection in fans; and shaft seizure in pumps.

Brittle failures are those failure modes that have little or no system inelastic energy absorption capability. Examples include: expansion anchor failures; component support weld failures; and shear pin failures.

Ductile failure modes are those in which the structural system can absorb a significant amount of energy through inelastic deformation. Examples include: pressure boundary failure of piping, structural failure of cable trays, and structural failure of ducting.

The equipment fragilities are based on plant-specific analyses, earthquake experience databases, and generic PRA databases.

The fragility parameters for selected equipment (with their abbreviations) are shown in Table B-2 below. This table also shows HCLPF capacities for this equipment.

Table B-2. Seismic Fragility Parameters for Equipment
(Adapted from Table 3-7 of Reference 22)

Equipment	**Abbrev**	**A_m** (g)	**β_R**	**β_U**	**HCLPF** (g)
Switchyard	SWYD	0.31	0.25	0.43	0.10
Containment Fan Coolers	CFCU				>0.5
CCS Heat Exchanger	CCSHX	1.03	0.28	0.28	0.41
125 and 28vdc Batteries A and B	BATT-AB	1.06	0.27	0.35	0.38
AFW Storage Tank	AFST	1.10	0.22	0.36	0.42
Room Coolers for ESF on AB84	RMCLRS-84	1.13	0.24	0.32	0.45
Room Coolers for RHR	RMCLRS-45	1.13	0.24	0.32	0.45
Control Room Ceiling Grid	CRCEILING	1.28	0.27	0.35	0.36
Refueling Water Storage Tank	RWST	1.29	0.22	0.36	0.50
DG Area Exhaust Fans	DGFANS	1.30	0.30	0.40	0.41
Switchgear Rooms Fans	SWGRFANS	1.30	0.30	0.40	0.41
125 and 28vdc Batteries A, B, and C	BATT-ALL	1.35	0.27	0.35	0.49
CCS Surge Tank	SURGETK	1.48	0.27	0.35	0.53
Small LOCA due to seismic event	SLOCA	1.50	0.30	0.50	0.40

The fragility evaluations are consistent with the guidelines in NUREG-1407 [13], EPRI TR-103959 [37], and EPRI NP-6041-SL [32]. In particular, fragilities were based on generic PRA data, plant-specific analyses, earthquake experience data, and design basis data. The fragilities of limiting structures and equipment are documented in the IPEEE Submittal Report [22] and its supporting references.

The methodology and results of the fragility evaluations are considered adequate for IPEEE screening purposes.

B.4.7 System Modeling

The IPEEE Submittal Report [22] and the NUS Seismic Qualification Report [38] describe in detail the system modeling and evaluations performed for the IPEEE seismic PRA. In summary, the event and fault tree models developed for the Salem Units 1 and 2 internal events IPE were used as the starting point for the seismic IPEEE models. Traditional event tree techniques were used to delineate the potential combinations of seismic-induced failures, and resulting seismic scenarios, which were termed "seismic damage states" (SDS). The frequencies of these seismic damage states were quantified by convolving the earthquake hazard curve with the structure and equipment seismic fragility curves.

For those seismic damage states with frequency greater than 10^{-7}, the impact on the plant and plant systems was evaluated using the internal events IPE model and its dependency matrices as the primary basis. Only 15 SDSs met this criterion, as shown in the IPEEE Submittal Report [22] Table 3-9, "Salem Seismic Core Damage Frequencies." This table, reproduced below in Table B-3, shows the Seismic Damage State (SDS) for both the EPRI and LLNL hazards, the conditional core damage probability (CDP) for each sequence, and annual core damage frequency (CDF) for each sequence. The meaning of the sequence abbreviations used in this table is based on the failure equations listed in Table B-4 and the equipment abbreviations shown in Table B-2.

