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

CNL-19-061

October 18, 2019

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
Renewed Facility Operating License Nos. DPR-77 and DPR-79
NRC Docket Nos. 50-327 and 50-328

Subject: **Seismic Probabilistic Risk Assessment for Sequoyah Nuclear Plant, Units 1 and 2 - Response to NRC Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident**

References:

1. NRC Letter, “Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident,” dated March 12, 2012 (ML12053A340)
2. EPRI Report 1025287, “Seismic Evaluation Guidance, Screening, Prioritization and Implementation Details [SPID] for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic,” dated February 2013 (ML12333A170)
3. TVA letter to NRC, “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,” dated March 31, 2014 (ML14098A478)
4. NRC letter to TVA, “Sequoyah Nuclear Plant, Units 1 and 2 - Staff Assessment of Information Provided Pursuant to Title 10 of the Code of Federal Regulations Part 50, Section 50.54(f), Seismic Hazard Reevaluations Relating to Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident (TAC Nos. MF3767 and MF3768),” dated April 27, 2015 (ML15098A641)
5. NRC Letter, “Final Determination of Licensee Seismic Probabilistic Risk Assessments Under the Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendation 2.1 ‘Seismic’ of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident,” dated October 27, 2015 (ML15194A015)

On March 12, 2012, the Nuclear Regulatory Commission (NRC) issued a Request for Information pursuant to Title 10 of the *Code of Federal Regulations* (CFR) Part 50.54(f) (Reference 1) to all power reactor licensees. Enclosure 1 of the 50.54(f) letter requested addressees to reevaluate the seismic hazards at their respective sites using present-day NRC requirements and guidance, and to identify any actions taken or planned to address plant-specific vulnerabilities associated with the updated seismic hazards.

EPRI Report 1025287 (Reference 2) provides the guidance for screening, prioritization, and implementation details for the resolution of the Fukushima Near-Term Task Force (NTTF) Recommendation 2.1: Seismic. The EPRI Screening, Prioritization and Implementation Details (SPID) guidance was used to compare the reevaluated seismic hazard to the design basis seismic hazard for Sequoyah Nuclear Plant, Units 1 and 2. As described in Reference 3, Enclosure 3, it was concluded that the reevaluated ground motion response spectrum (GMRS) exceeded the design basis response spectrum in the 1 to 10 Hz range. Accordingly, a seismic probabilistic risk assessment was required.
Reference 4 is the NRC Staff Assessment for Sequoyah Nuclear Plant, Units 1 and 2, seismic hazard submittals which concluded that the reevaluated seismic hazards described in Reference 3, Enclosure 4, are suitable for other activities associated with NTTF Recommendation 2.1: Seismic.

In Reference 5, NRC indicated that a seismic probabilistic risk assessment was required for Sequoyah Nuclear Plant, Units 1 and 2, and should be submitted to NRC by December 31, 2019.

The Enclosure to this letter provides the Seismic Probabilistic Risk Assessment Summary Report for Sequoyah Nuclear Plant, Units 1 and 2, as requested in Reference 5. The Enclosure provides the information requested in Item (8)B of the 50.54(f) letter associated with NTTF Recommendation 2.1: Seismic.

This letter contains no new regulatory commitments.

If you have any questions regarding this submittal, please contact Russell Thompson at (423) 751-2567.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 18th day of October 2019.

Respectfully,

James T. Polickoski
Director, Nuclear Regulatory Affairs

Enclosure

cc: See Page 3

Enclosure:

Sequoyah Nuclear Plant, Units 1 and 2, Seismic Probabilistic Risk Assessment in Response to 50.54(f) Letter with Regard to NTTF 2.1 Seismic Summary Report

cc (Enclosure):

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

ENCLOSURE

Sequoyah Nuclear Plant, Units 1 and 2
Seismic Probabilistic Risk Assessment
in Response to
50.54(f) Letter with Regard to NTTF 2.1 Seismic Summary Report

Sequoyah Nuclear Plant (SQN) Units 1 and 2 Seismic Probabilistic Risk Assessment in Response to 50.54(f) Letter with Regard to NTTF 2.1 Seismic

SUMMARY REPORT

September 2019

Table of Contents

Executive Summary

In response to the 10 CFR 50.54(f) letter issued by the Nuclear Regulatory Commission (NRC) on March 12, 2012, a Seismic Probabilistic Risk Assessment (PRA) has been developed for Sequoyah Nuclear Plant (SQN) Units 1 and 2. The Seismic PRA shows that the point estimate seismic Core Damage Frequency (CDF) is 4.1E-06 per reactor year (ry) for Unit 1 and is 4.9E-06 per ry for Unit 2. The seismic Large Early Release Frequency (LERF) is 2.6E-06 per ry for Unit 1 and is 2.4E-06 per ry for Unit 2. Note that CDF and LERF throughout this document are always referring to seismic CDF and seismic LERF, not CDF and LERF from all hazards.

Sensitivity studies were performed to identify critical assumptions, test the sensitivity to quantification parameters and the seismic hazard, and identify potential areas to consider for the reduction of seismic risk. These sensitivity studies demonstrated that the model results were robust to the modeling and assumptions used. No seismic hazard vulnerabilities were identified, and no plant actions have been taken or are planned given the insights from the seismic risk assessment.

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 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 [2], 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.

A comparison between the reevaluated seismic hazard and the design basis for Sequoyah Nuclear Plant (SQN) Units 1 and 2 has been performed, in accordance with the guidance in Electric Power Research Institute (EPRI) 1025287, “Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic” [3], and previously submitted to the NRC [4]. That comparison concluded that the Ground Motion Response Spectra (GMRS), which was developed based on the reevaluated seismic hazard, exceeds the design basis seismic response spectrum in the 1 to 10 Hz range, and a seismic risk assessment is required. A seismic PRA has been developed to perform the seismic risk assessment for SQN in response to the 50.54(f) letter, specifically item (8) in Enclosure 1 of the 50.54(f) letter.

This report describes the seismic PRA developed for SQN and provides the information requested in item (8)(B) of Enclosure 1 of the 50.54(f) letter and in Section 6.8 of the SPID. The SPRA model has been peer reviewed (as described in Appendix A) and found to be of appropriate scope and technical capability for use in assessing the seismic risk for SQN, identifying which structures, systems, and components (SSCs) are important to seismic risk, and describing plant-specific seismic issues and associated actions planned or taken in response to the 50.54(f) letter.

This report provides summary information regarding the seismic PRA as outlined in Section 2.

The level of detail provided in the report is intended to enable the NRC to understand the inputs and methods used, the evaluations performed, and the decisions made as a result of the insights gained from the SQN seismic PRA.

2.0 Information Provided in This Report

The following information is requested in the 50.54(f) letter [2], Enclosure 1, "Requested Information" Section, paragraph (8)B, for plants performing a seismic PRA.

(1) The list of the significant contributors to SCDF for each seismic acceleration bin, including importance measures (e.g., Fussell-Vesely)

(2) A summary of the methodologies used to estimate the SCDF and LERF, including the following:

  i. Methodologies used to quantify the seismic fragilities of SSCs, together with key assumptions

  ii. SSC fragility values with reference to the method of seismic qualification, the dominant failure mode(s), and the source of information

  iii. Seismic fragility parameters

  iv. Important findings from plant walkdowns and any corrective actions taken

  v. Process used in the seismic plant response analysis and quantification, including the specific adaptations made in the internal events PRA model to produce the seismic PRA model and their motivation

  vi. Assumptions about containment performance

(3) Description of the process used to ensure that the SPRA is technically adequate, including the dates and findings of any peer reviews

(4) Identified plant-specific vulnerabilities and actions that are planned or taken

Note that 50.54(f) letter Enclosure 1 paragraphs 1 through 6, regarding the seismic hazard evaluation reporting, also apply, but have been satisfied through the previously submitted SQN Seismic Hazard Submittal [4]. Further, 50.54(f) letter Enclosure 1 paragraph 9 requesting information on the Spent Fuel Pool has been satisfied [5,6].

Table 2.0-1 provides a cross-reference between the 50.54(f) reporting items noted above and the location in this report where the corresponding information is discussed.

The SPID [3] defines the principal parts of an SPRA, and the SQN SPRA has been developed and documented in accordance with the SPID. The main elements of the SPRA performed for SQN in response to the 50.54(f) Seismic letter correspond to those described in Section 6.1.1 of the SPID, i.e.:

- Seismic hazard analysis
- Seismic structure response and SSC fragility analysis
- Systems/accident sequence (seismic plant response) analysis
- Risk quantification

Table 2.0-2 provides a cross-reference between the reporting items noted in Section 6.8 of the SPID, other than those already listed in Table 2.0-1, and provides the location in this report where the corresponding information is discussed.

The SQN SPRA and associated documentation has been peer reviewed [7] against the ASME/ANS PRA Standard [8] in accordance with the process defined in Nuclear Energy Institute (NEI) 12-13 [9] as documented in the SQN SPRA Peer Review Report. The SQN SPRA, complete SPRA documentation, and details of the peer review are available for NRC review.

Subsequent to the peer review, an independent assessment was performed of the closure of Finding-Level Facts and Observations (F&O) of record from the peer review [20]. The assessment was performed via NEI 12-13 Appendix X guidance, which has been accepted by the NRC [10]. The details of the Finding-Level F&O independent assessment are available for NRC review.

This submittal provides a summary of the SPRA development, results and insights, the peer review process and results, and the independent assessment, sufficient to meet the 50.54(f) information request in a manner intended to enable NRC to understand and determine the validity of key input data and calculation models used, and to assess the sensitivity of the results to key aspects of the analysis.

