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

CNL-14-211

December 22, 2014

10 CFR 50.4
10 CFR 50.54(f)

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

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

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

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

On March 12, 2012, the U.S. Nuclear Regulatory Commission (NRC) issued the referenced letter to all power reactor licensees and holders of construction permits in active or deferred status. Enclosure 1 of the referenced letter requested each addressee located in the Central and Eastern United States (CEUS) to submit a Seismic Hazard Evaluation that includes "an interim evaluation and actions taken or planned to address the higher seismic hazard relative to the design basis, as appropriate, prior to completion of the risk evaluation."

In accordance with the referenced letter above, TVA is enclosing the Expedited Seismic Evaluation Process (ESEP) Report for Sequoyah Nuclear Plant.

Enclosure 2 provides a list of new regulatory commitments as described in Section 8.0 of the enclosed ESEP Report.

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 22nd day of December 2014.

Respectfully,

J. W. Shea
Vice President, Nuclear Licensing

Enclosures:
1. Expedited Seismic Evaluation Process (ESEP) Report for Sequoyah Nuclear Plant
2. List of Commitments

cc (Enclosures):

NRR Director - NRC Headquarters
NRO Director - NRC Headquarters
NRR JLD Director - NRC Headquarters
NRC Regional Administrator - Region II
NRR Project Manager - Sequoyah Nuclear Plant
NRR JLD Project Manager - Sequoyah Nuclear Plant
NRC Senior Resident Inspector - Sequoyah Nuclear Plant

ENCLOSURE 1

EXPEDITED SEISMIC EVALUATION PROCESS (ESEP) REPORT FOR SEQUOYAH NUCLEAR PLANT

EXPEDITED SEISMIC EVALUATION PROCESS (ESEP) REPORT FOR SEQUOYAH NUCLEAR PLANT

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1.0 PURPOSE AND OBJECTIVE

Following the accident at the Fukushima Dai-ichi nuclear power plant resulting from the March 11, 2011, Great Tohoku Earthquake and subsequent tsunami, the Nuclear Regulatory Commission (NRC) established a Near-Term Task Force (NTTF) to conduct a systematic review of NRC processes and regulations and to determine if the agency should make additional improvements to its regulatory system. The NTTF developed a set of recommendations intended to clarify and strengthen the regulatory framework for protection against natural phenomena. Subsequently, the NRC issued a 50.54(f) letter on March 12, 2012 [1], requesting information to assure that these recommendations are addressed by all U.S. nuclear power plants. The 50.54(f) letter requests that licensees and holders of construction permits under 10 CFR Part 50 reevaluate the seismic hazards at their sites against present-day NRC requirements and guidance. Depending on the comparison between the reevaluated seismic hazard and the current design basis, further risk assessment may be required. Assessment approaches acceptable to the staff include a seismic probabilistic risk assessment (SPRA), or a seismic margin assessment (SMA). Based upon the assessment results, the NRC staff will determine whether additional regulatory actions are necessary.

This report describes the Expedited Seismic Evaluation Process (ESEP) undertaken for Sequoyah Nuclear Plant Units 1 and 2. The intent of the ESEP is to perform an interim action in response to the NRC's 50.54(f) letter to demonstrate seismic margin through a review of a subset of the plant equipment that can be relied upon to protect the reactor core following beyond design basis seismic events.

The ESEP is implemented using the methodologies in the NRC endorsed guidance in Electric Power Research Institute (EPRI) 3002000704, Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic [2].

The objective of this report is to provide summary information describing the ESEP evaluations and results. The level of detail provided in the report is intended to enable the NRC to understand the inputs used, the evaluations performed, and the decisions made as a result of the interim evaluations.

This ESEP report is for both Sequoyah Unit 1 and Unit 2 which are identical. Unless noted otherwise, all descriptions in this report apply to both Unit 1 and Unit 2 structures, systems, and components. For this reason, unit designations are not included on equipment unit identifications in the FLEX strategy or Expedited Seismic Equipment List (ESEL) descriptions.

2.0 BRIEF SUMMARY OF THE FLEX SEISMIC IMPLEMENTATION STRATEGIES

The Sequoyah FLEX strategies for Reactor Core Cooling and Heat Removal, Reactor Inventory Control, and Containment Function are summarized below. This summary is derived from the Sequoyah Overall Integrated Plan (OIP) in Response to the March 12, 2012, Commission Order EA-12-049 submitted in February 2013 [3] [4] and is consistent with the third six month status report issued to the NRC in August 2014 [5].

For At Power Conditions

Core Cooling and Heat Removal

Reactor core cooling and heat removal is achieved via steam release from the Steam Generators (SGs) with SG makeup from the Turbine Driven Auxiliary Feedwater Pump (TDAFWP) during FLEX Phase 1 with suction from the Condensate Storage Tank (CST) [5]. Local control and operation of the SG

Atmospheric Relief Valves (ARVs) and the TDAFWP system is available and proceduralized so that operation from the main control room is not required.

To provide an unlimited supply of water for core cooling during Phase 2, low pressure FLEX Pumps will be staged at the Intake Pump Station (IPS) and take suction from the intake channel and discharge to four, Emergency Raw Cooling Water (ERCW) FLEX connections inside the IPS. They will be used to pressurize the ERCW headers, which can then be used for direct supply to the TDAFWP suction.

When the TDAFWP becomes unavailable due to reduction in available steam pressure, a portable intermediate pressure FLEX pump will be used to continue to supply feedwater to the SGs. Suction would be from an ERCW FLEX connection. The discharge is routed by hose to the TDAFWP discharge FLEX connections downstream of Flow Element 3-142.

Reactor Inventory Control

For Phase 1, Reactor Coolant System (RCS) makeup will be provided by the cold leg accumulators. RCS depressurization and cool down will be initiated as soon as possible to reduce the Reactor Coolant Pump (RCP) seal leakage rate.

In Phase 2, RCS makeup will be provided by repowering existing Safety Injection (SI) pumps and using the pumps to inject borated water as needed into the RCS. The SI pumps will be repowered with a 6.9 kVA FLEX Diesel Generator. The SI pumps can be manually controlled with hand switches on Panel M-6. The source of RCS makeup will be the Refueling Water Storage Tank (RWST).

Later in Phase 2, when the RCS is depressurized sufficiently, a high-pressure FLEX pump will be used to inject borated water into the RCS through SI piping. These pumps would be aligned with a suction hose from RWST FLEX connections and a discharge hose routed to a SI pump discharge FLEX header connection. The high pressure FLEX pumps are fed from and operated from the 480v Control and Auxiliary Building (C&A) Vent Boards 1A1-A and 2A1-A.

Containment Function

There are no Phase 1 FLEX actions to maintain containment integrity. The primary Phase 2 FLEX strategy for containment integrity entails repowering one train of hydrogen igniters. Phase 2 may entail repowering the Containment Air Return Fans inside of containment.

Support Systems

Key reactor parameters to be monitored during FLEX implementation are measured and indicated by instrumentation that is powered by the 125V DC vital battery. During Phase 1, the vital batteries provide power to needed instrumentation through the vital battery boards, vital inverters and vital instrument power boards.

During Phase 2, power to vital instrumentation will be maintained by supplying 480V AC power to the vital battery chargers through new, fused, FLEX distribution panels, which will be connected directly to the battery chargers. 480V AC power will be supplied to the distribution panels by pre-staged, 480V AC FLEX diesel generators that will be located on the roof of the auxiliary building.

During the early portion of Phase 2, the 6.9kV switchgear and 6.9kV Shutdown Boards will be energized with a pre-staged 6.9kV FLEX diesel generator that will be located in the additional diesel generator building. This will allow re-energizing the SI pumps for inventory control.

For Shutdown Conditions

During shutdown, both safety functions (maintaining core cooling and heat removal and maintaining RCS inventory control) are accomplished by the same FLEX strategy and rely on the same FLEX equipment needed for the at power condition. Core cooling and heat removal is achieved by coolant boil off. Injection of borated water to the RCS is needed to replenish the coolant lost to boiling. For shutdown configurations where the RCS is depressurized and open but the cavity is not flooded, gravity feed from the RWST may be used to maintain RCS inventory in Phase 1. A flow path from the RWST to the RCS would be established. If gravity feed is not sufficient to makeup coolant to the RCS, a pre-staged, intermediate pressure FLEX pump will be used to maintain RCS inventory (in Phase 2). Sufficient flushing flow will be needed to prevent boron precipitation. Connection of the FLEX pump discharge hoses will be to the safety injection piping using the same FLEX connections planned for RCS inventory control under at power conditions. The FLEX connections are shown in [6]

For shutdown configurations where the RCS head is off and the cavity is filled, there will be sufficient time to mobilize portable FLEX pumps to provide RCS makeup from the BATs or an alternate borated water source. The same FLEX connections to the safety injection system piping will be used in this mode.

In accordance with [7] (footnote to Table D-1), some shutdown configurations where the RCS is closed or pressurized so that injection of borated water cannot be accomplished are considered outside of ESEP because these configurations have short durations.

3.0 EQUIPMENT SELECTION PROCESS AND ESEL

The selection of equipment for the ESEL followed the guidelines of EPRI 3002000704 [2]. The ESEL for Sequoyah Units 1 and 2 is presented in Attachment A. Information presented in Attachment A is drawn from [8].

3.1 Equipment Selection Process and ESEL

The selection of equipment to be included on the ESEL was based on installed plant equipment credited in the FLEX strategies during Phase 1, 2 and 3 mitigation of a Beyond Design Basis External Event (BDBEE), as outlined in the Sequoyah OIP in Response to the March 12, 2012, Commission Order EA-12-049 [3] and is consistent with the second and third six month status reports issued to the NRC [4] [5]. The OIP provides the Sequoyah FLEX mitigation strategy and serves as the basis for equipment selected for the ESEP.

The scope of “installed plant equipment” includes equipment relied upon for the FLEX strategies to sustain the critical functions of core cooling and containment integrity consistent with the Sequoyah OIP. FLEX recovery actions are excluded from the ESEP scope per EPRI 3002000704 [2]. The overall list of planned FLEX modifications and the scope for consideration herein is limited to those required to support core cooling, reactor coolant inventory and subcriticality, and containment integrity functions. Portable and pre-staged FLEX equipment (not permanently installed) are excluded from the ESEL per EPRI 3002000704.

The ESEL component selection followed the EPRI guidance outlined in Section 3.2 of EPRI 3002000704.

1. The scope of components is limited to that required to accomplish the core cooling and containment safety functions identified in Table 3-2 of EPRI 3002000704. The instrumentation monitoring requirements for core cooling/containment safety functions

are limited to those outlined in the EPRI 3002000704 guidance, and are a subset of those outlined in the Sequoyah OIP.

2. The scope of components is limited to installed plant equipment and FLEX connections necessary to implement the Sequoyah OIP, as described in Section 2.
3. The scope of components assumes the credited FLEX connection modifications are implemented, and are limited to those required to support a single FLEX success path (i.e., either “Primary” or “Back-up/Alternate”).
4. The “Primary” FLEX success path is to be specified. Selection of the “Back-up/Alternate” FLEX success path must be justified.
5. Phase 3 coping strategies are included in the ESEP scope, whereas recovery strategies are excluded.
6. Structures, systems, and components excluded per the EPRI 3002000704 guidance are:
   - Structures (e.g. containment, reactor building, control building, auxiliary building, etc.).
   - Piping, cabling, conduit, HVAC, and their supports.
   - Manual valves and rupture disks.
   - Power-operated valves not required to change state as part of the FLEX mitigation strategies.
   - Nuclear steam supply system components (e.g. RPV and internals, reactor coolant pumps and seals, etc.).
7. For cases in which neither train was specified as a primary or back-up strategy, then only one train component (generally 'A' train) is included in the ESEL.

3.1.1 ESEL Development

The ESEL was developed by reviewing the Sequoyah Nuclear Plant OIP [3] [4] [5] to determine the major equipment involved in the FLEX strategies. Further reviews of plant drawings (e.g., Piping and Instrumentation Diagrams (P&IDs) and Electrical Single Line Diagrams) were performed to identify the boundaries of the flow paths to be used in the FLEX strategies and to identify specific components in the flow paths needed to support implementation of the FLEX strategies. Boundaries were established at an electrical or mechanical isolation device (e.g., isolation amplifier, valve, etc.) in branch circuits / branch lines off the defined strategy electrical or fluid flow path. P&IDs were the primary reference documents used to identify mechanical components and instrumentation. The flow paths used for FLEX strategies were selected and specific components were identified using detailed equipment and instrument drawings, piping isometrics, electrical schematics and one-line drawings, system descriptions, design basis documents, etc., as necessary. Host components were identified for sub-assemblies.

Cabinets and equipment controls containing relays, contactors, switches, potentiometers, circuit breakers and other electrical and instrumentation that could be affected by high-frequency earthquake motions and that impact the operation of equipment in the ESEL are required to be on the ESEL. These cabinets and components were identified in the ESEL.

For each parameter monitored during the FLEX implementation, a single indication was selected for inclusion in the ESEL. For each parameter indication, the components along the flow path from measurement to indication were included, since any failure along the path would lead to failure of that indication. Components such as flow elements were considered as part of the piping and were not included in the ESEL.

3.1.2 Power Operated Valves

Page 3-3 of EPRI 3002000704 [2] notes that power operated valves not required to change state as part of the FLEX mitigation strategies are excluded from the ESEL. Page 3-2 also notes that "functional failure modes of electrical and mechanical portions of the installed Phase 1 equipment should be considered (e.g. Auxiliary Feedwater (AFW) trips)." To address this concern, the following guidance is applied in the Sequoyah ESEL for functional failure modes associated with power operated valves:

- Power operated valves that remain energized during the ELAP events (such as DC powered valves), were included on the ESEL.
- Power operated valves not required to change state as part of the FLEX mitigation strategies were not included on the ESEL. The seismic event also causes the ELAP event; therefore, the valves are incapable of spurious operation as they would be de-energized.
- Power operated valves not required to change state as part of the FLEX mitigation strategies during Phase 1, and are re-energized and operated during subsequent Phase 2 and 3 strategies, were not evaluated for spurious valve operation as the seismic event that caused the ELAP has passed before the valves are re-powered.