Table B-3. Salem Seismic Core Damage Frequencies
(Reproduced from [22] Table 3-9)

Sequence		**Seismic Damage State (SDS) Frequency**		**Conditional CDP**	**CDF** (per yr)	
		EPRI Hazard	**LLNL Hazard**		**EPRI**	**LLNL**
2	S2	1.0E-07	1.8E-07	2.5E-03	2.5E-10	4.5E-10
5	FW	6.3E-08	1.3E-07	2.7E-01	1.7E-08	3.5E-08
15	SW	5.8E-08	1.5E-07	1.0E+00	5.8E-08	1.5E-07
17	OP	5.5E-05	5.3E-05	5.5E-02	3.0E-06	2.9E-06
17F	OP-FC	4.2E-06	9.9E-06	5.5E-02	2.3E-07	5.4E-07
18	OP-S2	1.5E-07	2.8E-07	2.3E-01	3.5E-08	6.5E-08
18F	OP-S2-FC	1.0E-07	3.7E-07	2.3E-01	2.3E-08	8.5E-08
19	OP-R	9.0E-08	3.9E-07	5.6E-02	5.0E-09	2.2E-08
21	OP-FW	8.1E-08	2.2E-07	5.2E-01	4.2E-08	1.2E-07
21F	OP-FW-FC	1.2E-07	5.5E-07	5.2E-01	6.2E-08	2.9E-07
24	OP-CW	4.7E-08	2.1E-07	6.3E-02	3.0E-09	1.3E-08
31	OP-SW	3.2E-07	1.3E-06	1.0E+00	3.2E-07	1.3E-06
33	OP-DAB	5.3E-07	2.0E-06	1.0E+00	5.3E-07	2.0E-06
34	OP-DAB-DG	1.6E-07	7.7E-07	1.0E+00	1.6E-07	7.7E-07
35	OP-IC	2.3E-07	1.2E-06	1.0E+00	2.3E-07	1.2E-06

Table B-3. Salem Seismic Core Damage Frequencies
(Reproduced from [22] Table 3-9)

Sequence		Seismic Damage State (SDS) Frequency		Conditional CDP	CDF (per yr)	
		EPRI Hazard	LLNL Hazard		EPRI	LLNL
				Total CDF	4.7E-06	9.5E-06

Table B-4. Failure Equations for Top Events
(From [22])

S	=	(no equation needed since this is the seismic event)
OP	=	SWYD + IC + DAB (where IC and DAB represent the equations below)
IC	=	CRCEILING
DG	=	BATT-ALL
DAB	=	BATT-AB
ERC	=	(RMCLRS-84 + SWGRFANS + DGFANS) * ALTRMCL (for sequences with loss of offsite power)
ERC	=	(RMCLRS-84 + SWGRFANS) * ALTRMCL (for sequences with offsite power available)
SW	=	CCSHX
CW	=	SURGETK
FW	=	AFST
R	=	RWST
S2	=	SLOCA
FC	=	CFCU

Of these 15 SDSs shown in Table B-3, five of them (SDS 15, 31, 33, 34, and 35) directly result in core damage, and loss of containment heat removal systems, i.e., each has a conditional core damage probability (CCDP) of 1.0, or guaranteed failure. Therefore, no conditional core damage probability calculation of non-seismic failures is needed because the plant and containment damage states are delineated. The internal events IPE and IPE models were used to determine CCDPs for the remaining 10 SDSs in Table B-3. For those scenarios that required additional non-seismic failures to occur to result in core damage, calculations incorporated random failures of equipment and operator actions.

Special attention was given to human interactions and recovery actions in the IPEEE evaluation. For example, offsite power recovery within the first 24 hours was not credited. No relay chatter interactions requiring human actions were needed, based on the results from the low ruggedness relay evaluation. The only operator action included in the seismic event tree analysis is the long term provision of alternate ventilation for the switchgear room, the diesel generator rooms, and the Engineered Safety Features (ESF) rooms in the Auxiliary Building. Since several hours are available for these operator actions, and they are similar to those previously analyzed and incorporated in the internal events IPE, it was judged that the IPE mean value of 3.0×10^{-2}, with an error factor of 5.0 is a conservative screening value for the seismic analysis as well.

The only significant operator action in the conditional core damage probability calculation is the feed and bleed action, which is prominent in the procedures and training, and need not be implemented for at least 30 minutes. It was judged that the IPE evaluation and human error probability (HEP) would remain appropriate for the seismic conditions.