The content of this report is organized as follows:

- Section 3 provides information related to the SQN seismic hazard analysis.
- Section 4 provides information related to the determination of seismic fragilities for SQN SSCs included in the seismic plant response.
- Section 5 provides information regarding the plant seismic response model (seismic accident sequence model) and the quantification of results.
- Section 6 summarizes the results and conclusions of the SPRA, including identified plant seismic issues and actions taken or planned.
- Section 7 provides references.
- Section 8 provides a list of acronyms used.
- Appendix A provides an assessment of SPRA Technical Adequacy for Response to NTTF 2.1 Seismic 50.54(f) letter, including a summary of SQN SPRA peer review and independent assessment as well as a discussion of the F&Os related to the SQN Internal Events PRA (IEPRA), which have all been closed.
- Appendix B provides a response for each of the generic observations associated with the staff's review of seismic probabilistic risk assessment (SPRA) reports provided in response to the March 12, 2012, 50.54(f) letter associated with reevaluated seismic hazards.
-

Table 2.0-1 Cross-Reference for 50.54(f) Enclosure 1 SPRA Reporting

50.54(f) Letter Reporting Item	Description	Location in this Report
1	List of the significant contributors to SCDF for each seismic acceleration bin, including importance measures	The significant contributors are provided in Section 5.
2	Summary of the methodologies used to estimate the SCDF and SLERF	A summary of the methodologies utilized to estimate SCDF and SLERF are provided in Sections 3, 4, and 5.
2i	Methodologies used to quantify the seismic fragilities of SSCs, together with key assumptions	Seismic methodologies are provided in Section 4.
2ii	SSC fragility values with reference to the method of seismic qualification, the dominant failure mode(s), and the source of information	Tables 5.4-3, 5.4-4, 5.5-3, and 5.5-4 provides fragilities (A_m, median acceleration capacity, and beta, uncertainty in capacity), failure mode information, and method of determining fragilities for the top risk-significant SSCs based on Fussell-Vesely (F-V).
2iii	Seismic fragility parameters	Tables 5.4-3, 5.4-4, 5.5-3, and 5.5-4 provide fragilities (A_m and beta), failure mode information, and method of determining fragilities for the top risk-significant SSCs based on F-V.
2iv	Important findings from plant walkdowns and any corrective actions taken	Section 4.2 addresses walkdowns and walkdown insights.
2v	Process used in the seismic plant response analysis and quantification, including specific adaptations made in the IEPRA model to produce the SPRA model and their motivation	Section 5 provides the processes used in the seismic plant response.
2vi	Assumptions about containment performance	Sections 4.3, 5.1.5, and 5.5 address containment and related SSC performance.
3	Description of the process used to ensure that the SPRA is technically adequate, including the dates and findings of any peer reviews	Appendix A describes the assessment of SPRA technical adequacy for the 50.54(f) submittal and results of the SPRA peer review and subsequent independent assessment.
4	Identified plant-specific vulnerabilities and actions that are planned or taken	Section 6 addresses the plant-specific vulnerabilities. No vulnerabilities were identified and no actions are planned as a result of the SPRA.

Table 2.0-2 Cross-Reference for Additional SPID Section 6.8 SPRA Reporting

SPID Section 6.8 Item [3] Description	Location in this Report
A report should be submitted to the NRC summarizing the SPRA inputs, methods, and results.	Entirety of the report addresses this.
The level of detail needed in the submittal should be sufficient to enable NRC to understand and determine the validity of all input data and calculation models used.	Entirety of the report addresses this. The key methods of analysis and referenced codes and standards are identified in the report.
The level of detail needed in the submittal should be sufficient to assess the sensitivity of the results to all key aspects of the analysis.	Entirety of the report addresses this. Results sensitivities are discussed in Section 5.7 (SPRA Quantification Sensitivity Analysis).
The level of detail needed in the submittal should be sufficient to make necessary regulatory decisions as a part of NTTF Phase 2 activities.	Entirety of the report addresses this.
It is not necessary to submit all the SPRA documentation for such an NRC review. Relevant documentation should be cited in the submittal and be available for NRC review in easily retrievable form.	Entirety of report addresses this. This report summarizes important information from the SPRA, with detailed information in lower-tier documentation.
Documentation criteria for a SPRA are identified throughout the ASME/ANS (American Society of Mechanical Engineers/American Nuclear Society) Standard [8]. Utilities are expected to retain that documentation consistent with the Standard.	This is an expectation relative to documentation of the SPRA that the utility retains to support application of the SPRA to risk-informed plant decision-making.

Note (1): The items listed here do not include those designated in SPID Section 6.8 as "guidance."

3.0 SQN Seismic Hazard and Plant Response

This section provides summary site information and pertinent features including location and site characterization. The subsections provide brief summaries of the site hazard and plant response characterization.

SQN is a dual-unit Westinghouse 4-loop pressurized water reactor (PWR) located approximately 7.5 miles northeast of the nearest city limit of Chattanooga, Tennessee, on a peninsula on the western shore of Chickamauga Lake at Tennessee River mile marker 484.5. The regional and site (local) geology is described in additional detail in the SQN NTTF 2.1 Seismic Hazard submittal [4]. SQN is a firm rock site. The foundation material and foundation elevation for the Category I plant structures is described in Table 3.0-1.

Table 3.0-1: Category I Structures and Geotechnical Foundation Material

Category I Structure	Geotechnical Foundation Material	Applicable Elevation
Essential Raw Cooling Water Pumping Station	Shale/limestone bedrock	618 ft
Reactor Building, Unit 1 and Unit 2	Shale/limestone bedrock	661 ft
Auxiliary Building	Shale/limestone bedrock	661 ft
Control Building	Shale/limestone bedrock	661 ft
Diesel Generator Building and Additional Diesel Generator Building	Residual soil above limestone with interbedded shale rock	722 ft
Refueling Water Storage Tank, Unit 1 and Unit 2	Residual soil above limestone with interbedded shale rock	705 ft

3.1 Seismic Hazard Analysis

This section discusses the seismic hazard methodology, presents the final seismic hazard results used in the SPRA, and discusses important assumptions and important sources of uncertainty.

The seismic hazard analysis determines the annual frequency of exceedance (AFE) for selected ground motion parameters. The analysis involves use of earthquake source models, ground motion attenuation models, characterization of the site response (e.g., soil column), and accounts for the uncertainties and randomness of these parameters to arrive at the site seismic hazard. Detailed information regarding the SQN site hazard was provided to the NRC in the seismic hazard information submitted to the NRC in response to the NTTF 2.1 Seismic information request [4]. As further discussed below, a supplemental seismic hazard analysis has been performed for SQN [21].

3.1.1 Seismic Hazard Analysis Methodology

A supplemental seismic hazard analysis [21] was performed for the SQN SPRA in lieu of the NTTF 2.1 Submittal [4] since the site analysis develops the additional elements required for the SPRA such as Foundation Input Response Spectra (FIRS), hazard-consistent strain-compatible properties, and vertical ground motions.

The GMRS at SQN is defined at the foundation control point corresponding to the Reactor Building (RB) coupled with East Steam Valve Room (ESVR), and the Auxiliary-Control Building (ACB) coupled with the Additional Equipment Buildings (AEBs).

The following six FIRS are developed for the structures listed in Table 3.0-1 and are summarized below:

- *GMRS/FIRS1* – equivalent to GMRS. FIRS1 corresponds to a surface-founded FIRS located at a control point corresponding to the surface spectra at elevation 661 ft above mean sea level (MSL), at the base of the RB coupled with the ESVR, and the ACB coupled with the AEB. The control point elevation adopted in the NTTF 2.1 Submittal [4] was defined at the base of the containment structures at a depth of 64 ft below the plant grade elevation, i.e., elevation 641 ft above MSL. Both the GMRS/FIRS1 as defined here and the NTTF 2.1 Submittal [4] GMRS lie in the same limestone with interbedded shale rock that is part of the Conasauga Formation of Middle Cambrian age.
- *FIRS2* – corresponds to a surface-founded FIRS located at control point corresponding to the surface spectra at elevation 618 ft above MSL, at the base of the Essential Raw Cooling Water (ERCW) Pumping Station. FIRS2 lie in the same limestone with interbedded shale rock that is part of the Conasauga Formation of Middle Cambrian age.
- *FIRS3* – corresponds to a surface-founded FIRS located at control point elevation 722 ft above MSL, at the base of the Diesel Generator Building (DGB) and Additional Diesel Generating Building (ADGB). FIRS3 consists of 10 ft of Class A backfill with native earth/residual soils down to an average elevation of 667 ft above MSL above the limestone with interbedded shale rock.
- *FIRS4* – corresponds to a surface-founded FIRS located at control point elevation 705 ft above MSL and corresponds to the input for yard equipment. FIRS4 consists of approximately 38 ft of Class A backfill above limestone with interbedded shale rock at an average elevation of elevation 667 ft above MSL. FIRS4 is applicable to yard equipment above Class A backfill material above the limestone with interbedded shale rock.
- *FIRS5* – corresponds to a surface-founded FIRS located at control point elevation 705 ft above MSL and is similar to FIRS4, with the exception that the Class A backfill in FIRS4 did not replace the native earth/residual soils above the limestone with interbedded shale rock at an average elevation of 667 ft above MSL. FIRS5 is applicable to yard equipment above residual soils above the limestone with interbedded shale rock.
- *FIRS6* – corresponds to a surface-founded FIRS located at control point elevation 705 ft above MSL and consists of 15 ft of engineered fill with residual soils above limestone with interbedded shale rock at an average elevation 667 ft above MSL. FIRS6 is applicable to the Refueling Water Storage Tank (RWST).

To perform the site response analyses for SQN, a random vibration theory approach was employed. This process is consistent with existing NRC guidance and the SPID [3]. The guidance contained in Appendix B of the SPID [3] on incorporating epistemic uncertainty in shear-wave velocities, non-linear dynamic properties and source spectra was followed for SQN in addition to development of High Frequency (HF) and Low Frequency (LF) controlling earthquakes (control motions) per recommendations in NRC Regulatory Guide (RG) 1.208 [13] for mean annual frequency of exceedance (MAFE) corresponding to 10^{-2}, 10^{-3}, 10^{-4}, 10^{-5}, and 10^{-6} at reference rock.

Idealized shear-wave velocity profiles were developed incorporating the existing geotechnical data, onshore geophysics survey, and the derived geologic profile at depth derived for the SQN NTTF 2.1 Seismic Hazard submittal [4], along with the general guidelines included in the SPID [3] to account for the soil profiles epistemic uncertainty and aleatory variability. The idealized shear-wave velocities developed for each of the three base case profiles for GMRS/FIRS1 through FIRS6 are presented in Figures 3.1-1 to 3.1-6, respectively.