3.1.3 Pull Boxes

Pull boxes were deemed unnecessary to be added to the ESELs as these components provide completely passive locations for pulling or installing cables. No breaks or connections in the cabling were included in pull boxes. Pull boxes were considered part of conduit and cabling, which were excluded in accordance with EPRI 3002000704 [2].

3.1.4 Termination Cabinets

Termination cabinets, including cabinets necessary for FLEX Phase 2 and Phase 3 connections, provide consolidated locations for permanently connecting multiple cables. The termination cabinets and the internal connections provide a completely passive function; however, the cabinets are included in the ESEL to ensure industry knowledge on panel/anchorage failure vulnerabilities is addressed.

3.1.5 Critical Instrumentation Indicators

Critical indicators and recorders are typically physically located on panels/cabinets and are included as separate components; however, seismic evaluation of the instrument indication may be included in the panel/cabinet seismic evaluation (rule-of-the-box).

3.1.6 Phase 2 and 3 Piping Connections

Item 2 in Section 3.1 above notes that the scope of equipment in the ESEL includes "… FLEX connections necessary to implement the Sequoyah OIP [3] [4] [5] as described in Section 2." Item 3 in Section 3.1 also notes that "The scope of components assumes the credited FLEX connection modifications are implemented, and are limited to those required to support a single FLEX success path (i.e., either "Primary" or "Back-up/Alternate")."

Item 6 in Section 3.1 above goes on to explain that "Piping, cabling, conduit, HVAC, and their supports" are excluded from the ESEL scope in accordance with EPRI 3002000704 [2].

Therefore, piping and pipe supports associated with FLEX Phase 2 and Phase 3 connections are excluded from the scope of the ESEP evaluation. However, any active valves in FLEX Phase 2 and Phase 3 connection flow path are included in the ESEL.

3.2 Justification for Use of Equipment That is Not the Primary Means for FLEX Implementation

The Sequoyah Nuclear Plant ESEL is based on the primary means of implementing the FLEX strategy. Therefore, no additional justification is required.

4.0 GROUND MOTION RESPONSE SPECTRUM (GMRS)

4.1 Plot of GMRS Submitted by the Licensee

The Safe Shutdown Earthquake (SSE) control point elevation is defined at the base of the Containment Structures, which corresponds to a depth of 64 ft. (Elevation 641 ft.) and is the deepest structure foundation elevation control point. Table 4-1 shows the GMRS accelerations for a range of frequencies. The GMRS at the control point elevation is shown in Figure 4-1 [9].

Table 4-1: GMRS for Sequoyah Nuclear Plant

Frequency (Hz)	GMRS (g)
100	3.79E-01
90	3.83E-01
80	3.89E-01
70	3.98E-01
60	4.18E-01
50	4.65E-01
40	5.54E-01
35	6.14E-01
30	6.72E-01
25	7.41E-01
20	7.59E-01
15	7.57E-01
12.5	7.49E-01
10	7.06E-01
9	6.82E-01
8	6.53E-01
7	6.11E-01
6	5.58E-01
5	5.00E-01

Table 4-1: GMRS for Sequoyah Nuclear Plant (Continued)

Frequency (Hz)	GMRS (g)
4	4.05E-01
3.5	3.78E-01
3	3.13E-01
2.5	2.50E-01
2	2.30E-01
1.5	1.92E-01
1.25	1.68E-01
1	1.42E-01
0.9	1.36E-01
0.8	1.25E-01
0.7	1.14E-01
0.6	9.98E-02
0.5	8.34E-02
0.4	6.67E-02
0.35	5.84E-02
0.3	5.00E-02
0.25	4.17E-02
0.2	3.34E-02
0.15	2.50E-02
0.125	2.08E-02
0.1	1.67E-02

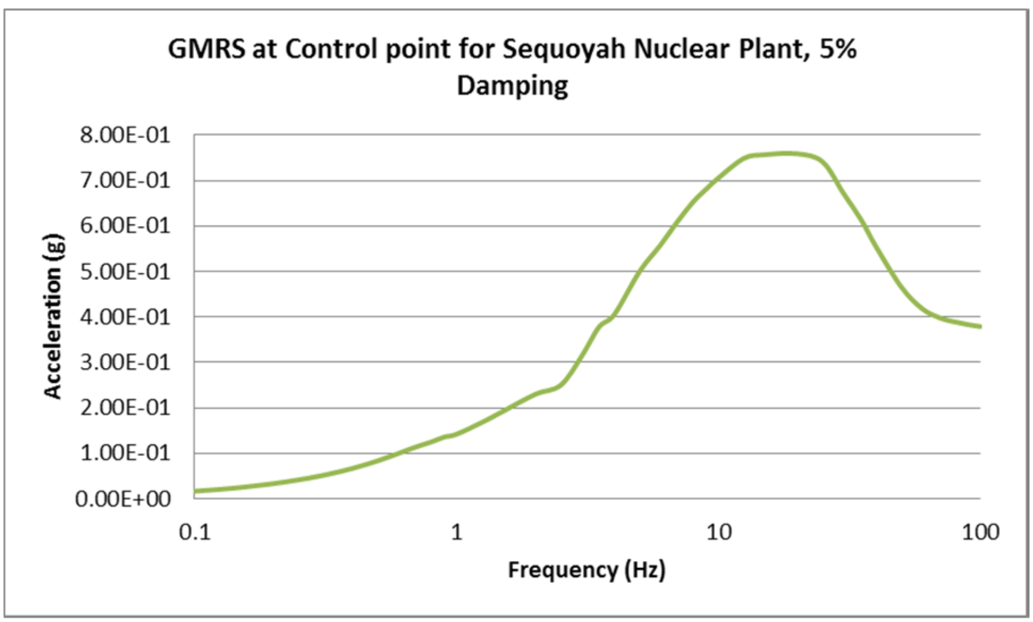

Figure 4-1: GMRS for Sequoyah Nuclear Plant

4.2 Comparison to SSE

The SSE was developed in accordance with 10 CFR Part 100 Appendix A through an evaluation of the maximum earthquake potential for the region surrounding the site. Considering the historic seismicity of the site region, the maximum potential earthquake was determined to be an intensity VIII on the Modified Mercalli Intensity Scale of 1931. The SSE is defined in terms of a Peak Ground Acceleration (PGA) and a design response spectrum. Considering a site intensity of VIII, a PGA of 0.18g was estimated. To be consistent with EPRI Report Nos. EPRI 3002000704 [2] and EPRI 1025287 [22] the site licensing basis earthquake is used for the SSE to GMRS comparison in this report. The design basis earthquake was used in the prior TVA submittal [9] which concluded that a risk analysis would be performed. However, the application of either the design basis or the licensing basis SSE curve to the prior evaluation will not alter the conclusion. The Sequoyah licensing basis SSE is based on a peak ground acceleration of 0.18g with a Housner spectral shape. Table 4-2 shows the spectral acceleration values as a function of frequency for the 5% damped horizontal Sequoyah licensing basis SSE.

Table 4-2: SSE for Sequoyah Nuclear Plant

Frequency (Hz)	Spectral Acceleration (g)
100	0.18
25	0.18
10	0.19
5	0.27
2.5	0.26
1.0	0.14
0.5	0.08

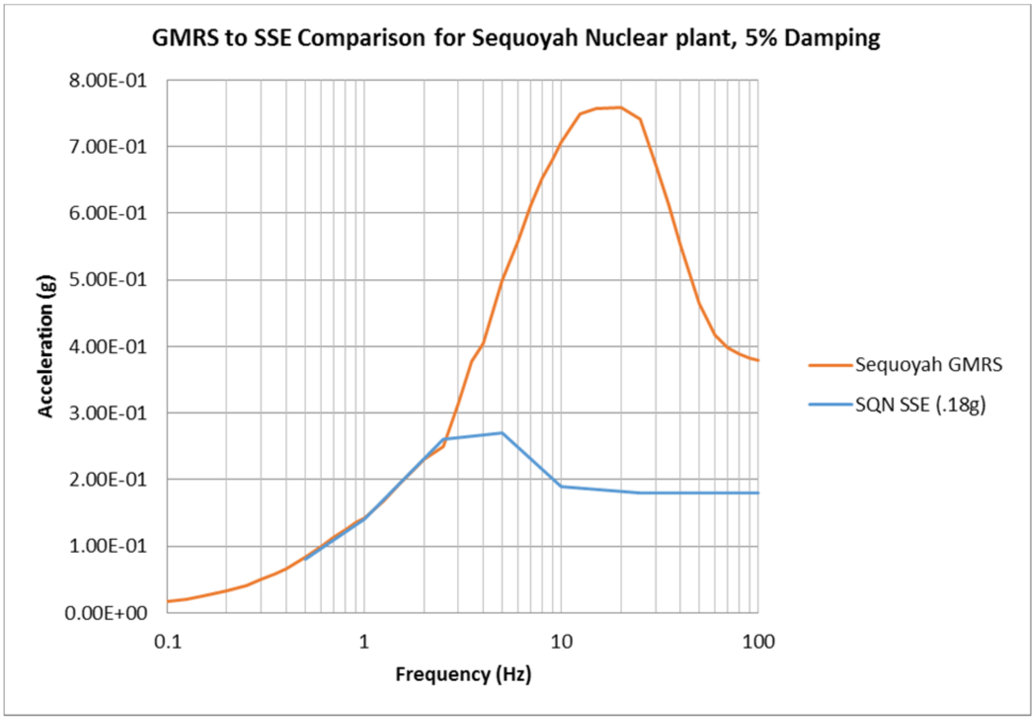

Figure 4-2: GMRS to SSE Comparison for Sequoyah Nuclear Plant

The SSE and the GMRS in the low frequency range up to about 2.5Hz are essentially the same amplitude. The GMRS exceeds the Sequoyah Nuclear Plant SSE beyond about 2.5Hz. As the GMRS exceeds the SSE in the 1 to 10Hz range, the plant does not screen out of the ESEP according to Section 2.2 of EPRI 3002000704 [2]. The two special screening considerations as described in Section 2.2.1 of EPRI 3002000704, namely a) Low Seismic Hazard Site and b) Narrow Band Exceedances in the 1 to 10Hz range do not apply for Sequoyah Nuclear Plant and hence High Confidence of a Low Probability of Failure (HCLPF) evaluations are required.

5.0 REVIEW LEVEL GROUND MOTION (RLGM)

5.1 Description of RLGM Selected

Section 4 of EPRI 3002000704 [2] presents two approaches for developing the RLGM to be used in the ESEP:

1. The RLGM may be derived by linearly scaling the SSE by the maximum ratio of the GMRS/SSE between the 1 and 10 Hz range (not to exceed 2x SSE). In-structure RLGM seismic motions would be derived using existing SSE based in-structure response spectra (ISRS) with the same scale factor.
2. Alternately, licensees who have developed appropriate structural/soil-structure interaction (SSI) models capable of calculating ISRS based on site GMRS/uniform hazard response spectrum (UHRS) input may opt to use these ISRS in lieu of scaled SSE ISRS.

Based on a review of tabulated data in Table 4-1 and the SSE values in Table 4-2, in the range between 1 and 10 Hz the maximum ratio of GMRS to the SSE is calculated to be:

$$SF_{max} = SA_{GMRS}(10\ HZ)/SA_{SSE}\ 10Hz) = 0.71g/0.19g = 3.7$$

Since the computed scale factor is greater than 2.0, the RLGM would be set a level of 2x SSE. This is shown in Table 5-1 and Figure 5-1.

Table 5-1: 2x SSE for Sequoyah Nuclear Plant

Frequency (Hz)	Spectral Acceleration (g)
100	0.36
25	0.36
10	0.38
5	0.54
2.5	0.52
1.0	0.28
0.5	0.16

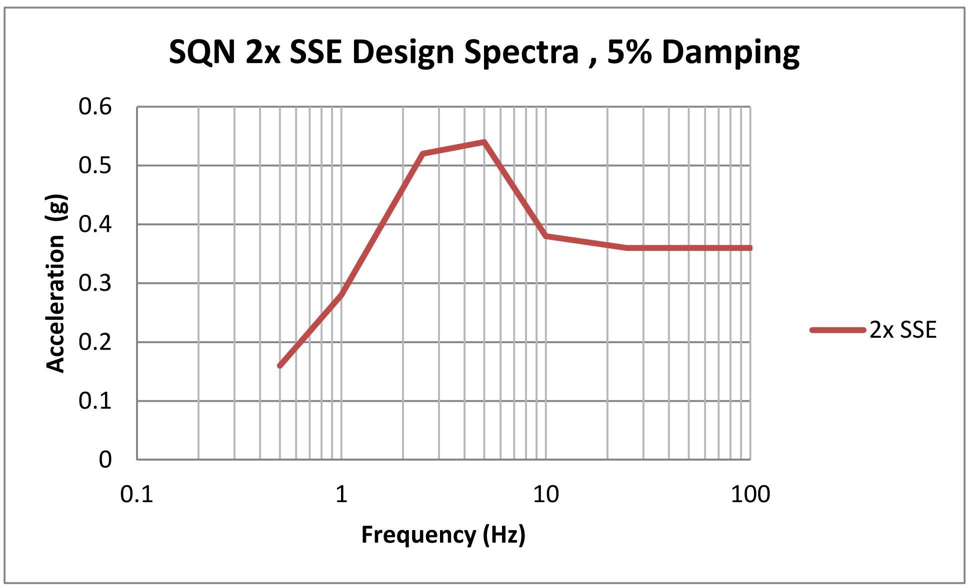

Figure 5-1: 2x SSE for Sequoyah Nuclear Plant

5.2 Method to Estimate In-Structure Response Spectra (ISRS)

A full scope SMA was performed to support the IPEEE for Sequoyah Nuclear Plant Units 1 and 2 [11]. The Review Level Earthquake (RLE) is defined as the NUREG/CR-0098 [10] median spectral shape for rock, anchored to 0.3g PGA. The RLE ISRS were defined at the 84% Non-Exceedance Probability (NEP). To determine the 84% NEP response, a probabilistic method of generating ISRS was used which accounts for the uncertainty in both the ground motion description and in the structural and soil parameters.