The system modeling and evaluations performed for the IPEEE seismic PRA are consistent with the guidelines in NUREG-1407 [13] (see Section 3.2.4.7). In particular, the IPEEE evaluation addressed the development of the event and fault trees, the treatment of non-seismic failures, and how human actions were treated. The Brookhaven National Laboratory report on page 7, included as Attachment 1 in the NRC Staff Evaluation Report of the Salem IPEEE program [27], concluded that the "construction of logic models seems to have been generally correct."

The methodology and results of the system modeling and evaluations are considered adequate for IPEEE screening purposes.

B.4.8 Containment Performance

Sections 3.1.5.5.2 and 3.1.6 of the IPEEE submittal report [22] describe in detail the evaluation of the containment performance during a seismic event. In summary, containment performance under seismic conditions was evaluated for containment structural integrity, containment isolation equipment to protect against containment bypass, and containment cooling systems. No vulnerabilities were identified for any aspect of the containment performance.

As defined in NUREG-1407 [13] (see Section 3.1.1.5), the purpose of the seismic containment performance evaluation was to identify vulnerabilities that could lead to early failure of containment functions including continued integrity of the containment, containment isolation, prevention of bypass functions, and some specific systems depending on containment design. The components of the containment system that were examined during the IPEEE program are shown in Table B-5.

Table B-5. Components of Containment System Examined in the IPEEE
(Reproduced from [22] Table 3-11)

- Containment Isolation Valves both Inside and Outside Containment and Associated Piping and Electrical/Mechanical Penetration Assemblies
- Main Steam Isolation Valves (MSIVs)
- Containment Spray Headers
- Containment Fan Coolers
- Activation Sensors and System for Containment Isolation
- Containment Purge Supply, Exhaust, and Pressure Vacuum Relief Valves
- Personal Hatch and Associated Seal
- Equipment Hatch and Associated Seal

Because no vulnerabilities were identified for any aspect of containment performance, it was not necessary to perform fragility or HCLPF evaluations.

The containment performance evaluation performed for the IPEEE program is consistent with the guidelines in NUREG-1407 [13]. This was confirmed by the NRC on page 3 of the NRC Staff Evaluation Report for the Salem IPEEE program [27] that the containment performance analyses for seismic events "appeared to have considered important severe phenomena and are consistent with the intent of Supplement 4 to GL 88-20."

The methodology and results of the containment performance evaluation are considered adequate for IPEEE screening purposes.

B.4.9 Peer Review

Section 6 of the IPEEE submittal report [22] describes in detail how the peer review was conducted, who serviced as peer reviewers, what findings were identified, and how those findings were dispositioned.

In summary, the individuals of the Independent Review Team (IRT) for the Salem IPEEE project had extensive, relevant experience related to the elements of the IPEEE program. The PSEG personnel on the IRT had knowledge of their plant, system configurations, and operating practices and procedures. They also had combined experience in the areas of systems engineering, seismic capacity engineering, and seismic PRAs. The two contractors who participated on the IRT had significant technical expertise in related areas. Dr. Michael Frank had significant experience in risk, safety, reliability, and uncertainty analysis. Dr. John Stevenson had expertise in structural and mechanical design and qualification of nuclear power plant structures and components as it applies to the effects of earthquakes on structures and associated designs and analyses. Section 6.2 of the IPEEE Submittal Report [22] provides the background and qualifications for each of the independent review team members.

The IRT focused their review on the assumptions, modeling approaches, results, and conclusions for the IPEEE. They divided their review into the following three stand-alone segments:

- Seismic/Soil
- Fire
- High Winds, Floods, and Other Environments

The areas evaluated in the Seismic/Soil segment included the seismic, soil, and soil/structure interaction studies. The IRT sought to ascertain whether the methodologies used were adequate and whether the results generated were reasonable. The Screening Evaluation Walkdown Sheets (SEWS) of selected Salem plant components were reviewed by the IRT. They also performed plant walkdowns to verify that the information recorded on the SEWS was reasonable. The IRT thoroughly reviewed the Tier 1 reports associated with the IPEEE seismic program. They also reviewed a sample of PRA fragility calculations and the relay chatter evaluations as well as the seismic quantification analyses. The comments generated by the IRT were resolved and changes were incorporated into the affected calculations and documents. They also concluded that the conservative methodology limitations used in the seismic evaluations appeared reasonable and that the LLNL seismic hazard information and walkdown results provided acceptable results.