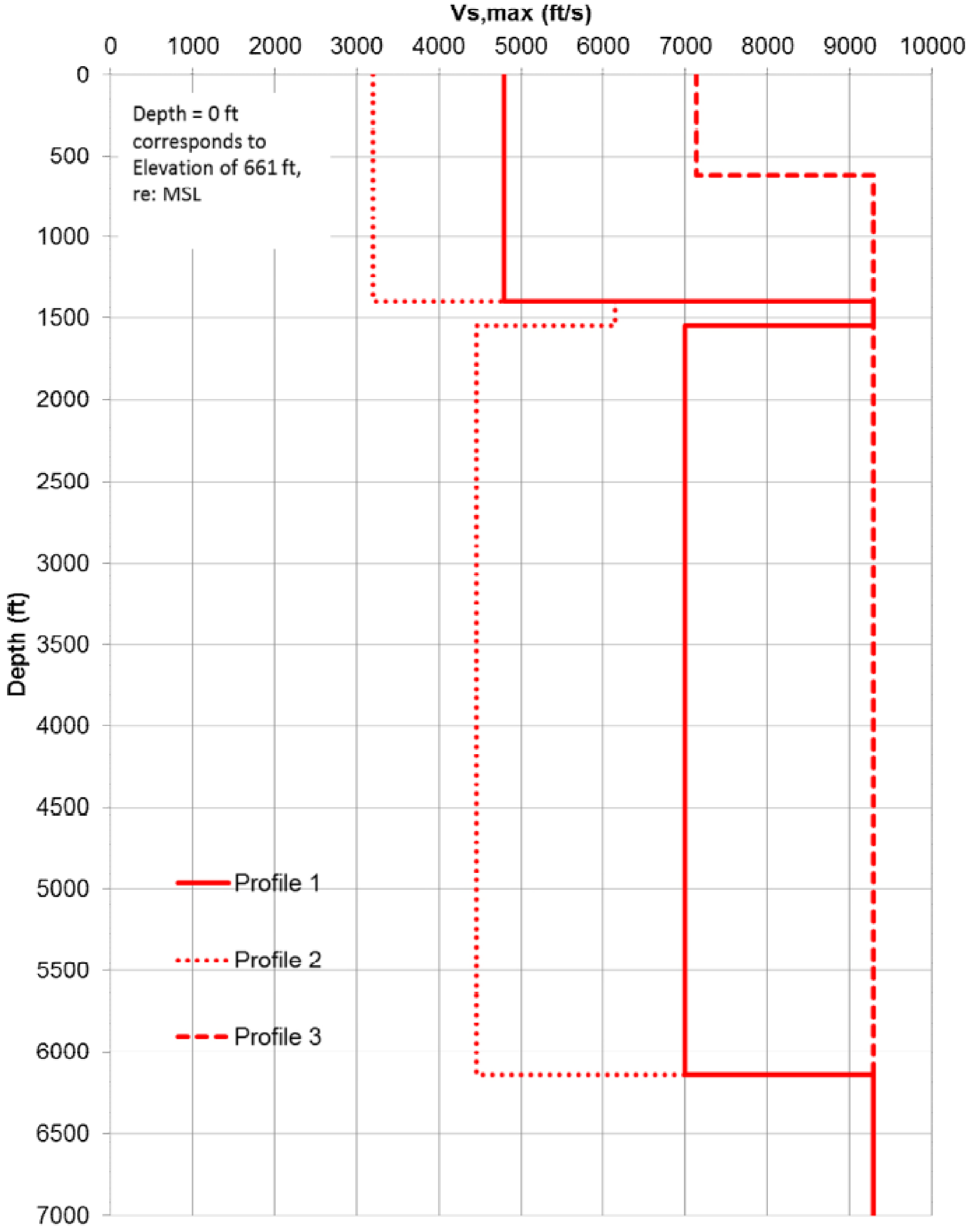


Figure 3.1-1: Idealized Shear-wave Velocity (V_S) Profile Representing Epistemic Uncertainty (GMRS/FIRS1)

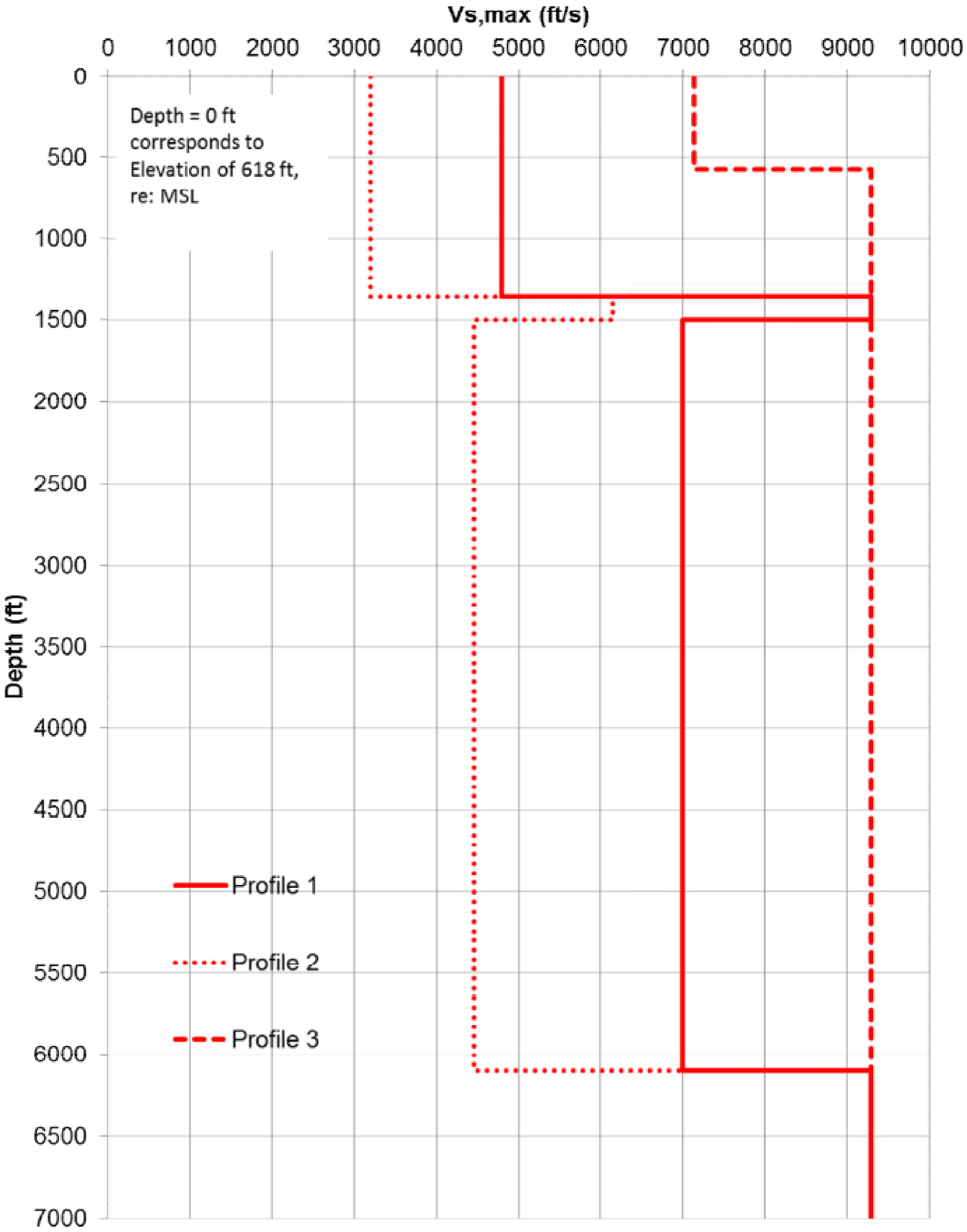


Figure 3.1-2: Idealized Shear-wave Velocity (V_S) Profile Representing Epistemic Uncertainty (FIRS2)

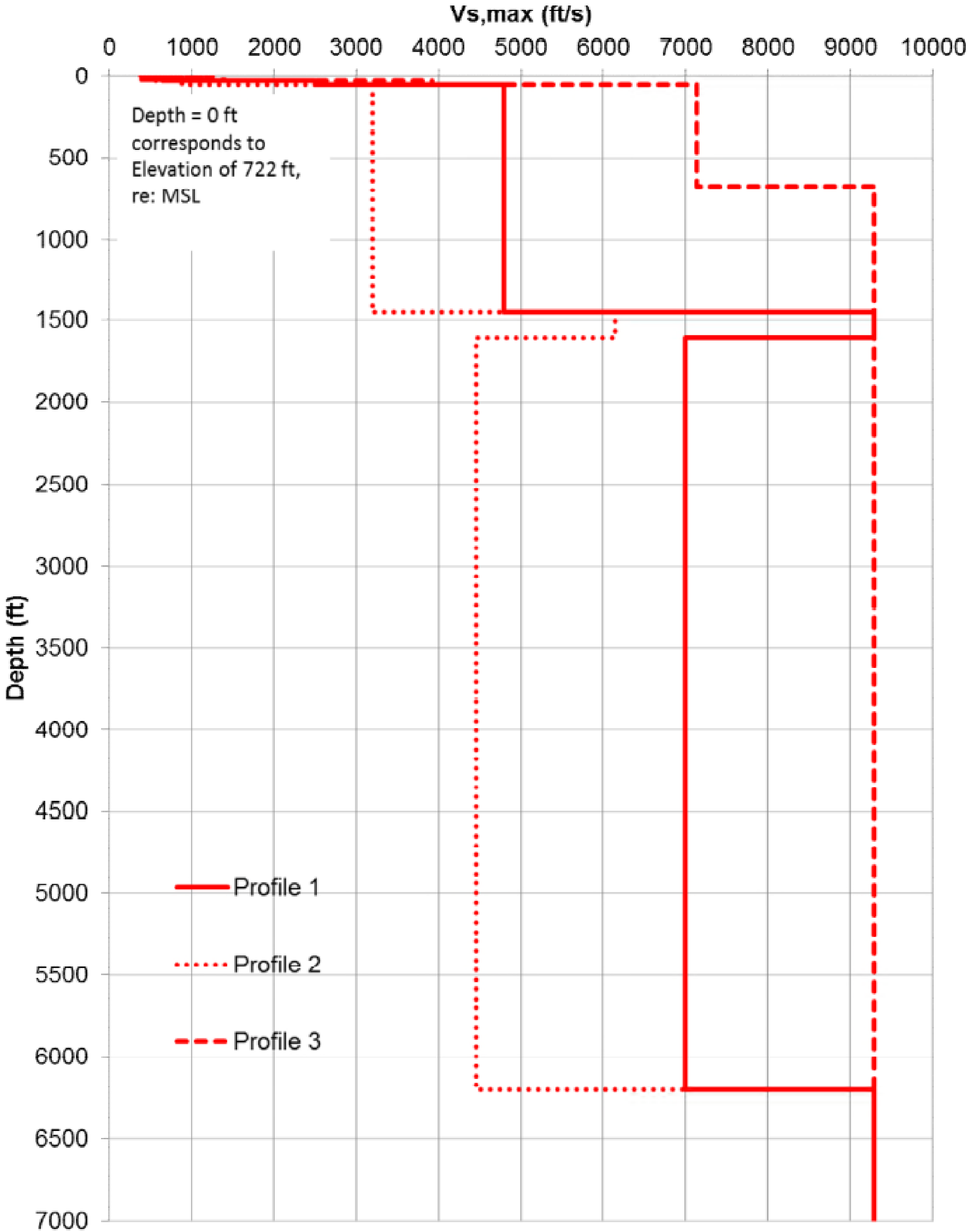


Figure 3.1-3: Idealized Shear-wave Velocity (V_S) Profile Representing Epistemic Uncertainty (FIRS3)