Uncertainties in the structural properties are accounted for by representing structural natural frequencies and damping ratios as a log-normally distributed random variable with specified median and Coefficient of Variation (COV) values. A total of thirty (30) earthquake time histories (each with three components) were generated such that the spectral ordinates were log-normally distributed with a COV equal to 0.25, and the 84% NEP value matches the NUREG/CR-0098 median rock shape.

The results of the IPEEE for Sequoyah Nuclear Plant Units 1 and 2 were submitted to the NRC [11]. It should be noted that the NRC [12] took exception to the approach used for Sequoyah Nuclear Plant in that Tennessee Valley Authority (TVA) defined the RLE as being in the free-field at the top of the soil surface, whereas the NRC concluded the RLE should have been defined on a rock. TVA reviewed the NRC Request for Additional Information (RAI) and made adjustments to the originally defied HCLPF capacity of 0.3g. The results of the adjustments of the full scope seismic margin assessment were submitted to the NRC [13], concluding that Sequoyah Nuclear Plant Units 1 and 2 had a plant level HCLPF capacity of 0.23g. Subsequent to TVA's docketed response [13] the NRC issued their Staff Evaluation Report (SER) [14]. In the SER, the NRC recognized the TVA HCLPF capacity value of 0.23g for Sequoyah Nuclear Plant, but also acknowledged a lower HCLPF value of 0.2g that was developed by the staff consultant. During the IPEEE adequacy review, TVA reviewed the NRC staff consultant's opinion regarding a lower HCLPF capacity of 0.2g for Sequoyah Nuclear Plant and concluded that the technical basis described by the NRC staff consultant in the SER is technically correct. Consequently, TVA decided that the assignment of 0.2g HCLPF capacity was appropriate.

Because of the significant effort expended by TVA to develop an updated dynamic analysis of the safety related structures for Sequoyah Nuclear Plant described above and in the IPEEE submittal [11], TVA felt this model provided improved dynamic behavior of Sequoyah Nuclear Plant structures. Consequently, for the purpose of evaluating seismic capacity of ESEP components, TVA chose to scale the 84th percentile values by an increase scale factor of 1.5 (0.3g/0.2g) to achieve a response equivalent to a 0.3g NUREG/CR-0098 shaped response. Figure 5-2 demonstrates that the use of a 0.3g NUREG/CR-0098 shape response bounds 2x SSE for Sequoyah Nuclear Plant from 1 to 10 Hz.

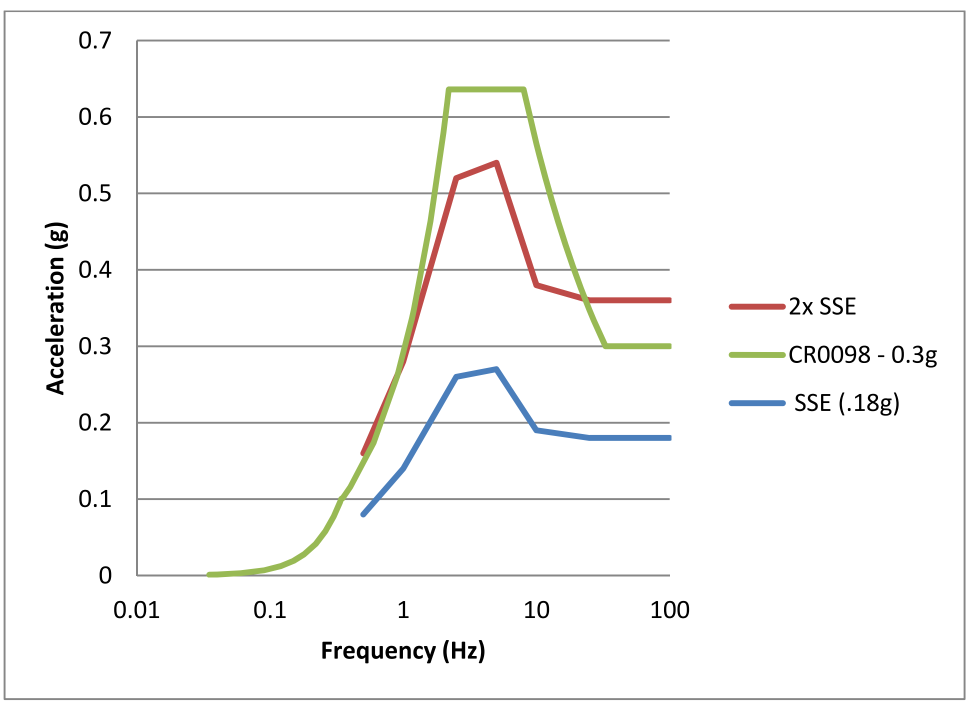

Figure 5-2: NUREG/CR-0098 (0.3g) versus Sequoyah Nuclear Plant SSE

6.0 SEISMIC MARGIN EVALUATION APPROACH

It is necessary to demonstrate that ESEL items have sufficient seismic capacity to meet or exceed the demand characterized by the RLGM. The seismic capacity is characterized as the PGA for which there is a HCLPF. The PGA is associated with a specific spectral shape, in this case the 5%-damped RLGM spectral shape. The HCLPF capacity must be equal to or greater than the RLGM PGA. The criteria for seismic capacity determination are given in Section 5 of EPRI 3002000704 [2].

There are two basic approaches for developing HCLPF capacities:

1. Deterministic approach using the conservative deterministic failure margin (CDFM) methodology of EPRI NP-6041-SL, A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Revision 1) [15].
2. Probabilistic approach using the fragility analysis methodology of EPRI TR-103959, Methodology for Developing Seismic Fragilities [16].

6.1 Summary of Methodologies Used

Sequoyah Nuclear Plant completed SMA for Units 1 and 2 in 1995. The SMA is documented in [11] that consisted of development of a Safe Shutdown Equipment List (SSEL), probabilistic approach for determining seismic demand based on 84% NEP, new building models, associated generation of ISRS, screening walkdowns, and HCLPF capacity calculations.

The screening walkdowns used the screening tables from Chapter 2 of EPRI NP-6041-SL [15]. The walkdowns were conducted by engineers trained in EPRI NP-6041-SL (the engineers attended the EPRI SMA Add-On course in addition to the SQUG Walkdown Screening and Seismic Evaluation Training Course), and were documented on Screening Evaluation Work Sheets from EPRI NP-6041-SL. Anchorage capacity calculations used the CDFM criteria from EPRI NP-6041-SL. The seismic demand is based on a probabilistic approach that involves the generation of an ensemble of artificial earthquake (ground motion) time histories as well as structural and soil parameters values. The probabilistic approach of determining seismic demand is based on guidance from EPRI NP-6041-SL:

"For the Specified SME, the elastic computed response (SME demand) of structures and components mounted thereon should be defined at the 84% non-exceedance probability (NEP)."

The results of the probabilistic approach for development of seismic demand for Sequoyah Nuclear plant is documented in [11].

Figure 6-1 shows the fit of the 84th percentile of the ensemble of the 30 response spectra (of the 30 generated time histories) to the target spectral shape (NUREG/CR-0098 median rock spectrum). Note this figure represents the input motion assuming the target spectrum is at the top of free field on the soil surface. Figure 6-2 shows the adjusted Sequoyah Nuclear Plant IPEEE HCLPF RLE response spectrum adjusted to 0.2g, compared to the ESEP RLGM response spectrum used for the Sequoyah Nuclear Plant ESEP. Note both spectra are rock input motions at the base of the containment structure. This demonstrates that the ESEP RLGM envelopes the RLGM used for SMA at all frequencies by an amplitude factor of 1.5.

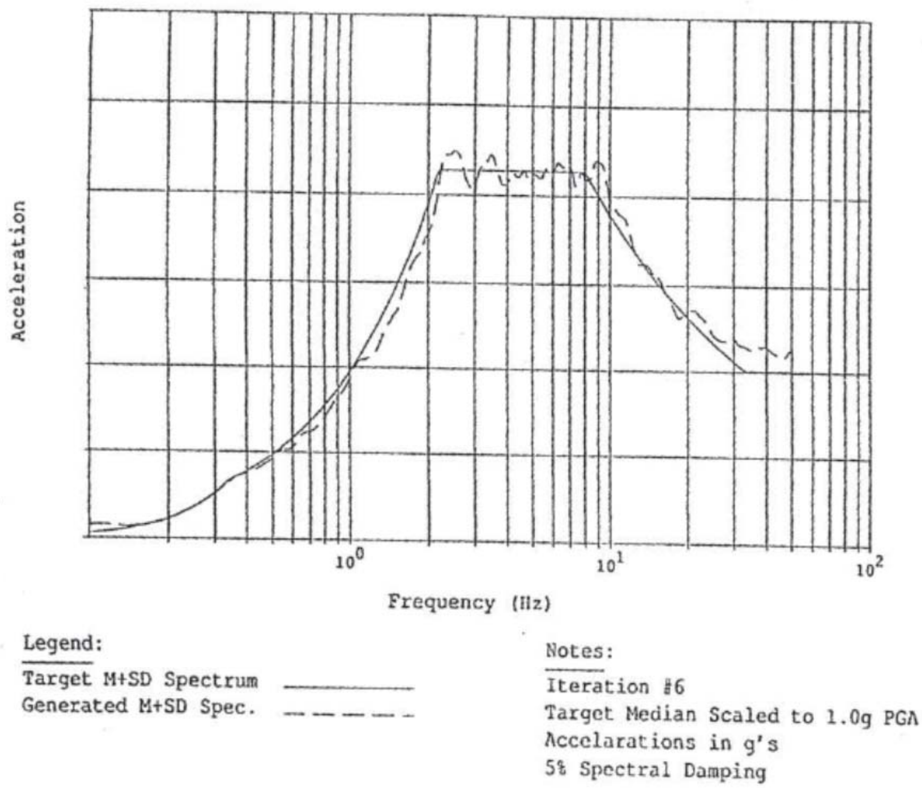

Figure 6-1: 84th Percentile of the Ensemble of the 30 Response Spectra

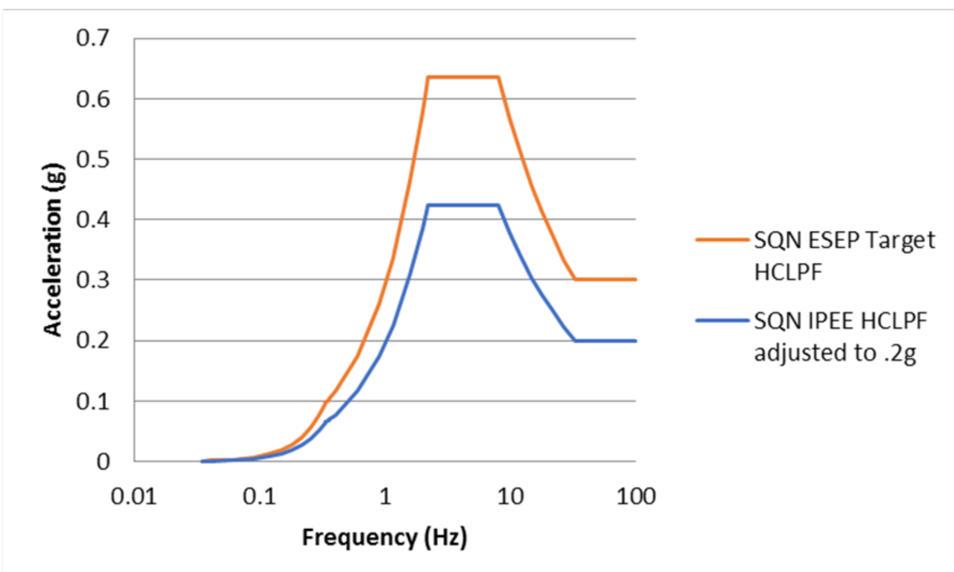

Figure 6-2: Sequoyah Nuclear Plant IPEEE adjusted HCLPF vs ESEP Target HCLPF

6.2 HCLPF Screening Process

For ESEP, the components are screened at RLGM (NUREG/CR-0098 curve) anchored at 0.3g PGA. The screening tables in EPRI NP-6041-SL [15] are based on ground peak spectral accelerations of 0.8g and 1.2g. These both exceed the RLGM peak spectral acceleration. The anchorage capacity calculations were on based floor response spectra developed for the Sequoyah Nuclear Plant IPEEE and scaled to the adjusted RLGM. Equipment for which the screening caveats were met and for which the anchorage capacity exceeded the RLGM seismic demand, can be screened out from ESEP seismic capacity determination because the HCLPF capacity exceeds the RLGM.

The Unit 1 ESEL contains 182 items. Of these, 27 are valves, both power-operated and relief. In accordance with Table 2-4 of EPRI NP-6041-SL [15], active valves may be assigned a functional capacity of 0.8g peak spectral acceleration without any review other than looking for valves with large extended operators on small diameter piping, and anchorage is not a failure mode. Therefore, valves on the ESEL may be screened out from ESEP seismic capacity determination, subject to the caveat regarding large extended operators on small diameter lines.

The non-valve components in the ESEL are generally screened based on the SMA methodology. If the SMA showed that the component met the EPRI NP-6041-SL screening caveats and the CDFM capacity exceeded the RLE demand, then the component can be screened out from the ESEP capacity determination.