As described in Section 6.3 of the IPEEE Submittal Report [22], significant effort was expended during the IRT review of seismic and soil topics on the following topics. The results of those reviews are summarized below:

- <u>Dynamic Soil Properties</u> – Values for Dynamic Soil Properties were derived primarily from existing soil data extracted from the Salem UFSAR and various Dames & Moore Reports. Due to the close proximity of Salem and Hope Creek sites, some soil data prepared for Hope Creek were also used. The IRT reported that the Dynamic Soil Properties used in the report were reasonable and representative of the site soil condition.

- <u>Soil Liquefaction and Slope Stability</u> – The report on soil liquefaction potential and slope stability was reviewed. Since the Salem power block foundation is resting on the Vincentown formation, which is a very old formation and has high shear wave velocity, the computed probabilities of soil liquefaction and seismically induced settlements and differential settlements are very small as anticipated.

  The site is generally level with no significant natural or constructed slopes beyond the shoreline. The IRT reported that the site conditions indicate that flow failures, typically associated with steep slopes, do not appear to be a concern.

- <u>Probabilistic Seismic Response Analyses</u> – The methodology and procedures used in calculating the Probabilistic Seismic Responses were reviewed. While verifying the numerical results was beyond the scope of the Independent Peer Review, the IRT endorsed the methods used in the process. The IRT reported that the results appear to be reasonable and were consistent with their expectations.

The peer review performed for the IPEEE program included how the peer review was conducted, whether the peer review conformed to the guidance, who served as peer reviewers, and the peer review findings and their disposition. These evaluations are consistent with the guidelines in NUREG-1407 [13], which specify that the IPEEE peer review team should be independent (or capable of providing an objective and critical review) and have combined experience in the areas of systems engineering and specific external events.

The methodology and results of the peer review is considered adequate for IPEEE screening purposes.

B.5 Conclusion

The adequacy review concludes that the IPEEE evaluation is adequate to support screening of the updated seismic hazard for Salem provided a Full Scope detailed review of relay chatter is performed in accordance with SPID [2] Section 3.3.1. The review also concludes that the risk insights obtained from the IPEEE are still valid under the current plant configuration.

Summary of Commitments

The following table identifies commitments made in this document. (Any other actions discussed in the submittal represent intended or planned actions. They are described to the NRC for the NRC's information and are not regulatory commitments.)

Commitment	Committed Date or Milestone	Commitment Type	
		One-Time Action (Yes/No)	Programmatic (Yes/No)
1. PSEG will transmit the Expedited Seismic Evaluation Process (ESEP) report for SGS Units 1 and 2, using the guidance in the NEI proposed path forward dated April 9, 2013 (ADAMS Accession No. ML13101A379).	12/31/2014	Yes	No
2. PSEG will complete non-outage plant modifications, if needed based on the ESEP results as determined by Commitment #1.	12/31/2016	Yes	No
3a. PSEG will complete SGS Unit 1 outage-related plant modifications, if needed based on the ESEP results as determined by Commitment #1.	Prior to restart from the second refueling outage after December 2014	Yes	No
3b. PSEG will complete SGS Unit 2 outage-related plant modifications, if needed based on the ESEP results as determined by Commitment #1.	Prior to restart from the second refueling outage after December 2014	Yes	No

Commitment	Committed Date or Milestone	Commitment Type	
		One-Time Action (Yes/No)	Programmatic (Yes/No)
4. PSEG will perform a spent fuel pool evaluation for SGS Units 1 and 2, using the guidance in EPRI Report 1025287 (ADAMS Accession No. ML12333A170) as endorsed by NRC letter dated February 15, 2013 (ADAMS Accession No. ML12319A074).	One of the following dates, to be determined using an NRC prioritization process following transmittal of this Seismic Hazard and Screening Report: 6/30/2017, or 12/31/2019, or 12/31/2020	Yes	No
5. PSEG will perform a relay chatter review for SGS Units 1 and 2 on the same schedule as the High Frequency Confirmation in the NEI proposed path forward dated April 9, 2013 (ADAMS Accession No. ML13101A379).	One of the following dates, to be determined using an NRC prioritization process following transmittal of this Seismic Hazard and Screening Report: 6/30/2017, or 12/31/2019, or 12/31/2020	Yes	No