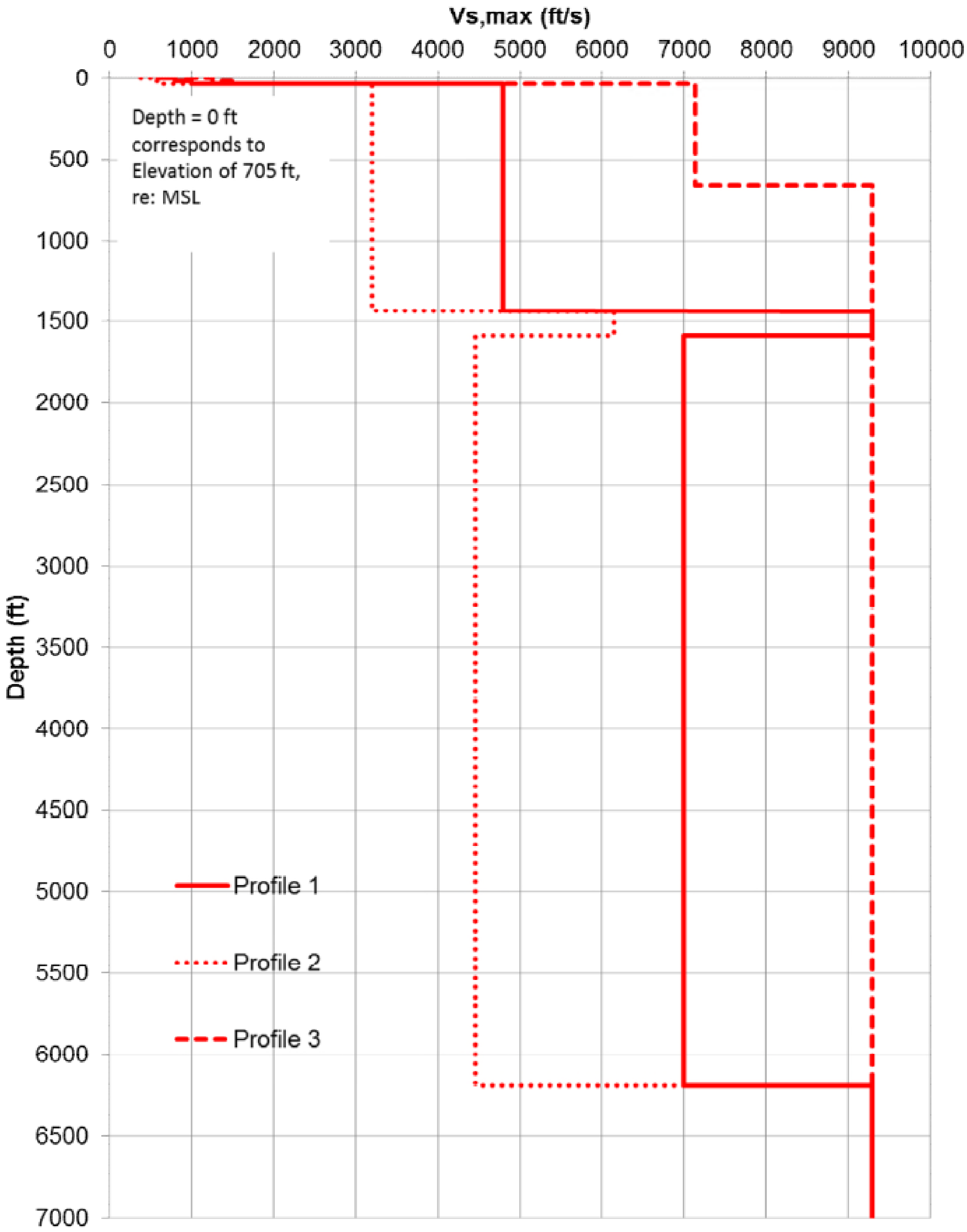


Figure 3.1-4: Idealized Shear-wave Velocity (V_S) Profile Representing Epistemic Uncertainty (FIRS4)

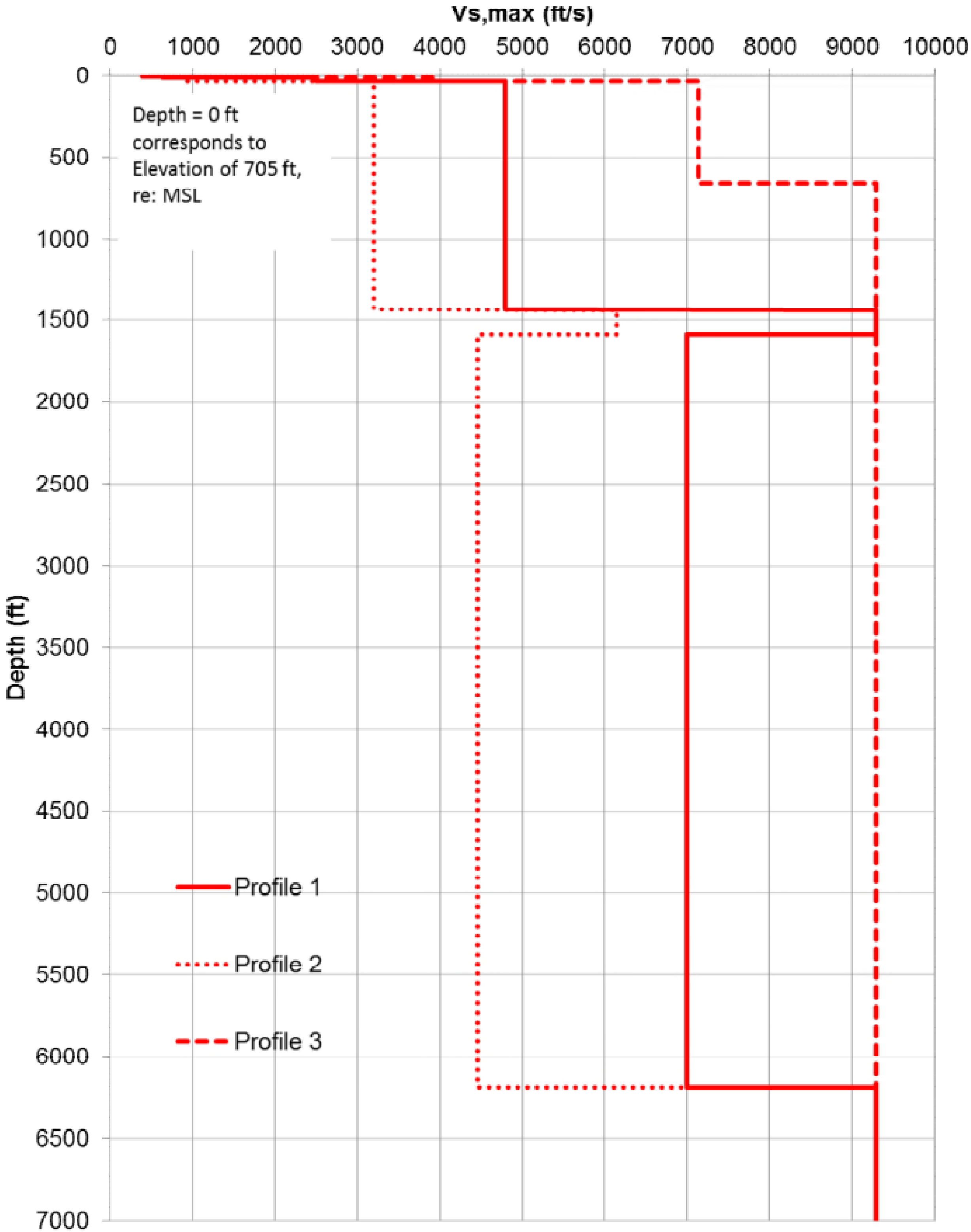


Figure 3.1-5: Idealized Shear-wave Velocity (Vs) Profile Representing Epistemic Uncertainty (FIRS5)

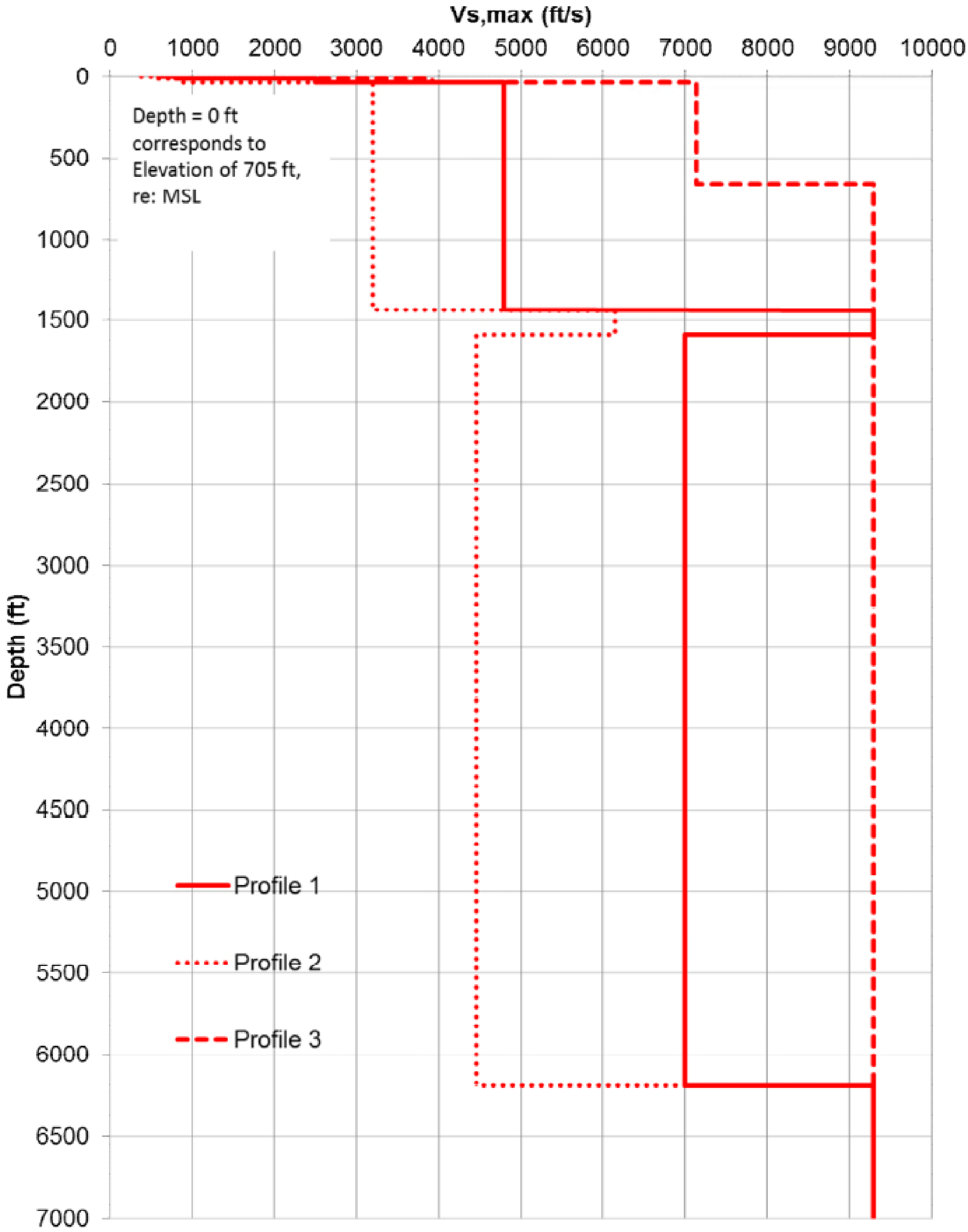


Figure 3.1-6: Idealized Shear-wave Velocity (V_S) Profile Representing Epistemic Uncertainty (FIRS6)

To accommodate the full range in expected dynamic material behavior for the firm rock profiles, linear and nonlinear soil dynamic models were included, with equal weights given to each approach. Peninsular Range curves were also applied to the native soils/residual soils for FIRS3, FIRS5, and FIRS6. Shear modulus reduction and hysteretic damping curves were used for the various soil layers for the six FIRS. The base case profiles were randomized to account for aleatory variability in shear-wave velocities and dynamic material properties; sixty randomized profiles were generated.