6.3 Seismic Walkdown Approach

6.3.1 Walkdown Approach

Walkdowns were performed in accordance with the criteria provided in Section 5 of EPRI 3002000704 [2], which refers to EPRI NP-6041-SL [15] for the Seismic Margin Assessment process. Pages 2-26 through 2-30 of EPRI NP-6041-SL describe the seismic walkdown criteria, including the following key criteria.

> *"The SRT [Seismic Review Team] should "walk by" 100% of all components which are reasonably accessible and in non-radioactive or low radioactive environments. Seismic capability assessment of components which are inaccessible, in high-radioactive environments, or possibly within contaminated containment, will have to rely more on alternate means such as photographic inspection, more reliance on seismic reanalysis, and possibly, smaller inspection teams and more hurried inspections. A 100% "walk by" does not mean complete inspection of each component, nor does it mean requiring an electrician or other technician to de-energize and open cabinets or panels for detailed inspection of all components. This walkdown is not intended to be a QA or QC review or a review of the adequacy of the component at the SSE level.*
>
> *If the SRT has a reasonable basis for assuming that the group of components are similar and are similarly anchored, then it is only necessary to inspect one component out of this group. The "similarity-basis" should be developed before the walkdown during the seismic capability preparatory work (Step 3) by reference to drawings, calculations or specifications. The one component or each type which is selected should be thoroughly inspected which probably does mean de-energizing and opening cabinets or panels for this very limited sample. Generally, a spare representative component can be found so as to enable the inspection to be performed while the plant is in operation. At least for the one component of each type which is selected, anchorage should be thoroughly inspected.*
>
> *The walkdown procedure should be performed in an ad hoc manner. For each class of components the SRT should look closely at the first items and compare the field configurations with the construction drawings and/or specifications. If a one-to-one correspondence is found, then subsequent items do not have to be inspected in as great a detail. Ultimately the*

walkdown becomes a "walk by" of the component class as the SRT becomes confident that the construction pattern is typical. This procedure for inspection should be repeated for each component class; although, during the actual walkdown the SRT may be inspecting several classes of components in parallel. If serious exceptions to the drawings or questionable construction practices are found then the system or component class must be inspected in closer detail until the systematic deficiency is defined.

The 100% "walk by" is to look for outliers, lack of similarity, anchorage which is different from that shown on drawings or prescribed in criteria for that component, potential SI [Seismic Interaction] problems, situations that are at odds with the team members' past experience, and any other areas of serious seismic concern. If any such concerns surface, then the limited sample size of one component of each type for thorough inspection will have to be increased. The increase in sample size which should be inspected will depend upon the number of outliers and different anchorages, etc., which are observed. It is up to the SRT to ultimately select the sample size since they are the ones who are responsible for the seismic adequacy of all elements which they screen from the margin review. Appendix D gives guidance for sampling selection."

6.3.2 Application of Previous Walkdown Information

Several ESEL items were previously walked down during the Sequoyah Nuclear Plant Units 1 and 2 seismic IPEEE program. Those walkdown results were reviewed and the following steps were taken to confirm that the previous walkdown conclusions remained valid.

- A walk by was performed to confirm that the equipment material condition and configuration is consistent with the walkdown conclusions and that no new significant interactions related to block walls or piping attached to tanks exist.
- If the ESEL item was screened out based on the previous walkdown, that screening evaluation was reviewed and reconfirmed for the ESEP.

Except for inaccessible items as described below in Section 7, in all cases it was determined that the HCLPF capacities established for these items under the seismic IPEEE program remained valid. Thus, all ESEL components that were part of the IPEEE program have a HCLPF capacity of 0.3g or greater and are thus adequate for ESEP [9].

6.3.3 Significant Walkdown Findings

Consistent with the guidance from EPRI NP-6041-SL [15], no significant outliers and only one (1) anchorage concern was identified during the Sequoyah Nuclear Plant seismic walkdowns. The following findings were noted during the walkdowns.

One anchor for Sequoyah Unit 2 TDAFW Pump Control Panel 2-L-381 was observed to be significantly corroded. An evaluation was performed of the configuration assuming that the anchor was inactive. The evaluation determined that the configuration (using 3 of 4 anchors) satisfied design requirements. The corroded anchored is scheduled to be replaced in upcoming U2R20 outage.

Based on walkdown results, HCLPF capacity evaluations were recommended for the following twelve (12) components, on a bounding basis:

- Turbine Driven Auxiliary Feedwater Pump
- Instrument Rack

- R Panels
- Benchboard M Panels
- Vertical M and L Panels
- Main Control Room Ceiling
- Wall Mounted Panel
- Boric Acid Storage Tank
- TDAFWP Control Panel
- PHMS Transformers and Distribution Panel
- Valves
- Block Walls

6.4 HCLPF Calculation Process

ESEL items not included in the previous IPEEE evaluations at Sequoyah were evaluated using the criteria in EPRI NP-6041 [7]. Those evaluations included the following steps:

- Performing seismic capability walkdowns for equipment not included in previous seismic walkdowns (SQUG, IPEEE, or NTTF 2.3) to evaluate the equipment installed plant conditions
- Performing screening evaluations using the screening tables in EPRI NP-6041 as described in Section 6.2 and
- Performing HCLPF calculations considering various failure modes that include both structural failure modes (e.g. anchorage, load path etc.) and functional failure modes.

All HCLPF calculations were performed using the CDFM methodology and are documented in a TVA Calculation: CDQ999 2014 000140 “SQN Expedited Seismic Evaluation Process (ESEP) HCLPF Capacity Calculation” [17].

6.5 Functional Evaluations of Relays

ESEP considers cabinets and equipment controls containing relays, contactors, switches, circuit breakers and other electrical and instrumentation components that could be affected by high-frequency earthquake motions and that impact the operation of equipment in the ESEL.

A full scope SMA was performed to support the IPEEE for Sequoyah Nuclear Plant Units 1 and 2 as summarized in the IPEEE submittal [11]. A slightly modified version of the EPRI NP-6041-SL [15] recommended approach was implemented for the Sequoyah Nuclear Plant IPEEE in order to increase efficiency. All relays were screened out by: (1) ground rules and assumptions; (2) comparison of design qualification test spectrum (or generic equipment ruggedness spectra) with Seismic Margin Earthquake (SME) in-cabinet response spectrum; or (3) analysis showing that relay chatter does not disable safe shutdown equipment without the possibility of recovery. In all cases, equipment actuation was determined to not affect the safe shutdown capability of the equipment in the SSEL. Low ruggedness relays were screened out only if the effects of chatter could be reset by operator action. All but two low-ruggedness relays fell into this category. The remaining two relays were found to not be used in safe shutdown equipment at Sequoyah Nuclear Plant. The principal conclusion from the

IPEEE relay evaluation was that safe shutdown systems will not be adversely affected by relay malfunction during or after an SME.

For the Sequoyah Nuclear Plant ESEP analysis, an evaluation was performed to identify components that are (1) needed for FLEX implementation, (2) not on the Sequoyah IPEEE SSEL, and (3) that have the potential for relay chatter issues. The only cases identified are the FCV-1-17 and FCV-1-18 steam isolation valves that can isolate the steam supply to the TDAFW pump. In the event of a steam line break, both of these valves can receive a close signal if high temperature is detected in the TDAFW pump room. However, because these valves are motor operated valves (MOVs), with a Loss of Offsite Power (LOOP) the valves will not isolate even with a spurious "close" signal. Therefore, these valves do not present a problem for successful FLEX implementation. No other relay chatter cases were identified. No seismic capacity to demand relay evaluations were necessary for ESEP.

6.6 Tabulated ESEL HCLPF Values (Including Key Failure Modes)

Tabulated ESEL HCLPF values are provided in Attachment B. The following notes apply to the information in the tables.

- Items previously included in the seismic IPEEE programs are not listed. Walk-by verifications re-confirmed the HCLPF capacities from the IPEEE, and the IPEEE 0.3g RLE HCLPF capacity exceeds the RLGM [18].
- HCLPF capacity evaluations were performed for the non-IPEEE items on the ESEL, addressing both structural/anchorage and functional failure modes. The HCLPF capacity of each item is listed in the tables, with associated governing failure mode.
- Rugged items not specifically evaluated are conservatively assigned a 0.50g HCLPF capacity based on the EPRI screening tables or by engineering judgment.
- New pre-staged and permanently installed FLEX items are not listed. TVA design criteria SQN-DC-V-48.0 [19] requires that new FLEX items have HCLPF capacity exceeding the RLGM.

All ESEP components have a HCLPF capacity greater than the RLGM for the frequency range of 1 to 10Hz.

7.0 INACCESSIBLE ITEMS

7.1 Identification of ESEL Item Inaccessible for Walkdowns

There are four (4) valves and seven (7) instrument racks that could not be walked down since they are in the Unit 1 Reactor Building (inaccessible area). These components' Unit 2 counter parts were walked down during the recent Unit 2 outage. The Unit 2 components were determined to be acceptable. It is expected that the same conclusion can be made for the Unit 1 components. The following is a list of the Unit 1 components located in the Reactor Building that were not walked down:

- 1-FCV 63-118 - Cold leg Accumulator Isolation Valve #1
- 1-FVC 63-67 - Cold leg Accumulator Isolation Valve #4
- 1-FCV 63-80 - Cold leg Accumulator Isolation Valve #3
- 1-FCV 63-98 - Cold leg Accumulator Isolation Valve #2
- Instrument Rack 1-L-182 located in Fan Room 2

- Instrument Rack 1-L-183 located in Fan Room 1
- Instrument Rack 1-L-179
- Instrument Rack 1-L-185
- Instrument Rack 1-L-704
- Instrument Rack 1-L-706
- Instrument Rack 1-L-194

Also there are two (2) panels that could not be walked down in the Unit 1 Auxiliary Building because the components are in a Contaminated and Radiation Area. They are:

- Instrument Rack 1-L-196
- Instrument Rack 1-L-216

In addition, a walk by inside Unit 1 containment was not possible.

7.2 Planned Walkdown / Evaluation Schedule / Close Out

The following Unit 1 components will be walked down in upcoming Unit 1 outage:

- FCV 63-118 - Cold Leg Accumulator Isolation Valve #1
- FVC 63-67 - Cold Leg Accumulator Isolation Valve #4
- FCV 63-80 - Cold Leg Accumulator Isolation Valve #3
- FCV 63-98 - Cold Leg Accumulator Isolation Valve #2
- Instrument Rack 1-L- 182 located in Fan Room 2
- Instrument Rack 1-L- 183 located in Fan Room 1
- Instrument Rack 1-L-179
- Instrument Rack 1-L-185
- Instrument Rack 1-L-704
- Instrument Rack 1-L-706
- Instrument Rack 1-L-194
- Instrument Rack 1-L-196
- Instrument Rack 1-L-216

In addition, as performed inside the Unit 2 containment, a walk by will be conducted to verify that HCLPF capacity of at least 0.3g is maintained for IPEEE items on the ESEL. It is expected that the same conclusions will be made for the Unit 1 components that were completed for the counterpart components in Unit 2.

8.0 ESEP CONCLUSIONS AND RESULTS

8.1 Supporting Information

Sequoyah Nuclear Plant Units 1 and 2 have performed the ESEP as an interim action in response to the NRC's 50.54(f) letter [1]. It was performed using the methodologies in the NRC endorsed guidance in EPRI 3002000704 [2].

The ESEP provides an important demonstration of seismic margin and expedites plant safety enhancements through evaluations and potential near-term modifications of plant equipment that can be relied upon to protect the reactor core following beyond design basis seismic events.

The ESEP is part of the overall Sequoyah Nuclear Plant Units 1 and 2 in response to the NRC's 50.54(f) letter. On March 12, 2014, NEI submitted to the NRC results of a study [21] of seismic core damage risk estimates based on updated seismic hazard information as it applies to operating nuclear reactors in the Central and Eastern United States (CEUS). The study concluded that "site-specific seismic hazards show that there has not been an overall increase in seismic risk for the fleet of U.S. plants" based on the re-evaluated seismic hazards. As such, the "current seismic design of operating reactors continues to provide a safety margin to withstand potential earthquakes exceeding the seismic design basis."

The NRC's May 9, 2014 NTTF 2.1 Screening and Prioritization letter [20] concluded that the "fleet wide seismic risk estimates are consistent with the approach and results used in the Gl-199 safety/risk assessment." The letter also stated that "As a result, the staff has confirmed that the conclusions reached in Gl-199 safety/risk assessment remain valid and that the plants can continue to operate while additional evaluations are conducted."

An assessment of the change in seismic risk for Sequoyah Nuclear Plant Units 1 and 2 was included in the fleet risk evaluation submitted in the March 12, 2014 NEI letter [21]; therefore, the conclusions in the NRC's May 9 letter also apply to Sequoyah Nuclear Plant Units 1 and 2.

In addition, the March 12, 2014 NEI letter provided an attached "Perspectives on the Seismic Capacity of Operating Plants," which (1) assessed a number of qualitative reasons why the design of SSCs inherently contain margin beyond their design level, (2) discussed industrial seismic experience databases of performance of industry facility components similar to nuclear SSCs, and (3) discussed earthquake experience at operating plants.

The fleet of currently operating nuclear power plants was designed using conservative practices, such that the plants have significant margin to withstand large ground motions safely. This has been borne out for those plants that have actually experienced significant earthquakes. The seismic design process has inherent (and intentional) conservatisms which result in significant seismic margins within SSCs. These conservatisms are reflected in several key aspects of the seismic design process, including:

- Safety factors applied in design calculations
- Damping values used in dynamic analysis of SSCs
- Bounding synthetic time histories for ISRS calculations
- Broadening criteria for ISRS
- Response spectra enveloping criteria typically used in SSC analysis and testing applications

- Response spectra based frequency domain analysis rather than explicit time history based time domain analysis
- Bounding requirements in codes and standards
- Use of minimum strength requirements of structural components (concrete and steel)
- Bounding testing requirements
- Ductile behavior of the primary materials (that is, not crediting the additional capacity of materials such as steel and reinforced concrete beyond the essentially elastic range, etc.)

These design practices combine to result in margins such that the SSCs will continue to fulfill their functions at ground motions well above the SSE.

The intent of the ESEP is to perform an interim action in response to the NRC's 50.54(f) letter to demonstrate seismic margin through a review of a subset of the plant equipment that can be relied upon to protect the reactor core following beyond design basis seismic events. In order to complete the ESEP in an expedited amount of time, the RLGM used for the ESEP evaluation is a scaled version of the plant's SSE rather than the actual GMRS. To more fully characterize the risk impacts of the seismic ground motion represented by the GMRS on a plant specific basis, a more detailed seismic risk assessment (SPRA or risk-based SMA) is to be performed in accordance with EPRI 1025287 [22] . As identified in the Sequoyah Nuclear Plant Units 1 and 2 Seismic Hazard and GMRS submittal [9], Sequoyah Nuclear Plant Units 1 and 2 screens in for a risk evaluation. The complete risk evaluation will more completely characterize the probabilistic seismic ground motion input into the plant, the plant response to that probabilistic seismic ground motion input, and the resulting plant risk characterization. Sequoyah Nuclear Plant Units 1 and 2 will complete that evaluation in accordance with the schedule identified in NEI's letter dated April 9, 2013 [23] and endorsed by the NRC in their May 7, 2013 letter [24].

8.2 Identification of Planned Modifications

One modification was identified in unit 2, to repair a corroded anchor observed for TDAFW pump Control Panel 2-L-381.

8.3 Modification Implementation Schedule

Plant modifications will be performed in accordance with the schedule identified in NEI letter dated April 9, 2013 [23], which states that plant modifications not requiring a planned refueling outage will be completed by December 2016 and modifications requiring a refueling outage will be completed within two planned refueling outages after December 31, 2014.

The plant modification identified in Section 8.2 requires a refueling outage that will be performed in the upcoming unit 2 refueling outage (U2R20) and will be completed by the end of December 2015.

8.4 Summary of Regulatory Commitments

The following actions will be performed as a result of the ESEP.

Table 8-1: Summary of Regulatory Commitments

Action #	Equipment ID	Equipment Description	Action Description	Completion Date
1	NA	N/A	Perform seismic walkdowns, generate HCLPF calculations, and design and implement any necessary modifications for Unit 1 inaccessible items listed in Section 7.1	No later than the end of the second planned Unit 1 refueling outage after December 31, 2014
2	2-L-381	TDAFP Control Panel	Modify anchorage to replace corroded anchor such that HCLPF >RLGM	No later than the end of U2R20 Refueling Outage, December 31, 2015
3	N/A	N/A	Submit a letter to NRC summarizing the HCLPF results of Item 1 and confirming implementation of the plant modifications associated with Item 2	Within 60 days following completion of ESEP activities, including Items 1 through 2

9.0 REFERENCES

1. NRC (E Leeds and M Johnson) Letter to All Power Reactor Licensees et al., “Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3 and 9.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident,” March 12, 2012.
2. EPRI 3002000704, “Seismic Evaluation Guidance, Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic,” May 2013.
3. TVA Letter to U.S. NRC, “Tennessee Valley Authority – Overall Integrated Plan in Response to March 12, 2012, Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond Design Basis External Events (Order Number EA-12-049),” February 28, 2013.
4. TVA Letter to U.S. NRC, “Second Six-Month Status Report and Revised Overall Integrated Plan in Response to the March 12, 2012, Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049) for Sequoyah Nuclear Plant,” February 28, 2014.
5. TVA Letter to U.S. NRC, “Third Six-Month Status Report in Response to the March 12, 2012, Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond Design Basis External Events (Order Number EA-12-049) for Browns Ferry Nuclear Plant (TAC Nos. MF0864 and MF0865),” August 28, 2014.
6. TVA Drawing 1-47W811-1-FLEX, “Flow Diagram Safety Injection System,” Revision 74 (Modified for FLEX).
7. NEI 12-06, “Diverse and Flexible Coping Strategies (FLEX) Implementation Guide,” Revision 0, August 2012.
8. AREVA NP Document 51-9217523-005, “ESEP Expedited Seismic Equipment List (ESEL) – Sequoyah Nuclear Plant.”
9. TVA Letter to U.S. NRC, letter number CNL-14-038, “Tennessee Valley Authority’s Seismic Hazard and Screening Report (CEUS Sites), response to NRC Request for Information Pursuant to 10 CFR 50.54(f) regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident”, March 31, 2014
10. U.S. NRC NUREG/CR-0098, “Development of Criteria for Seismic Review of Selected Nuclear Power Plants,” May 1978.
11. TVA Letter from R. H. Shell to U.S. NRC, “Sequoyah Nuclear Plant (SQN) – Generic Letter GL 88-20, Supplement No. 4, Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities – 10CFR 50.54(f),” June 29, 1995.
12. NRC Letter to TVA, “Sequoyah Nuclear Plant, Units 1 and 2 – Request for Additional Information on Individual Plant Examination of External Events (TAC Nos. M86374 and M86375),” August 2, 2000.
13. Letter from Pedro Salas to NRC, “Sequoyah Nuclear Plant Units 1 and 2 – Response to Request for Additional Information of the Individual Plant Examination of External Events (IPEEE) (TAC Nos. M83674 and M83675),” December 5, 2000.

14. Letter from NRC to J. A. Scalice, "Sequoyah Nuclear Plant, Units 1 and 2 – Review of Sequoyah Individual Plant Examination of External Events Submittal (TAC Nos. M83764 and M83675)," February 21, 2001.

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

16. EPRI TR-103959, "Methodology for Developing Seismic Fragilities," July 1994.

17. TVA Calculation CDQ999 2014 000140, "SQN Expedited Seismic Evaluation Process (ESEP) HCLPF Capacity Calculation."

18. TVA letter to U.S. NRC, Letter number CNL-14-013, "Highlights of Improvements to the Sequoyah Nuclear Plant IPEEE Seismic Analysis Results and Supplemental Response to NRC Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) regarding the Sequoyah Nuclear Plant Unit 1 Seismic Walkdown Results of Recommendations 2.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident", January 31, 2014.

19. TVA Design Criteria, SQN-DC-V-48.0, Revision 4, "FLEX Response System."

20. NRC (E. Leeds) Letter to All Power Reactor Licensees et al., "Screening and Prioritization Results Regarding Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(F) Regarding Seismic Hazard Re-Evaluations for Recommendation 2.1 of the Near-Term Task Force Review of Insights From the Fukushima Dai-Ichi Accident," May 9, 2014.

21. Nuclear Energy Institute (NEI), A. Pietrangelo, Letter to D. Skeen of the USNRC, "Seismic Core Damage Risk Estimates Using the Updated Seismic Hazards for the Operating Nuclear Plants in the Central and Eastern United States," March 12, 2014.

22. EPRI 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," February 2013.

23. Nuclear Energy Institute (NEI), A. Pietrangelo, Letter to D. Skeen of the USNRC, "Proposed Path Forward for NTTF Recommendation 2.1: Seismic Reevaluations," April 9, 2013. NRC Adams Accession No. ML13101A379.

24. NRC (E Leeds) Letter to NEI (J Pollock), "Electric Power Research Institute Final Draft Report xxxxx, "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.

ATTACHMENT A – SEQUOYAH NUCLEAR PLANT ESEL

TABLE A-1: Expedited Seismic Equipment List (ESEL) for Sequoyah Nuclear Plant

ESEL Item Number	Equipment		Operating State		Notes/Comments
	ID	Description	Normal State	Desired State	
1	VLV-1-512	Steam Generator #3 Main Steam Safety Valve	Operational	Operational	-
2	VLV-1-517	Steam Generator #2 Main Steam Safety Valve	Operational	Operational	-
3	VLV-1-522	Steam Generator #1 Main Steam Safety Valve	Operational	Operational	-
4	VLV-1-527	Steam Generator #4 Main Steam Safety Valve	Operational	Operational	-
5	PCV-1-5	Steam Generator #1 ARV (SG PORV)	Operational	Operational	Fail closed on loss of Train A essential air
6	PCV-1-12	Steam Generator #2 ARV (SG PORV)	Operational	Operational	Fail closed on loss of Train B essential air
7	PCV-1-23	Steam Generator #3 ARV (SG PORV)	Operational	Operational	Fail closed on loss of Train A essential air
8	PCV-1-30	Steam Generator #4 ARV (SG PORV)	Operational	Operational	Fail closed on loss of Train B essential air
9	PCV-1-5	Steam Generator #1 ARV (SG PORV) Hand-wheel	Operational	Operational	Local control of steam generator PORV during ELAP
10	Air Bottles	Steam Generator #2 ARV (SG PORV) Local Control Station	Operational	Operational	Emergency control station per EA-1-2 FLEX compressed air cylinders
11	Air Bottles	Steam Generator #3 ARV (SG PORV) Local Control Station	Operational	Operational	Emergency control station per EA-1-2 FLEX compressed air cylinders
12	PCV-1-30	Steam Generator #4 ARV (SG PORV) Hand-wheel	Operational	Operational	Local control of steam generator PORV during ELAP

TABLE A-1: Expedited Seismic Equipment List (ESEL) for Sequoyah Nuclear Plant (Continued)

ESEL Item Number	Equipment		Operating State		Notes/Comments
	ID	Description	Normal State	Desired State	
13	L-501	PCV-1-12 Local Control Station	Operational	Operational	Emergency control station per EA-1-2
14	L-502	PCV-1-23 Local Control Station	Operational	Operational	Emergency control station per EA-1-2
15	PMP-3-142	Turbine Driven AFW Pump	Standby	Operating	Automatic start on LOOP
16	FCV-1-51	TDAFW Pump Trip and Throttle Valve	Closed	Open	Normal power supply is from 125V Vital Battery Board III
17	FCV-1-52	TDAFW Pump Governor Valve	Closed	Open	Fails open on loss of DC control power
18	XS-46-57	AFWT A-S Backup Control Transfer Switch	Operational	Operational	These components are on the ESEL if manual operation of TDAFW is implemented
19	HS-1-51B	TDAFW Pump Trip and Throttle Valve Handswitch	Operational	Operational	-
20	SI-46-56B	TDAFW Pump Speed Indicator	Operational	Operational	-
21	FIC-46-57	TDAFW Pump Master Speed Controller	Operational	Operational	-
22	L-381	TDAFW Pump Control Panel	Operational	Operational	-
23	PI-3-138	TDAFW Pump Discharge Pressure Indicator	Operational	Operational	-
24	L-215	AFW Flow Monitoring Panel	Operational	Operational	Portable delta press gauge can be used for local monitoring of AFW flow

TABLE A-1: Expedited Seismic Equipment List (ESEL) for Sequoyah Nuclear Plant (Continued)

ESEL Item Number	Equipment		Operating State		Notes/Comments
	ID	Description	Normal State	Desired State	
25	FT-3-147	AFW Flow to Steam Generator #3 Flow Transmitter	Operational	Operational	-
26	FT-3-155	AFW Flow to Steam Generator #2 Flow Transmitter	Operational	Operational	-
27	FT-3-163	AFW Flow to Steam Generator #1 Flow Transmitter	Operational	Operational	-
28	FT-3-170	AFW Flow to Steam Generator #4 Flow Transmitter	Operational	Operational	-
29	FI-3-147B	AFW Flow to Steam Generator #3 Flow Indication	Operational	Operational	-
30	FI-3-155B	AFW Flow to Steam Generator #2 Flow Indication	Operational	Operational	-
31	FI-3-163B	AFW Flow to Steam Generator #1 Flow Indication	Operational	Operational	-
32	FI-3-170B	AFW Flow to Steam Generator #4 Flow Indication	Operational	Operational	-
33	L-341	AFW Flow to Steam Generator #3 Flow Transmitter Rack	Operational	Operational	-
34	L-217	AFW Flow to Steam Generator #2 Flow Transmitter Rack	Operational	Operational	-
35	L-216	AFW Flow to Steam Generator #1 Flow Transmitter Rack	Operational	Operational	-
36	L-703	AFW Flow to Steam Generator #4 Flow Indication Rack	Operational	Operational	-

TABLE A-1: Expedited Seismic Equipment List (ESEL) for Sequoyah Nuclear Plant (Continued)

ESEL Item Number	Equipment		Operating State		Notes/Comments
	ID	Description	Normal State	Desired State	
37	LCV-3-172	Steam Generator #3 Level Control Valve	Closed	Open	Fails open on loss of AX power or control air. Backup air supply bottles available. Manual operation with hand-wheel is available.
38	LCV-3-173	Steam Generator #2 Level Control Valve	Closed	Open	Fails open on loss of AC power or control air. Backup air supply bottle available. Manual operation with hand-wheel is available.
39	LCV-3-174	Steam Generator #1 Level Control Valve	Closed	Open	Fails open on loss of AC power or control air. Backup air supply bottles available. Manual operation with hand-wheel is available.
40	LCV-3-175	Steam Generator #4 Level Control Valve	Closed	Open	Fails open on loss of AC power or control air. Backup air supply bottles available. Manual operation with hand-wheel is available.
41	XS-3-172	Steam Generator #3 Level Control Valve Transfer Switch	Operational	Operational	-
42	XS-3-173	Steam Generator #2 Level Control Valve Transfer Switch	Operational	Operational	-
43	XS-3-174	Steam Generator #1 Level Control Valve Transfer Switch	Operational	Operational	-
44	XS-3-175	Steam Generator #4 Level Control Valve Transfer Switch	Operational	Operational	-
45	L-11A	Steam Generator Level Control Panel	Operational	Operational	-
46	L-11B	Steam Generator Level Control Panel	Operational	Operational	-
47	HS-3-172B	Steam Generator #3 Level Control Valve Hand Switch	Operational	Operational	-