The results of the site response analyses consist of amplification factors that describe the amplification (or de-amplification) of hard reference rock motion as a function of frequency and input reference rock amplitude. The amplification factors are represented in terms of a median amplification value and an associated standard deviation (sigma) for each oscillator frequency and input rock amplitude. Consistent with the SPID [3], a minimum median amplification value of 0.5 was employed in the present analysis.

The site amplification factors and logarithmic standard deviations are inputs to develop the full set of site-specific hazard curves that accommodate the randomness and uncertainty in the local dynamic material properties. Sample amplification factors are presented in Figure 3.1-7.

The seismic hazard calculations use a minimum earthquake moment magnitude of 5.0 since the cumulative absolute velocity filter is not used. Soil seismic hazard curves are calculated for frequencies of 0.5, 1, 2.5, 5, 10, and 25 Hz and peak ground acceleration (PGA) (100 Hz). Horizontal uniform hazard response spectrum (UHRS) are calculated for MAFEs of 10^{-2}, 10^{-3}, 10^{-4}, 10^{-5}, and 10^{-6}.

The GMRS and FIRS were developed in accordance with NRC RG 1.208 [13]. Sixty randomizations were generated for the site response for each epistemic branch in the soil logic tree, compared to a minimum of thirty recommended in the SPID [3]. The site response analyses were completed using the HF and LF control motions. Site-specific horizontal hazard curves for each of the FIRS site conditions were used and were developed using Approach 3 of NUREG/CR-6728 [45].

Vertical spectra are developed using vertical-over-horizontal (V/H) scaling relations. The idealized V/H ratios are used to derive the vertical design response spectra from their horizontal equivalents. The procedure is consistent with the methodology described in EPRI 3002004396 [35].

The reference earthquake ground motion to which the fragilities are referenced is represented by the horizontal GMRS also at the RB foundation control point. The PGA is the ground motion parameter used for the SPRA.

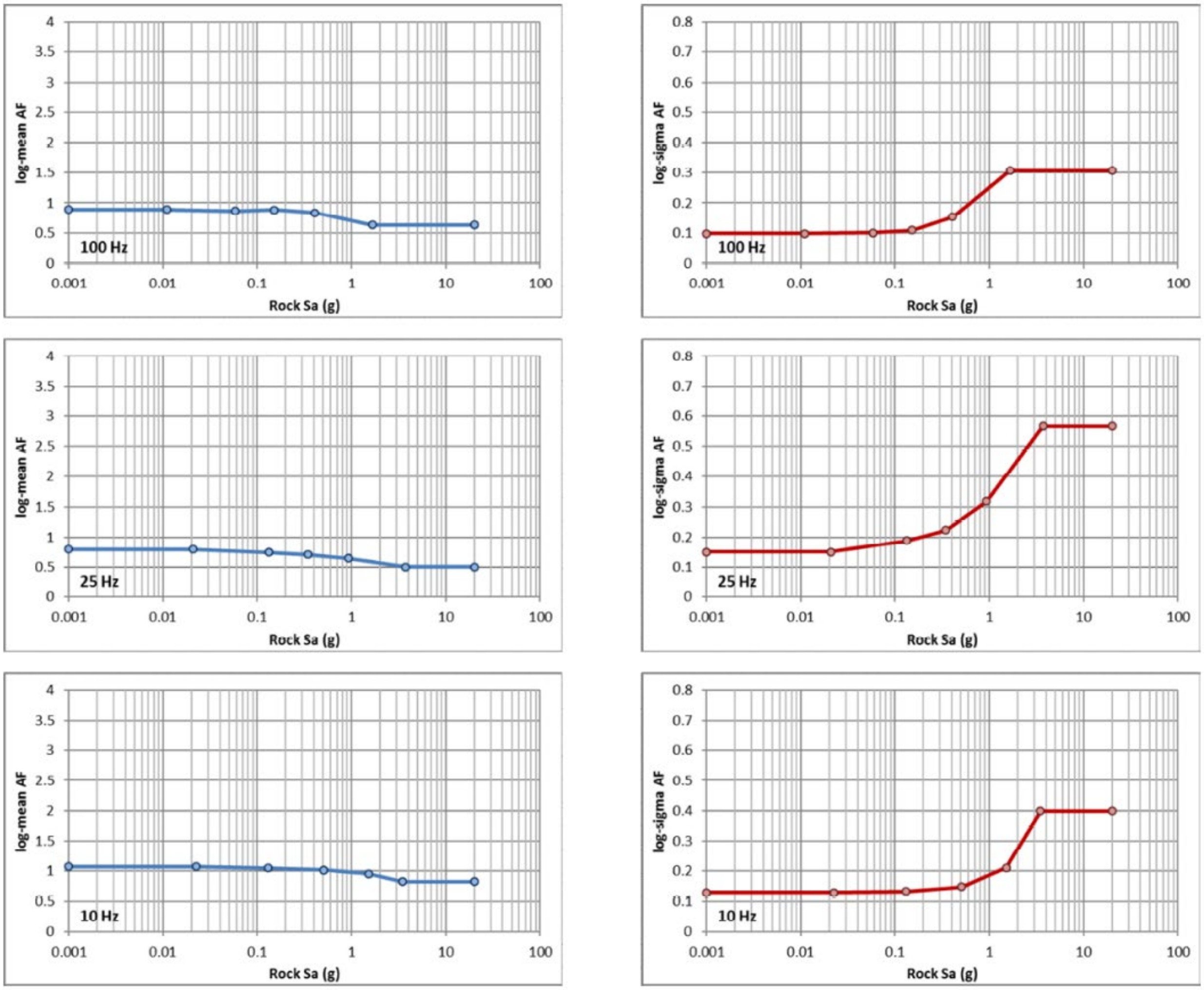


Figure 3.1-7: GMRS/FIRS1 Soil Profile Site Amplification Factor and Logarithmic Sigmas (100 Hz, 25 Hz, and 10 Hz)

3.2 Comparison of NTTF 2.1 Seismic Hazard Submittal and PRA Supplemental Seismic Hazard Analysis

The SQN SPRA used the supplemental seismic hazard analysis documented in SQN Probabilistic Seismic Hazard Analysis (PSHA) report [21]. Table 3.2-1 and Figure 3.2-1 provide the vertical and horizontal GMRS.

An onshore geophysics program encompassing SQN was completed to better define the shear-wave velocities of the SQN units. The existing geotechnical information available at the project site was used to characterize the depth of the various units, e.g., limestone with interbedded shale, and the shear-wave velocities from the geophysics were then assigned to their corresponding units, since the geophysics surveys profiles were acquired at the perimeter of the SQN site.

Figures 3.2-2 and 3.2-3 compare the NTTF 2.1 Seismic Hazard submittal, assessed by the NRC staff [18], with the SPRA Supplemental Seismic Hazard Analysis.

Figure 3.2-2 shows the idealized site profiles developed. The key difference between the base profiles developed in the current study and the NTTF 2.1 Submittal [4] study is that the profiles developed in the current study are softer without a base case profile that corresponds to a hard rock (average time-weighted shear-wave velocity greater than or equal to 9,200 fps) outcrop at the ground surface.

Figure 3.2-3 compares the NTTF 2.1 Submittal GMRS and the current GMRS/FIRS1 (both considered equivalent). As expected, since the site was idealized as being softer, the high-frequency spectral accelerations (>10 Hz) are lower than the NTTF 2.1 Submittal GMRS and higher at the low frequencies less than 0.5 Hz. Overall, the shapes of the spectra are comparable.

Table 3.2-1 Smoothed Horizontal and Vertical GMRS/FIRS1 and V/H Ratio

Frequency (Hz)	Horizontal GMRS/FIRS1 (g)	Vertical GMRS/FIRS1 (g)	V/H Ratio
0.1	1.66E-02	1.16E-02	6.95E-01
0.125	2.16E-02	1.50E-02	6.95E-01
0.15	2.71E-02	1.88E-02	6.95E-01
0.2	3.92E-02	2.72E-02	6.95E-01
0.3	6.19E-02	4.30E-02	6.95E-01
0.4	7.94E-02	5.52E-02	6.95E-01
0.5	9.80E-02	6.81E-02	6.95E-01
0.6	1.17E-01	8.11E-02	6.95E-01
0.7	1.34E-01	9.29E-02	6.95E-01
0.8	1.44E-01	9.98E-02	6.95E-01
0.9	1.46E-01	1.01E-01	6.95E-01
1	1.49E-01	1.04E-01	6.95E-01
1.25	1.66E-01	1.16E-01	6.95E-01
1.5	1.88E-01	1.31E-01	6.95E-01
2	2.49E-01	1.73E-01	6.95E-01
2.5	3.09E-01	2.15E-01	6.95E-01
3	3.51E-01	2.44E-01	6.95E-01
4	4.34E-01	3.02E-01	6.95E-01
5	5.00E-01	3.48E-01	6.95E-01
6	5.58E-01	3.88E-01	6.95E-01
7	6.06E-01	4.21E-01	6.95E-01
8	6.45E-01	4.48E-01	6.95E-01
9	6.81E-01	4.73E-01	6.95E-01
10	7.07E-01	4.92E-01	6.95E-01
12.5	7.32E-01	5.08E-01	6.95E-01
15	7.22E-01	5.06E-01	7.01E-01
20	6.79E-01	5.00E-01	7.36E-01
25	6.29E-01	5.01E-01	7.97E-01
30	5.61E-01	4.80E-01	8.55E-01
35	5.00E-01	4.51E-01	9.02E-01
40	4.55E-01	4.34E-01	9.52E-01
45	4.25E-01	4.26E-01	1.00E+00
50	4.06E-01	4.17E-01	1.03E+00
60	3.80E-01	3.92E-01	1.03E+00
70	3.65E-01	3.74E-01	1.03E+00
80	3.57E-01	3.59E-01	1.01E+00
90	3.54E-01	3.46E-01	9.77E-01
100	3.53E-01	3.38E-01	9.57E-01