TABLE A-1: Expedited Seismic Equipment List (ESEL) for Sequoyah Nuclear Plant (Continued)

ESEL Item Number	Equipment		Operating State		Notes/Comments
	ID	Description	Normal State	Desired State	
48	HS-3-173B	Steam Generator #2 Level Control Valve Hand Switch	Operational	Operational	-
49	HS-3-174B	Steam Generator #1 Level Control Valve Hand Switch	Operational	Operational	-
50	HS-3-175B	Steam Generator #4 Level Control Valve Hand Switch	Operational	Operational	-
51	L-661	Steam Generator #2 Level Control Valve FLEX Backup Air Station	Operational	Operational	-
52	L-662	Steam Generator #1 Level Control Valve FLEX Backup Air Station	Operational	Operational	-
53	L-663	Steam Generator #3 Level Control Valve FLEX Backup Air Station	Operational	Operational	-
54	L-664	Steam Generator #4 Level Control Valve FLEX Backup Air Station	Operational	Operational	-
55	LT-3-43	Steam Generator #1 Wide Range Level Transmitter	Operational	Operational	-
56	1-LT-3-56	Steam Generator #2 Wide Range Level Transmitter	Operational	Operational	-
57	2-LT-3-56	Steam Generator #2 Wide Range Level Transmitter	Operational	Operational	Unit 2 SG3 Level Transmitter on Rack L-182 Unit 1 SG3 Level Transmitter on Rack L-706
58	LT-3-98	Steam Generator #3 Wide Range Level Transmitter	Operational	Operational	-
59	LT-3-111	Steam Generator #4 Wide Range Level Transmitter	Operational	Operational	-
60	L-183	Steam Generator Level Transmitter Rack	Operational	Operational	-

TABLE A-1: Expedited Seismic Equipment List (ESEL) for Sequoyah Nuclear Plant (Continued)

ESEL Item Number	Equipment		Operating State		Notes/Comments
	ID	Description	Normal State	Desired State	
61	L-706	Steam Generator Level Transmitter Rack (Unit 1)	Operational	Operational	-
62	L-704	Steam Generator Level Transmitter Rack	Operational	Operational	-
63	L-185	Steam Generator Level Transmitter Rack	Operational	Operational	-
64	L-182	Steam Generator Level Transmitter Rack (Unit 2)	Operational	Operational	-
65	LI-3-43	Steam Generator #1 Wide Range Level Indicator	Operational	Operational	-
66	LI-3-56	Steam Generator #2 Wide Range Level Indicator	Operational	Operational	-
67	LI-3-98	Steam Generator #3 Wide Range Level Indicator	Operational	Operational	-
68	LI-3-111	Steam Generator #4 Wide Range Level Indicator	Operational	Operational	-
69	PT-1-2A	Steam Generator #1 Discharge Pressure Transmitter	Operational	Operational	120V VIPB I Rack 3
70	PT-1-9A	Steam Generator #2 Discharge Pressure Transmitter	Operational	Operational	120V VIPB I Rack 3
71	PT-1-20A	Steam Generator #3 Discharge Pressure Transmitter	Operational	Operational	120V VIPB I Rack 4
72	PT-1-27A	Steam Generator #4 Discharge Pressure Transmitter	Operational	Operational	120V VIPB I Rack 4
73	L-194	Steam Generator Discharge Pressure Transmitter Rack	Operational	Operational	-

TABLE A-1: Expedited Seismic Equipment List (ESEL) for Sequoyah Nuclear Plant (Continued)

ESEL Item Number	Equipment		Operating State		Notes/Comments
	ID	Description	Normal State	Desired State	
74	L-196	Steam Generator Discharge Pressure Transmitter Rack	Operational	Operational	-
75	PI-1-2D	Steam Generator #1 Discharge Pressure Indicator	Operational	Operational	Channel A input
76	PI-1-9D	Steam Generator #2 Discharge Pressure Indicator	Operational	Operational	Channel A input
77	PI-1-20D	Steam Generator #3 Discharge Pressure Indication	Operational	Operational	Channel A input
78	PI-1-27D	Steam Generator #4 Discharge Pressure Indication	Operational	Operational	Channel A input
79	1-TNK-002-0229	Unit 1 Condensate Storage Tank	Operational	Operational	DCN 23191 will seismically qualify CST to 2x SSE HCLPF
80	2-TNK-002-0232	Unit 2 Condensate Storage Tank	Operational	Operational	DCN 23191 will seismically qualify CST to 2x SSE HCLPF
81	FCV-3-179A	ERCW Header B AFW Supply Valve	Closed	Open	Switchover to ERCW header 480V MOV Board 1(2)B2-B/11E or hand wheel
82	FCV-3-179B	ERCW Header B AFW Supply Valve	Closed	Open	Switchover to ERCW header 480V MOV Board 1(2)B2-B/11B or hand wheel
83	FCV-3-136A	ERCW Header A AFW Supply Valve	Closed	Open	Switchover to ERCW header 480V MOV Board 1(2)A2-A/2E or hand wheel
84	FCV-3-136B	ERCW Header A AFW Supply Valve	Closed	Open	Switchover to ERCW header 480V MOV Board 1(2)A2-A/2B or hand wheel
85	FCV-63-118	Cold Leg Accumulator #1 Isolation Valve	Open	Closed	-
86	FCV-63-98	Cold Leg Accumulator #2 Isolation Valve	Open	Closed	-

TABLE A-1: Expedited Seismic Equipment List (ESEL) for Sequoyah Nuclear Plant (Continued)

ESEL Item Number	Equipment		Operating State		Notes/Comments
	ID	Description	Normal State	Desired State	
87	FCV-63-80	Cold Leg Accumulator #3 Isolation Valve	Open	Closed	-
88	FCV-63-67	Cold Leg Accumulator #4 Isolation Valve	Open	Closed	-
89	1-PMP-63-10A	Safety Injection Pump	Standby	Operational	-
90	2-PMP-63-10A	Safety Injection Pump	Standby	Operational	-
91	HS-63-10A	Safety Injection Pump Hand Switch	Operational	Operational	-
92	M-6	Panel M-6	Operational	Operational	-
93	1-TNK-062-0239	Boric Acid Tank (BAT) A	Available	Available	-
94	2-TNK-062-0239	Boric Acid Tank (BAT) B	Available	Available	-
95	0-TNK-062-0243	Boric Acid Tank (BAT) C	Available	Available	-
96	HEX-074-0015	RHR Heat Exchanger 1A	Intact	Intact	RWST gravity feed path
97	TNK-063-044	RWST	Operational	Operational	-
98	PCV-068-0340A	RCS Pressurizer Power Relief Valve	Operational	Operational	125V DC Vital Battery Board I
99	HS-68-340 AA	Pressurizer PORV Hand Switch	Operational	Operational	-

TABLE A-1: Expedited Seismic Equipment List (ESEL) for Sequoyah Nuclear Plant (Continued)

ESEL Item Number	Equipment		Operating State		Notes/Comments
	ID	Description	Normal State	Desired State	
100	PDT-30-44	Containment Pressure Differential Transmitter	Operational	Operational	-
101	PDI-30-44	Containment Pressure Differential Indicator	Operational	Operational	-
102	1-FAN-030-0038	Unit 1 Containment Air Return Fan A	Standby	Operational	Powered by 480V SDB 1A1
103	2-FAN-030-0038	Unit 2 Containment Air Return Fan A	Standby	Operational	Powered by 480V SDB 2A1
104	HS-30-38A	Containment Air Return Fan A Hand Switch	Operational	Operational	-
105	M-9	Panel M-9	Operational	Operational	-
106	1-XFA-268-1A-A	PHMS Xfrm 1A	Operational	Operational	Power supply to hydrogen igniters
107	2-XFA-268-2A-A	PHMS Xfrm 2A	Operational	Operational	Power supply to hydrogen igniters
108	1-PNL-268-YA	120V AC PHMS Distribution Panel 1A	Operational	Operational	Power supply to hydrogen igniters
109	2-PNL-268-YC	120V AC PHMS Distribution Panel 2A	Operational	Operational	Power supply to hydrogen igniters
110	TNK-018-38	1A-A 7-Day Oil Supply Tank	Available	Available	Use to supply diesel-powered FLEX equipment
111	TNK-018-40	1B-B 7-Day Oil Supply Tank	Available	Available	Use to supply diesel-powered FLEX equipment
112	TNK-018-39	2A-A 7-Day Oil Supply Tank	Available	Available	Use to supply diesel-powered FLEX equipment

TABLE A-1: Expedited Seismic Equipment List (ESEL) for Sequoyah Nuclear Plant (Continued)

ESEL Item Number	Equipment		Operating State		Notes/Comments
	ID	Description	Normal State	Desired State	
113	TNK-018-41	2B-B 7-Day Oil Supply Tank	Available	Available	Use to supply diesel-powered FLEX equipment
114	1-BDB-201-DJ	480V Shutdown Board A1-A	Operational	Operational	-
115	1-BDB-201-DK	480V Shutdown Board A2-A	Operational	Operational	-
116	1-BDB-201-DL	480V Shutdown Board B1-B	Operational	Operational	-
117	1-BDB-201-DM	480V Shutdown Board B2-B	Operational	Operational	-
118	2-BDB-201-DN	480V Shutdown Board A1-A	Operational	Operational	-
119	2-BDB-201-DO	480V Shutdown Board A2-A	Operational	Operational	-
120	2-BDB-201-DP	480V Shutdown Board B1-B	Operational	Operational	-
121	2-BDB-201-DQ	480V Shutdown Board B2-B	Operational	Operational	-
122	1-BDC-201-GG	480V Reactor MOV Board 1A1-A	Operational	Operational	Power to safety injection accumulator isolation MOVs
123	1-BDC-201-GJ	480V Reactor MOV Board 1B1-B	Operational	Operational	Power to safety injection accumulator isolation MOVs
124	2-BDC-201-GL	480V Reactor MOV Board 2A1-A	Operational	Operational	Power to safety injection accumulator isolation MOVs
125	2-BDC-201-GN	480V Reactor MOV Board 2B1-B	Operational	Operational	Power to safety injection accumulator isolation MOVs

TABLE A-1: Expedited Seismic Equipment List (ESEL) for Sequoyah Nuclear Plant (Continued)

ESEL Item Number	Equipment		Operating State		Notes/Comments
	ID	Description	Normal State	Desired State	
126	1-BDC-201-JE	480V C&A Vent Board 1A1-A	Operational	Operational	Power to high pressure FLEX pump Repowered by 6.9kV FLEX diesel generator
127	1-BDC-201-JF	480V C&A Vent Board 1A2-A	Operational	Operational	Power to intermediate pressure FLEX pump Repowered by 6.9kV FLEX diesel generator
128	2-BDC-201-JJ	480V C&A Vent Board 2A1-A	Operational	Operational	Power to high pressure FLEX pump Repowered by 6.9kV FLEX diesel generator
129	2-BDC-201-JK	480V C&A Vent Board 2A2-A	Operational	Operational	Power to intermediate pressure FLEX Pump Repowered by 6.9kV FLEX diesel generator
130	BDG-250-KE	125V DC Vital Battery Board I	Operational	Operational	-
131	BDG-250-KF	125V DC Vital Battery Board II	Operational	Operational	-
132	BDG-250-KG	125V DC Vital Battery Board III	Operational	Operational	-
133	BDG-250-KH	125V DC Vital Battery Board IV	Operational	Operational	-
134	1-BDE-250-NC-D	120V AC Vital Instrument Power Board 1-I	Operational	Operational	-
135	1-BDE-250-NE-E	120V AC Vital Instrument Power Board 1-II	Operational	Operational	-
136	1-BDE-250-NG-F	120V AC Vital Instrument Power Board 1-III	Operational	Operational	-
137	1-BDE-250-NJ-G	120V AC Vital Instrument Power Board 1-IV	Operational	Operational	-
138	2-BDE-250-ND-D	120V AC Vital Instrument Power Board 2-I	Operational	Operational	-

TABLE A-1: Expedited Seismic Equipment List (ESEL) for Sequoyah Nuclear Plant (Continued)

ESEL Item Number	Equipment		Operating State		Notes/Comments
	ID	Description	Normal State	Desired State	
139	2-BDE-250-NF-E	120V AC Vital Instrument Power Board 2-II	Operational	Operational	-
140	2-BDE-250-NH-F	120V AC Vital Instrument Power Board 2-III	Operational	Operational	-
141	2-BDE-250-NK-G	120V AC Vital Instrument Power Board 2-IV	Operational	Operational	-
142	0-BATB-250-QV	125V DC Vital Battery I	Operational	Operational	-
143	0-BATB-250-QW	125V DC Vital Battery II	Operational	Operational	-
144	0-BATB-250-QX	125V DC Vital Battery III	Operational	Operational	-
145	0-BATB-250-QY	125V DC Vital Battery IV	Operational	Operational	-
146	0-CHGB-250-QE	125V DC Vital Battery Charger I	Operational	Operational	-
147	0-CHGB-250-QG	125V DC Vital Battery Charger II	Operational	Operational	-
148	0-CHGB-250-QH	125V DC Vital Battery Charger III	Operational	Operational	-
149	0-CHGB-250-QJ	125V DC Vital Battery Charger IV	Operational	Operational	-
150	1-INVB-250-QL	120V AC Vital Inverter 1-I	Operational	Operational	-
151	1-INVB-250-QN	120V AC Vital Inverter 1-II	Operational	Operational	-

TABLE A-1: Expedited Seismic Equipment List (ESEL) for Sequoyah Nuclear Plant (Continued)

ESEL Item Number	Equipment		Operating State		Notes/Comments
	ID	Description	Normal State	Desired State	
152	1-INVB-250-QR	120V AC Vital Inverter 1-III	Operational	Operational	-
153	1-INVB-250-QT	120V AC Vital Inverter 1-IV	Operational	Operational	-
154	2-INVB-250-QM	120V AC Vital Inverter 2-I	Operational	Operational	-
155	2-INVB-250-QP	120V AC Vital Inverter 2-II	Operational	Operational	-
156	2-INVB-250-QS	120V AC Vital Inverter 2-III	Operational	Operational	-
157	2-INVB-250-QU	120V AC Vital Inverter 2-IV	Operational	Operational	-
158	1-XE-92-5001	N31 Neutron Detector	Operational	Operational	Unit 1 NIS Channel 1
159	1-XM-92-5001A	N31 Neutron Source Range Amplifier	Operational	Operational	-
160	1-XM-92-5001B	N31 Neutron Source Range Optical Isolator	Operational	Operational	-
161	1-XI-92-5	N31 Signal Processor App R	Operational	Operational	-
162	1-XX-92-5001	N31 Source Range Indicator	Operational	Operational	-
163	1-XI-92-5001A	N31B Source Range Indicator	Operational	Operational	-
164	2-XE-92-5002	N32 Neutron Detector	Operational	Operational	Unit 2 NIS Channel 2

TABLE A-1: Expedited Seismic Equipment List (ESEL) for Sequoyah Nuclear Plant (Continued)

ESEL Item Number	Equipment		Operating State		Notes/Comments
	ID	Description	Normal State	Desired State	
165	2-XM-92-5002A	N32 Neutron Source Range Amplifier	Operational	Operational	-
166	2-XE-92-5002B	N32 Neutron Source Range Optical Isolator	Operational	Operational	-
167	2-XI-92-5	N32 Signal Processor App R	Operational	Operational	-
168	2-XX-92-5002	N32 Source Range Indicator	Operational	Operational	-
169	2-XI-92-5002B	N32B Source Range Indicator	Operational	Operational	-
170	L-10	Instrument Rack	Operational	Operational	-
171	M-4	Instrument Panel	Operational	Operational	-
172	M-13	Instrument Panel	Operational	Operational	-
173	PT-68-69	RCS Loop WR Pressure Transmitter Loop 1	Operational	Operational	-
174	PT-68-66	RCS Loop WR Pressure Transmitter Loop 3	Operational	Operational	-
175	PI-68-69	RCS Loop WR Pressure Indication Loop 1	Operational	Operational	-
176	PI-68-66A	RCS Loop WR Pressure Indication Loop 3	Operational	Operational	-
177	L-388	Instrument panel	Operational	Operational	-

TABLE A-1: Expedited Seismic Equipment List (ESEL) for Sequoyah Nuclear Plant (Continued)

ESEL Item Number	Equipment		Operating State		Notes/Comments
	ID	Description	Normal State	Desired State	
178	L-340	Instrument Panel	Operational	Operational	-
179	L-649	Instrument Panel	Operational	Operational	-
180	R-4	Instrument Rack	Operational	Operational	-
181	R-5	Instrument Rack	Operational	Operational	-
182	TE-68-18	Cold Leg WR Temperature Element Loop 1	Operational	Operational	-
183	TE-68-41	Cold Leg WR Temperature Element Loop 2	Operational	Operational	-
184	TE-68-60	Cold Leg WR Temperature Element Loop 3	Operational	Operational	-
185	TE-68-83	Cold Leg WR Temperature Element Loop 4	Operational	Operational	-
186	TI-68-18	Cold Leg WR Temperature Indication Loop 1	Operational	Operational	-
187	TI-68-41	Cold Leg WR Temperature Indication Loop 2	Operational	Operational	-
188	TI-68-60	Cold Leg WR Temperature Indication Loop 3	Operational	Operational	-
189	TI-68-83	Cold Leg WR Temperature Indication Loop 4	Operational	Operational	-
190	TE-68-1	Hot Leg WR Temperature Element Loop 1	Operational	Operational	-

TABLE A-1: Expedited Seismic Equipment List (ESEL) for Sequoyah Nuclear Plant (Continued)

ESEL Item Number	Equipment		Operating State		Notes/Comments
	ID	Description	Normal State	Desired State	
191	TE-68-24	Hot Leg WR Temperature Element Loop 2	Operational	Operational	-
192	TE-68-43	Hot Leg WR Temperature Element Loop 3	Operational	Operational	-
193	TE-68-65	Hot Leg WR Temperature Element Loop 4	Operational	Operational	-
194	TI-68-1	Hot Leg WR Temperature Indication Loop 1	Operational	Operational	-
195	TI-68-24	Hot Leg WR Temperature Indication Loop 2	Operational	Operational	-
196	TI-68-43	Hot Leg WR Temperature Indication Loop 3	Operational	Operational	-
197	TI-68-65	Hot Leg WR Temperature Indication Loop 4	Operational	Operational	-
198	M-5	Instrument Panel	Operational	Operational	-
199	R-2	Instrument Rack	Operational	Operational	-
200	R-6	Instrument Rack	Operational	Operational	-
201	LT-68-325C	RCS Pressurizer Level Transmitter	Operational	Operational	-
202	LI-68-325C	RCS Pressurizer Level Indication	Operational	Operational	-
203	L-179	Instrumentation Panel	Operational	Operational	-

TABLE A-1: Expedited Seismic Equipment List (ESEL) for Sequoyah Nuclear Plant (Continued)

ESEL Item Number	Equipment		Operating State		Notes/Comments
	ID	Description	Normal State	Desired State	
204	0-DG-360-000A	480V FLEX Diesel Generator	Standby	Operational	-
205	0-DG-360-003A	6.9kV FLEX Diesel Generator	Standby	Operational	-
206	1-PNLA-082-TU	Diesel Generator 1B-B Control Panel	Standby	Operational	-
207	1-PNLA-082-TT	Diesel Generator G 1A-A Control Panel	Standby	Operational	-
208	2-PNLA-082-TV	Diesel Generator 2A-A Control Panel	Standby	Operational	-
209	2-PNLA-082-TW	Diesel Generator 2B-B Control Panel	Standby	Operational	-
210	0-BD-360-0003A	FLEX Diesel Generator 3A Switchgear	Standby	Operational	-
211	0-BKR-360-0003A/1/A2	FLEX DG 3A Switchgear Breaker A2	Standby	Operational	DCN 23197
212	0-TANK-360-113	6900V 3MW FLEX DG Fuel Oil Storage Tank 3A	Standby	Operational	DCN 23197
213	0-SW-360-0003A/1	6900V 3MW FLEX Diesel GEN 3A Fused Disconnect Switch	Standby	Operational	DCN 23197
214	0-XFMR-360-3A/1	6900V-480V 3MW FLEX Diesel GEN 3A 20 KVA Dry Type Transformer	Standby	Operational	DCN 23197
215	0-XFMR-360-3A/2	480V-120/240V 3MW FLEX Diesel GEN 3A 5 KVA Dry Type Transformer	Standby	Operational	DCN 23197
216	0-DPL-360-0003A/1	480-Volt Distribution Panel with 100A Main Circuit Breaker	Standby	Operational	DCN 23197

TABLE A-1: Expedited Seismic Equipment List (ESEL) for Sequoyah Nuclear Plant (Continued)

ESEL Item Number	Equipment		Operating State		Notes/Comments
	ID	Description	Normal State	Desired State	
217	0-DPL-360-0003A/2	120/240 VAC Panelboard	Standby	Operational	DCN 23197
218	0-FU1-360-0103A	Primary Cntrl Fuse for Fuel Oil Pump A Starter	Standby	Operational	DCN 23197
219	0-FU1-360-0103B	Secondary Cntrl Fuse for Fuel Oil Pump A Starter	Standby	Operational	DCN 23197
220	0-FU1-360-0103C	Primary Cntrl Fuse for Fuel Oil Pump A Starter	Standby	Operational	DCN 23197
221	0-HS-360-103C	Fuel Oil Transfer Pmp A Emer Stop SW	Standby	Operational	DCN 23197
222	0-PMP-360-103	Fuel Oil System Transfer Pump 3A	Standby	Operational	DCN 23197
223	0-RES-360-003A	3MW FLEX Diesel Generator A Neutral Grounding Resistor	Standby	Operational	DCN 23197
224	0-STR-360-0103	3MW FLEX Diesel Generator Fuel Oil Transfer Pump Starter	Standby	Operational	DCN 23197
225	0-LS-360-0103	3MW FLEX Diesel Generator Fuel Oil Float Switch (Fill Pump Control)	Standby	Operational	DCN 23197
226	0-XSW-082-0001A	Transfer Switch 1A	Standby	Operational	DCN 23197
227	0-XSW-082-0002A	Transfer Switch 2A	Standby	Operational	DCN 23197
228	0-XSW-082-0003B	Transfer Switch 2B	Standby	Operational	DCN 23197
229	0-XSW-082-0004B	Transfer Switch 1B	Standby	Operational	DCN 23197

TABLE A-1: Expedited Seismic Equipment List (ESEL) for Sequoyah Nuclear Plant (Continued)

ESEL Item Number	Equipment		Operating State		Notes/Comments
	ID	Description	Normal State	Desired State	
230	1-BDA-202-CM-A	6.9kV Shutdown Board 1A-A	Operational	Operational	-
231	2-BDA-202-CO-A	6.9kV Shutdown Board 2A-A	Operational	Operational	-
232	1-BDA-202-CN	6.9kV Shutdown Board 1B-B	Operational	Operational	-
233	2-BDA-202-CP	6.9kV Shutdown Board 2B-B	Operational	Operational	-
234	1-OXF-202-DL	480V Shutdown Transformer 1B1	Operational	Operational	Listed as 1-XFA-202-0317 in IPEEE
235	1-OXF-202-DM	480V Shutdown Transformer 1B2	Operational	Operational	Listed as 1-XFA-202-0319 in IPEEE
236	1-OXF-202-DJ	480V Shutdown Transformer 1A1	Operational	Operational	Listed as 1-XFA-202-0313 in IPEEE
237	1-OXF-202-DK	480V Shutdown Transformer 1A2	Operational	Operational	Listed as 1-XFA-202-0315 in IPEEE
238	2-OXF-202-DN	480V Shutdown Transformer 2A1	Operational	Operational	Listed as 2-XFA-202-0315 in IPEEE
239	2-OXF-202-DO	480V Shutdown Transformer 2A2	Operational	Operational	Listed as 2-XFA-202-0313 in IPEEE
240	2-OXF-202-DP	480V Shutdown Transformer 2B1	Operational	Operational	Listed as 2-XFA-202-0319 in IPEEE
241	2-OXF-202-DQ	480V Shutdown Transformer 2B2	Operational	Operational	Listed as 2-XFA-202-0317 in IPEEE
242	1-PMP-360-IP01	U1 Intermediate Pressure FLEX Pump	Standby	Operational	Powered by 480V C&A Vent BD 1A2, DCN 23193

TABLE A-1: Expedited Seismic Equipment List (ESEL) for Sequoyah Nuclear Plant (Continued)

ESEL Item Number	Equipment		Operating State		Notes/Comments
	ID	Description	Normal State	Desired State	
243	2-PMP-360-IP01	U2 Intermediate Pressure FLEX Pump	Standby	Operational	Powered by 480V C&A Vent BD 2A2, DCN 23193
244	1-PMP-360-HPCS	U1 High Pressure FLEX Pump	Standby	Operational	Powered by 480V C&A Vent BD 1A1, DCN 23193
245	2-PMP-360-HPCS	U2 High Pressure FLEX Pump	Standby	Operational	Powered by 480V C&A Vent BD 2A1, DCN 23193
246	SQN-1-IGN-268-0142-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	-
247	SQN-1-IGN-268-0130-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	-
248	SQN-1-IGN-268-0125-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	-
249	SQN-1-IGN-268-0123-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	-
250	SQN-1-IGN-268-0116-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	-
251	SQN-1-IGN-268-0128-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	-
252	SQN-1-IGN-268-0129-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	-
253	SQN-1-IGN-268-0114-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	-
254	SQN-1-IGN-268-0133-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	-
255	SQN-1-IGN-268-0102-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	-

TABLE A-1: Expedited Seismic Equipment List (ESEL) for Sequoyah Nuclear Plant (Continued)

ESEL Item Number	Equipment		Operating State		Notes/Comments
	ID	Description	Normal State	Desired State	
256	SQN-1-IGN-268-0115-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	-
257	SQN-1-IGN-268-0132-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	-
258	SQN-1-IGN-268-0108-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	-
259	SQN-1-IGN-268-0127-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	-
269	SQN-1-IGN-268-0155-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	-
270	SQN-1-IGN-268-0136-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	-
271	SQN-1-IGN-268-0131-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	-
272	SQN-1-IGN-268-0121-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	-
273	SQN-1-IGN-268-0122-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	-
274	SQN-1-IGN-268-0135-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	-
275	SQN-1-IGN-268-0159-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	-
276	SQN-1-IGN-268-0126-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	-
277	SQN-1-IGN-268-0107-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	-

TABLE A-1: Expedited Seismic Equipment List (ESEL) for Sequoyah Nuclear Plant (Continued)

ESEL Item Number	Equipment		Operating State		Notes/Comments
	ID	Description	Normal State	Desired State	
278	SQN-2-IGN-268-0226-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
279	SQN-2-IGN-268-0235-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
280	SQN-2-IGN-268-0202-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
281	SQN-2-IGN-268-0223-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
282	SQN-2-IGN-268-0222-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
283	SQN-2-IGN-268-0231-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
284	SQN-2-IGN-268-0206-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
285	SQN-2-IGN-268-0205-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
286	SQN-2-IGN-268-0234-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
287	SQN-2-IGN-268-0214-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
288	SQN-2-IGN-268-0208-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
289	SQN-2-IGN-268-0259-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
290	SQN-2-IGN-268-0250-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-