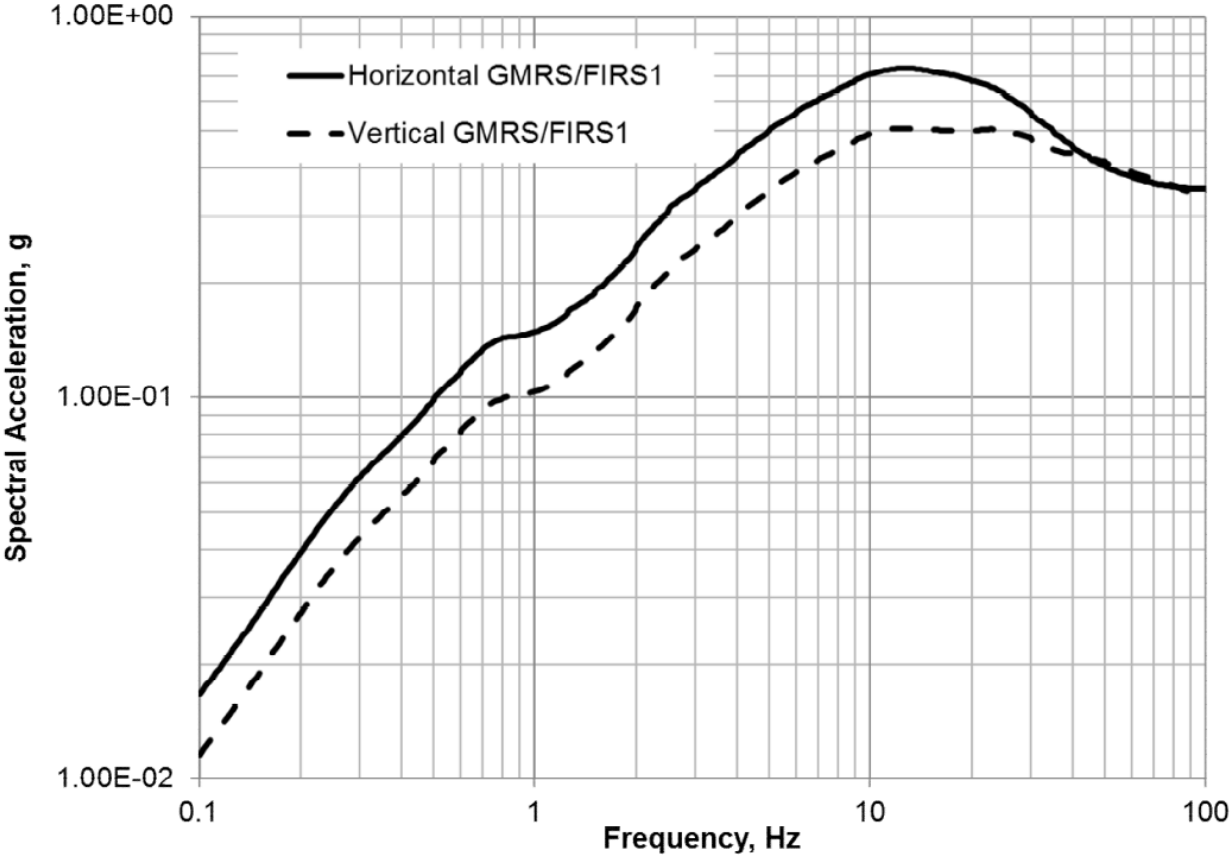


Figure 3.2-1: Horizontal and Vertical GMRS/FIRS1

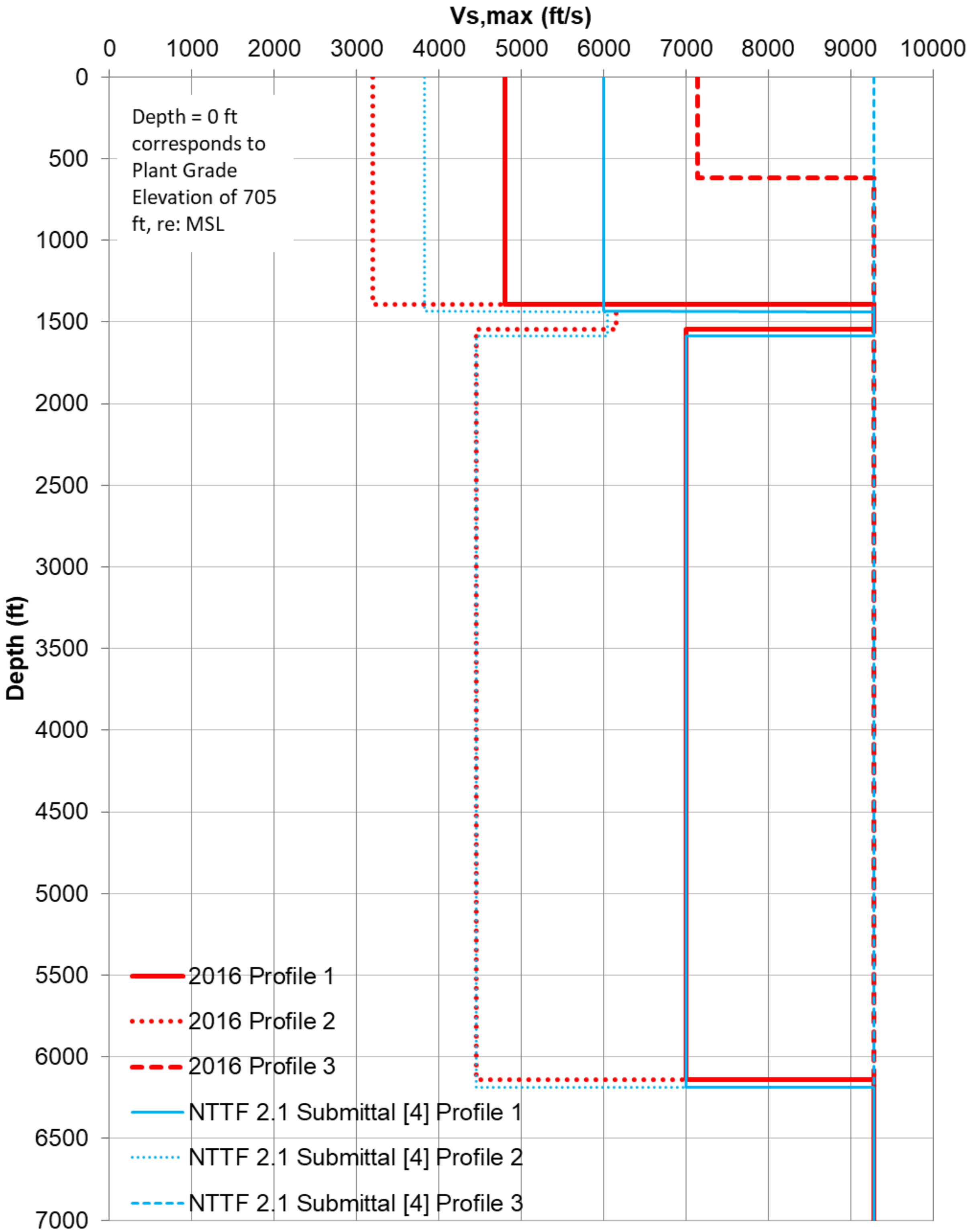


Figure 3.2-2: Comparison of Base Case Soil Profiles NTTF 2.1 Seismic Hazard Submittal and SPRA Supplemental Seismic Hazard Analysis

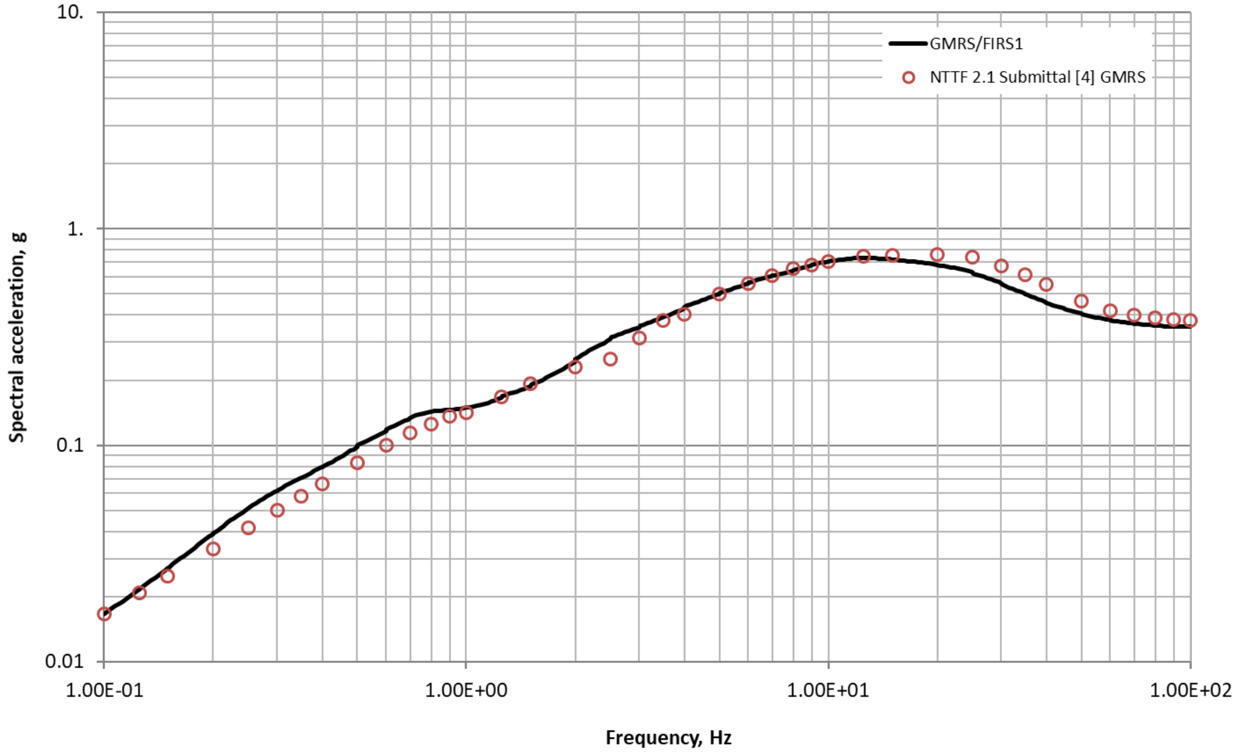


Figure 3.2-3: Comparison of Horizontal GMRS/FIRS1 NTTF 2.1 Seismic Hazard Submittal and SPRA Supplemental Seismic Hazard Analysis

3.2.1 Seismic Hazard Analysis Technical Adequacy

The SQN SPRA hazard methodology and analysis was subjected to an independent peer review against the pertinent requirements in the ASME/ANS PRA Standard [8]. After completion of the subsequent independent assessment, the full set of supporting requirements was met. The seismic hazard analysis was determined to be acceptable for use in the SPRA.

The peer review assessment, and subsequent disposition of peer review findings through an independent assessment, is further described in Appendix A and references [7] and [20].

3.2.2 Seismic Hazard Analysis Results and Insights

Table 3.2-2 and Figure 3.2-4 present the mean and fractile exceedance frequencies for hard rock at 100 Hz. Table 3.2-2 provides the final seismic hazard results used as input to the SQN SPRA, in terms of exceedance frequencies as a function of PGA level at hard rock.

Table 3.2-2 SQN GMRS/FIRS1 Mean and Fractile Exceedance Frequencies at PGA (100 Hz)

Amplitude (g)	Mean	Fractile Hazard Curves				
		0.05	0.16	0.5	0.84	0.95
0.0001	1.668E-01	9.484E-02	1.280E-01	1.660E-01	2.051E-01	2.406E-01
0.00025	1.301E-01	5.999E-02	1.014E-01	1.302E-01	1.615E-01	1.853E-01
0.0005	9.932E-02	3.913E-02	7.532E-02	9.880E-02	1.281E-01	1.481E-01
0.00075	8.039E-02	2.971E-02	5.977E-02	7.809E-02	1.062E-01	1.267E-01
0.001	6.736E-02	2.437E-02	4.940E-02	6.386E-02	9.000E-02	1.120E-01
0.0015	5.072E-02	1.835E-02	3.583E-02	4.755E-02	6.730E-02	9.205E-02
0.002	4.067E-02	1.490E-02	2.716E-02	3.837E-02	5.330E-02	7.949E-02
0.003	2.923E-02	1.056E-02	1.773E-02	2.752E-02	3.769E-02	6.360E-02
0.005	1.885E-02	6.337E-03	1.022E-02	1.689E-02	2.551E-02	4.616E-02
0.0075	1.299E-02	3.980E-03	6.333E-03	1.120E-02	1.838E-02	3.389E-02
0.01	9.794E-03	2.793E-03	4.304E-03	8.199E-03	1.421E-02	2.649E-02
0.015	6.344E-03	1.684E-03	2.274E-03	4.969E-03	9.527E-03	1.797E-02
0.02	4.530E-03	1.075E-03	1.438E-03	3.300E-03	7.245E-03	1.346E-02
0.03	2.715E-03	5.472E-04	7.898E-04	1.763E-03	4.553E-03	8.356E-03
0.05	1.370E-03	2.333E-04	3.490E-04	8.476E-04	2.239E-03	4.351E-03
0.075	7.806E-04	1.285E-04	1.874E-04	4.791E-04	1.261E-03	2.379E-03
0.1	5.147E-04	8.474E-05	1.237E-04	3.168E-04	8.240E-04	1.548E-03
0.15	2.753E-04	4.421E-05	6.909E-05	1.686E-04	4.417E-04	8.308E-04
0.2	1.700E-04	2.749E-05	4.333E-05	1.076E-04	2.728E-04	5.221E-04
0.3	7.964E-05	1.205E-05	2.013E-05	5.130E-05	1.324E-04	2.526E-04
0.5	2.571E-05	3.187E-06	5.892E-06	1.598E-05	4.480E-05	9.003E-05
0.75	8.894E-06	8.106E-07	1.710E-06	5.346E-06	1.564E-05	3.343E-05
1	3.841E-06	2.434E-07	5.895E-07	2.148E-06	6.929E-06	1.509E-05
1.5	1.059E-06	3.068E-08	1.050E-07	5.447E-07	1.964E-06	4.347E-06
2	3.983E-07	4.545E-09	2.450E-08	1.684E-07	7.105E-07	1.752E-06
3	9.237E-08	2.622E-11	1.738E-09	3.143E-08	1.470E-07	4.206E-07
5	1.256E-08	2.201E-29	1.410E-12	1.880E-09	1.630E-08	5.960E-08
7.5	2.237E-09	2.200E-29	7.748E-25	1.353E-10	2.318E-09	1.072E-08
10	6.068E-10	2.200E-29	4.054E-27	1.551E-11	5.359E-10	2.728E-09

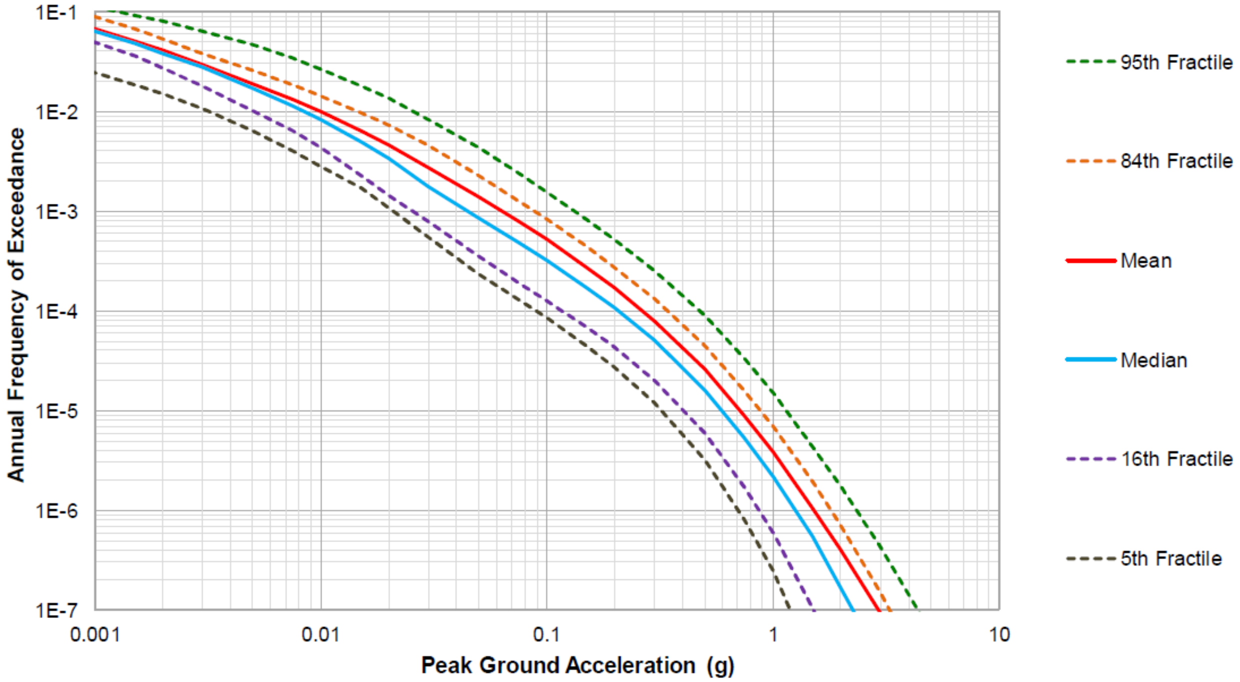


Figure 3.2-4: PGA (100 Hz) GMRS/FIRS1 Soil Profile Fractile Hazard Curves for SQN

3.2.2.1 Uncertainties in the Seismic Hazard Result from Input Parameters and Models

The epistemic and aleatory uncertainties in components of the model, including seismic source characterization and ground motion models, were incorporated using logic trees. Sensitivity analyses were also performed to assess the input parameters. Sensitivity analyses were performed on the ground motion models and several of the seismic source characterization, including alternatives for magnitude completeness, alternate earthquake recurrence rates, and maximum magnitude alternatives. Based on the sensitivity analyses performed, the epistemic uncertainty in the ground motion models dominates the contribution to the total epistemic uncertainty for the SQN site.

The Central and Eastern United States Seismic Source Characterization (CEUS-SSC) concluded its data gathering efforts in 2008. As a result, a literature search of published and unpublished data was completed to identify any data that may have an impact on the SSC, or any other site-specific modifications based on new information. An updated CEUS-SSC seismicity catalog was developed for the whole CEUS-SSC Study Region for the period of January 1, 2009 through January 31, 2015 for the region encompassed by the 250-mile (400-km) radius around the SQN site. The final seismicity catalog used for the SQN PSHA is the combination of the original CEUS-SSC seismicity catalog (1568 through 2008) and the updated SQN site regional catalog (January 1, 2009 through January 31, 2015). After the review and studies of new information, it was concluded that the CEUS-SSC recurrence parameters did not require an update.

The PSHA performed incorporated the entire CEUS-SSC logic tree published in NUREG-2115 [12] with its revisions published in 2015. The only ‘simplification’ performed to the entire CEUS-SSC was related to using point sources for the background sources. No seismic sources were screened out of the analyses. The use of point sources for modeling the background sources is supported by the sensitivities presented in NUREG-2115 [12].

3.2.2.2 Horizontal and Vertical GMRS

This section provides the control point horizontal and vertical GMRS.

The GMRS at the control point is provided in Table 3.2-1 and plotted in Figure 3.2-1. The development of the control point response spectra is summarized in Section 3.1 and further described in detail in the SQN PSHA report [21].

3.2.2.2.1 Vertical GMRS

Vertical ground motions were developed by applying V/H ratios to the horizontal GMRS and FIRS. A logic tree was adopted to incorporate epistemic uncertainty by weighting alternative models consistent with the methodology in EPRI 3002004396 [35]. The development of the V/H ratios is documented in the SQN PSHA report [21].

Table 3.2-1 summarizes the horizontal and vertical response spectra at the control point. Figure 3.2-5 provides a plot of the vertical and horizontal GMRS as well as V/H ratios.