TABLE A-1: Expedited Seismic Equipment List (ESEL) for Sequoyah Nuclear Plant (Continued)

ESEL Item Number	Equipment		Operating State		Notes/Comments
	ID	Description	Normal State	Desired State	
291	SQN-2-IGN-268-0232-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
292	SQN-2-IGN-268-0201-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
293	SQN-2-IGN-268-0230-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
294	SQN-2-IGN-268-0249-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
295	SQN-2-IGN-268-0254-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
296	SQN-2-IGN-268-0224-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
297	SQN-2-IGN-268-0233-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
298	SQN-2-IGN-268-0255-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
299	SQN-2-IGN-268-0207-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
300	SQN-2-IGN-268-0236-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
301	SQN-2-IGN-268-0221-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
302	SQN-2-IGN-268-0213-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
303	SQN-2-IGN-268-0228-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-

TABLE A-1: Expedited Seismic Equipment List (ESEL) for Sequoyah Nuclear Plant (Continued)

ESEL Item Number	Equipment		Operating State		Notes/Comments
	ID	Description	Normal State	Desired State	
304	SQN-2-IGN-268-0216-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
305	SQN-2-IGN-268-0229-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
306	SQN-2-IGN-268-0242-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
307	SQN-2-IGN-268-0227-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
308	SQN-2-IGN-268-0215-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
309	SQN-2-IGN-268-0225-A	Unit 2 Train A Hydrogen Igniter	Standby	Operational	-
310	SQN-1-IGN-268-0105-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	
311	SQN-1-IGN-268-0154-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	
312	SQN-1-IGN-268-0106-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	
313	SQN-1-IGN-268-0113-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	
314	SQN-1-IGN-268-0149-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	
315	SQN-1-IGN-268-0101-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	
316	SQN-1-IGN-268-0150-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	

TABLE A-1: Expedited Seismic Equipment List (ESEL) for Sequoyah Nuclear Plant (Continued)

ESEL Item Number	Equipment		Operating State		Notes/Comments
	ID	Description	Normal State	Desired State	
317	SQN-1-IGN-268-0124-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	
318	SQN-1-IGN-268-0134-A	Unit 1 Train A Hydrogen Igniter	Standby	Operational	

ATTACHMENT B – ESEP HCLPF VALUES AND FAILURE MODES TABULATION FOR SEQUOYAH NUCLEAR PLANT

TABLE B-1: ESEP HCLPF Values and Failure Modes for Sequoyah Nuclear Plant						
Equipment ID	**Equipment Description**	**Equipment Class**	**Building**	**Floor Elevation**	**Failure Mode**	**HCLPF Capacity (g)**
1-PCV-1-5 2-PCV-1-5	Steam Generator #1 ARV Hand-wheel	7	ACB	734	Screen	0.50
1-PCV-1-30 2-PCV-1-30	Steam Generator #4 ARV Hand-wheel	7	ACB	734	Screen	0.50
1-L-501 2-L-501	PCV-1-12 Local Control Station	18	ACB	714	Functional	0.62
1-L-502 2-L-502	PCV-1-23 Local Control Station	18	ACB	714	Functional	0.62
1-PMP-3-142 2-PMP-3-142	TDAFW Pump	5	ACB	669	Functional	1.01
1-FCV-1-51 2-FCV-1-51	TDAFW Pump Trip and Throttle Valve	8A	ACB	669	Screen	0.50
1-FCV-1-52 2-FCV-1-52	TDAFW Pump Governor Valve	7	ACB	669	Functional	1.01
1-XS-46-57 2-XS-46-57	AFWT A-S Backup Control Transfer Service Water	14	ACB	669	Functional	0.88
1-HS-1-51B 2-HS-1-51B	TDAFW Pump Trip and Throttle Valve HS	20	ACB	669	Functional	0.88
1-L-381 2-L-381	TDAFW Pump Control Panel	20	ACB	669	Functional	0.85
1-L-215 2-L-215	AFW Flow Monitoring Panel	18	ACB	669	Functional	0.62
1-L-341 2-L-341	AFW Flow to Steam Generator #3 FT Rack	18	ACB	714	Functional	0.62
1-L-217 2-L-217	AFW Flow to Steam Generator #2 FT Rack	18	ACB	714	Functional	0.62
1-L-216 2-L-216	AFW Flow to Steam Generator #1 FT Rack	18	ACB	690	Functional	0.62
1-L-703 2-L-703	AFW Flow to Steam Generator #4 Flow Indication Rack	18	ACB	690	Functional	0.62
1-L-11A 2-L-11A	Steam Generator Level Control Panel	20	ACB	734	Functional	0.75
1-L-11B 2-L-11B	Steam Generator Level Control Panel	20	ACB	734	Functional	0.75

TABLE B-1: ESEP HCLPF Values and Failure Modes for Sequoyah Nuclear Plant (Continued)						
Equipment ID	**Equipment Description**	**Equipment Class**	**Building**	**Floor Elevation**	**Failure Mode**	**HCLPF Capacity (g)**
1-L-183 2-L-183	Steam Generator Level Transmitter Rack	18	RB	697	Functional	0.62
1-L-706 2-L-182	Steam Generator Level Transmitter Rack	18	RB	697	Functional	0.62
1-L-704 2-L-704	Steam Generator Level Transmitter Rack	18	RB	697	Functional	0.62
1-L-185 2-L-185	Steam Generator Level Transmitter Rack	18	RB	697	Functional	0.62
1-L-194 2-L-194	Steam Generator Discharge Pressure Transmitter Rack	18	ACB	690	Functional	0.62
1-L-196 2-L-196	Steam Generator Discharge Pressure Transmitter Rack	18	ACB	690	Functional	0.62
1-TNK-002-0229	Unit 1 Condensate Storage Tank	21	YARD	705	DCN 23191	>2x SSE
2-TNK-002-0232	Unit 2 Condensate Storage Tank	21	YARD	705	DCN 23191	>2x SSE
1-FCV-63-118 2-FCV-63-118	Cold Leg Accumulator #1 Isolation Valve	8a	RB	693	Screen	0.50
1-FCV-63-98 2-FCV-63-98	Cold Leg Accumulator #2 Isolation Valve	8a	RB	693	Screen	0.50
1-FCV-63-80 2-FCV-63-80	Cold Leg Accumulator #3 Isolation Valve	8a	RB	693	Screen	0.50
1-FCV-63-67 2-FCV-63-67	Cold Leg Accumulator #4 Isolation Valve	8a	RB	693	Screen	0.50
1-M-6 2-M-6	MCR Benchboard M-6	20	ACB	732	Main Control Room Ceiling	0.425
1-TNK-062-0239	Boric Acid Tank (BAT) A	21	ACB	690	Overturning Moment	0.78
2-TNK-062-0239	Boric Acid Tank (BAT) B	21	ACB	690	Overturning Moment	0.78
0-TNK-062-0243	Boric Acid Tank (BAT) C	21	ACB	690	Overturning Moment	0.78
1-M-9 2-M-9	MCR Vertical Panel M-9	20	ACB	732	Main Control Room Ceiling	0.425

TABLE B-1: ESEP HCLPF Values and Failure Modes for Sequoyah Nuclear Plant (Continued)						
Equipment ID	**Equipment Description**	**Equipment Class**	**Building**	**Floor Elevation**	**Failure Mode**	**HCLPF Capacity (g)**
SQN-1-IGN-268-(MANY)	Hydrogen Igniters	0	RB	SCV	Screen	0.50
1-XFA-268-1A-A	PHMS Xfrm 1A	4	ACB	759	Functional	0.36
2-XFA-268-2A-A	PHMS Xfrm 2A	4	ACB	759	Functional	0.36
1-PNL-268-YA	120V AC PHMS Distribution Panel 1A	14	ACB	759	Functional	0.36
2-PNL-268-YC	120V AC PHMS Distribution Panel 2A	14	ACB	759	Functional	0.36
TNK-018-38	1A-A 7-Day Oil Supply Tank	21	DGB	719	Screen	0.50
TNK-018-40	1B-B 7-Day Oil Supply Tank	21	DGB	719	Screen	0.50
TNK-018-39	2A-A 7-Day Oil Supply Tank	21	DGB	719	Screen	0.50
TNK-018-41	2B-B 7-Day Oil Supply Tank	21	DGB	719	Screen	0.50
1-XE-92-5001	N31 Neutron Detector	18	RB	697	Screen	0.50
1-XM-92-5001A	N31 Neutron Source Range Amplifier	20	ACB	734	Screen	0.50
1-XM-92-5001B	N31 Neutron Source Range Optical Isolation	20	ACB	734	Screen	0.50
2-XE-92-5002	N32 Neutron Detector	18	RB	697	Screen	0.50
2-XM-92-5002A	N32 Neutron Source Range Amplifier	20	ACB	714	Screen	0.50

TABLE B-1: ESEP HCLPF Values and Failure Modes for Sequoyah Nuclear Plant (Continued)						
Equipment ID	**Equipment Description**	**Equipment Class**	**Building**	**Floor Elevation**	**Failure Mode**	**HCLPF Capacity (g)**
2-XM-92-5002B	N32 Neutron Source Range Optical Isolation	20	ACB	714	Screen	0.50
1-L-10 2-L-10	Remote Control Panel L-10	20	ACB	734	Functional	0.425
1-M-4 2-M-4	MCR Benchboard M-4	20	ACB	732	Main Control Room Ceiling	0.425
1-M-13 2-M-13	MCR Vertical Panel M-13	20	ACB	732	Main Control Room Ceiling	0.425
1-L-388 2-L-388	RCS Loop WR PT Loop 1 Instrument Rack	28	ACB	690	Functional	0.62
1-L-340 2-L-340	RCS Loop WR PT Loop 3 Instrument Rack	28	ACB	690	Functional	0.62
1-R-4 2-R-4	AIR Panel R-4	20	ACB	685	Functional	0.64
1-R-5 2-R-5	AIR Panel R-5	20	ACB	685	Functional	0.64
1-TE-68-18 2-TE-68-18	Cold Leg WR Temperature Element Loop 1	19	RB	693	Screen	0.50
1-TE-68-41 2-TE-68-41	Cold Leg WR Temperature Element Loop 2	19	RB	693	Screen	0.50
1-TE-68-60 2-TE-68-60	Cold Leg WR Temperature Element Loop 3	19	RB	693	Screen	0.50
1-TE-68-83 2-TE-68-83	Cold Leg WR Temperature Element Loop 4	19	RB	693	Screen	0.50
1-TE-68-1 2-TE-68-1	Hot Leg WR Temperature Element Loop 1	19	RB	679	Screen	0.50
1-TE-68-24 2-TE-68-24	Hot Leg WR Temperature Element Loop 2	19	RB	679	Screen	0.50
1-TE-68-43 2-TE-68-43	Hot Leg WR Temperature Element Loop3	19	RB	679	Screen	0.50
1-TE-68-65 2-TE-68-65	Hot Leg WR Temperature Element Loop 4	19	RB	679	Screen	0.50
1-M-5 2-M-5	MCR Benchboard M-5	20	ACB	732	Main Control Room Ceiling	0.425

TABLE B-1: ESEP HCLPF Values and Failure Modes for Sequoyah Nuclear Plant (Continued)						
Equipment ID	**Equipment Description**	**Equipment Class**	**Building**	**Floor Elevation**	**Failure Mode**	**HCLPF Capacity (g)**
1-R-2 2-R-2	AIR Panel R-2	20	ACB	685	Functional	0.64
1-R-6 2-R-6	AIR Panel R-6	20	ACB	685	Functional	0.64
1-L-179 2-L-179	RCS Pressurizer Level Transmitter Instrument Rack	18	RB		Functional	0.62
2-XM-92-5002B	N32 Neutron Source Range Optical Iso.	20	ACB	714	Screen	0.50
1-M-2	Cabinet M-2	20	ACB	732	Main Control Room Ceiling	0.425
1-FCV-3-136A 2-FCV-3-136A	ERCW Header A AFW Supply Valve	8a	ACB	669	Screen	0.50
1-FCV-3-136B 2-FCV-3-136B	ERCW Header A AFW Supply Valve	8a	ACB	669	Screen	0.50
1-FCV-3-179A 2-FCV-3-179A	ERCW Header B AFW Supply Valve	8a	ACB	669	Screen	0.50
1-FCV-3-179B 2-FCV-3-179B	ERCW Header B AFW Supply Valve	8a	ACB	669	Screen	0.50

ENCLOSURE 2

LIST OF COMMITMENTS

1. Perform seismic walkdowns, generate HCLPF calculations, and design and implement any necessary modification for the following Unit 1 inaccessible items no later than the end of the second planned Unit 1 refueling outage after December 31, 2014:

   a. FCV 63-118 - Cold Leg Accumulator Isolation Valve #1
   b. FCV 63-067 - Cold Leg Accumulator Isolation Valve #4
   c. FCV 63-080 - Cold Leg Accumulator Isolation Valve #3
   d. FCV 63-098 - Cold Leg Accumulator Isolation Valve #2
   e. Instrument Rack 1 - L - 182 located in Fan Room 2
   f. Instrument Rack 1 - L - 183 located in Fan Room 1
   g. Instrument Rack 1 - L - 179
   h. Instrument Rack 1 - L - 185
   i. Instrument Rack 1 - L - 704
   j. Instrument Rack 1 - L - 706
   k. Instrument Rack 1 - L - 194
   l. Instrument Rack 1 - L - 196
   m. Instrument Rack 1 - L - 216

2. Modify TDAFP Control Panel 2-L-381 anchorage to replace the corroded anchor such that HCLPH > RLGM no later than the end of the U2 Cycle 20 refueling outage.

3. Submit a letter to NRC summarizing the HCLPF results of Commitment 1 and confirming implementation of the plant modifications associated with Commitment 2 within 60 days following completion of ESEP activities.