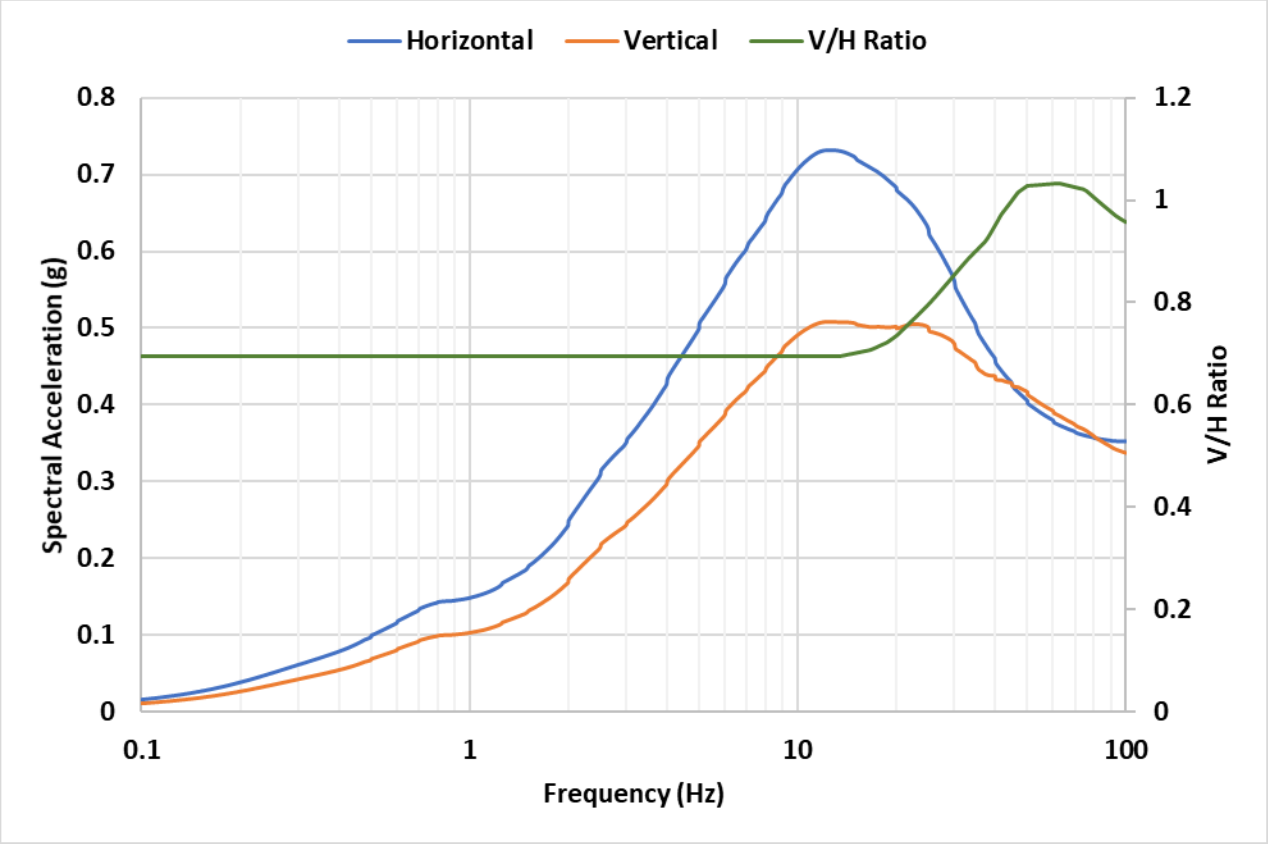


Figure 3.2-5: Horizontal and Vertical GMRS/FIRS1 and V/H Ratio

3.3 Soil Failure and Fragility Analysis

The SPRA soil failure and fragility analysis is performed in the report CJC-SQN-C-001 [46]. Soil failure modes considered in the analysis include liquefaction, seismic induced settlements, seismic induced lateral deformation, slope stability, sliding of earth and building structures, and seismic bearing capacity. The evaluations performed and described in this report followed an overall graded approach for developing soil failure mode fragilities for inputs to the SQN SPRA model. The graded approach uses increasing levels of rigor for screening out or estimating soil fragilities depending upon the contribution to risk of a given soil failure mode.

The Category I structures at SQN that may be susceptible to damage as a result of ground motions due to earthquakes were identified and either screened out, evaluated using scaling of results of existing analyses, or analyzed to develop estimates of deformation and behavior, as shown on Table 3.3-1. These analyses were based on geotechnical data available at the site using contemporary methodologies to estimate slope stability, vertical settlement, and lateral deformation, as appropriate.

The ground motion levels and associated site amplification factors for the analysis are taken from the SQN PSHA report [21].

Table 3.3-1 SQN Soil Failure and Fragility Analysis

Structure	Geotechnical Foundation Material	Evaluation
Reactor Building and Steel Containment Vessel	Shale/limestone bedrock	Screened out[1]
Auxiliary Control Building	Shale/limestone bedrock	Screened out[1]
Additional Equipment Building	Shale/limestone bedrock	Screened out[1]
Condenser Cooling Water Pumping Station	Shale/limestone bedrock	Screened out[1]
Condenser Cooling Water Pumping Station Retaining Walls	Shale/limestone bedrock	Lateral Deformation
East Steam Valve Rooms	Shale/limestone bedrock	Lateral Deformation Vertical Settlement Screened Out
Diesel Generator Building	Backfill/In-situ soil	Slope Stability
Underground Concrete Encased Electrical Conduit Banks, Manholes, and Handholes for Class 1E Circuits	Backfill/In-situ soil	Vertical Settlement and Lateral Deformation
Auxiliary Building - Essential Raw Cooling Water Pipe Tunnel	Shale/limestone bedrock	Screened out[1]
Essential Raw Cooling Water Intake Pumping Structure	Shale/limestone bedrock	Screened out
Essential Raw Cooling Water Access Dike Cells	Shale/limestone bedrock	Screened out[1]
Intake Canal Slope Stability	Shale/limestone bedrock	Slope Stability
Pile Supported Essential Raw Cooling Water Piping Support Slab	Shale/limestone bedrock	Vertical Settlement and Lateral Deformation

Table 3.3-1 SQN Soil Failure and Fragility Analysis

Structure	Geotechnical Foundation Material	Evaluation
Refueling Water Storage Tanks/Foundations	Backfill/In-situ soil	Vertical Settlement and Lateral Deformation
Piping Tunnels Containing Class A, B, C, or D Piping or Tubing	In-situ soil	Vertical Settlement and Lateral Deformation
Condensate Demineralizer Waste Evaporator Building	Shale/limestone bedrock	Lateral Deformation Vertical Settlement Screened Out
Waste Packaging Area of the Auxiliary Building	Shale/limestone bedrock	Lateral Deformation Vertical Settlement Screened Out
Additional Diesel Generator Building	Backfill/In-situ soil	Slope Stability and Lateral Deformation

[1] Lateral soil pressures calculated

3.3.1 Soil Failure and Fragility Analysis Technical Adequacy

The SQN Soil Failure and Fragility Analysis methodology and analysis was subjected to an independent peer review against the pertinent requirements in the ASME/ANS PRA Standard [8]. After completion of the subsequent independent assessment, the full set of supporting requirements was met. The seismic hazard analysis was determined to be acceptable for use in the SPRA.

The peer review assessment, and subsequent disposition of peer review findings through an independent assessment, is further described in Appendix A and references [7] and [20].

3.3.2 Soil Failure and Fragility Analysis Results and Insights

In general, the resulting soil deformations were used as input to fragility analyses of important SSCs. The controlling failure mode of interest is the ERCW buried piping susceptible to movement in the soil around the Diesel Generator Building. Ground motions associated with AFE between 10^{-4} and 10^{-7} were considered in the analysis.

Equivalent linear soil models were used for this analysis. Sensitivity studies were used to evaluate the effect of the use of equivalent linear soil model given relatively large shear strains that develop for the larger ground motions. The results of the sensitivity analyses demonstrate that for soil failure and fragility evaluations, the use of the equivalent linear models introduces a slight conservative bias into the evaluations. Additional sensitivity analyses were completed to test the impact of the conservative bias to the risk assessment. Sensitivity studies 16 and 17, as shown in Table 5.7-1, increase and decrease the ERCW piping fragility by 30 percent with no change to risk.

4.0 Determination of Seismic Fragilities for the SPRA

This section provides a summary of the process for identifying and developing fragilities for SSCs that participate in the plant response to a seismic event for the SQN SPRA. The subsections provide brief summaries of these elements.

4.1 Seismic Equipment List

For the SQN SPRA, a seismic equipment list (SEL) was developed to include SSCs that are important to achieving safe shutdown following a seismic event and to mitigating radioactivity release if core damage occurs, and that are included in the SPRA model. The methodology used to develop the SEL is consistent with the guidance provided in EPRI 3002000709 [22].

4.1.1 SEL Development

The comprehensive SEL was developed by starting with the list of components modeled in the SQN IEPRA, including internal flooding. That list was then augmented by reviewing equipment contained in the SQN individual plant examination of external events (IPEEE), fire safe shutdown equipment lists (SSELs), and the NTTF 2.3 seismic walkdown equipment list. Diverse and flexible coping strategies (FLEX) systems included in the model were added to the SEL. Table 4.1-1 includes a list of systems considered in the SEL development. In addition, a separate effort was conducted by the Human Reliability Analysis (HRA) analyst to identify instrumentation needed by operators to support actions modeled in the IEPRA. Components typically not modeled in IEPRAs, such as cable trays; conduits; motor control centers (MCCs); electrical cabinets and panels; heating, ventilating, and air conditioning (HVAC) ducting; and piping, were identified and included in the SEL. The SEL was also updated after the seismic walkdowns to incorporate additional items such as block walls. The final comprehensive SEL includes any additional SSCs identified after the seismic walkdowns (i.e., relay and breaker chatter events that could not be screened out). The SEL includes structures, buildings and substructures that either contain safety-related equipment or whose failure could impact safety functions or cause a reactor trip. The SEL includes nuclear steam supply system (NSSS) components and components required for containment integrity.

The resulting SEL includes a total of about 5,300 component entries for both units combined (including common components). The final SEL was documented for the SPRA in the SQN Probabilistic Risk Assessment - Seismic Equipment List (SEL) Report [23].

Table 4.1-1 Systems Considered for the Seismic Equipment List

System Number	System	Mitigation Potential	In IE PRA	In IPEEE	In 2.3 SWEL	In Appendix R	Notes
1	MAIN STEAM	Yes	Yes	Yes	Yes	Yes	
2	CONDENSATE	Yes	Yes	No	No	No	System screened out
3	MAIN & AUXILIARY FEEDWATER	Yes	Yes	Yes	Yes	Yes	
5	EXTRACTION STEAM	No	No	No	No	No	
6	HEATER DRAINS & VENTS	No	Yes	No	No	No	System screened out
7	TURBINE EXTRACTION TRAPS & DRAINS	No	No	No	No	No	
8	MISC TURBINE CONNECTIONS	No	No	No	No	No	
9	MISC TURBINE VENTS	No	No	No	No	No	
12	AUXILIARY BOILER	No	No	No	No	No	
13	FIRE PROTECTION SYSTEM (OTHER THAN HIGH-PRESSURE FIRE PROTECTION AND CO2 FIRE PR)	No	No	No	No	No	
14	CONDENSATE DEMINERALIZER	No	Yes	No	No	No	System screened out
15	STEAM GENERATOR BLOWDOWN	No	Yes	No	No	Yes	
18	FUEL OIL	Yes	with EDG	Yes	No	No	
19	LIGHTING-OFF OIL AND AIR PIPING	No	No	No	No	No	
20	CENTRAL LUBRICATING OIL	No	No	No	No	No	
24	RAW COOLING WATER	Yes	Yes	No	No	No	System screened out
25	RAW SERVICE WATER	No	No	No	No	No	
26	HIGH PRESSURE FIRE PROTECTION	No	No	Yes	No	No	
27	CONDENSER CIRC WATER	Yes	Yes	No	No	No	System screened out
28	WATER TREATMENT	No	No	No	No	No	
29	POTABLE WATER DISTRIBUTION	No	No	No	No	No	
30	VENTILATING	Yes	Yes	Yes	Yes	Yes	
31	AIR CONDITIONING	No	No	Yes	Yes	Yes	
32	CONTROL AIR	Yes	Yes	Yes	Yes	Yes	
33	SERVICE AIR	No	Yes	No	No	No	
34	VACUUM PRIMING	No	No	No	No	No	
35	GENERATOR COOLING	No	No	No	No	No	

Table 4.1-1 Systems Considered for the Seismic Equipment List

System Number	System	Mitigation Potential	In IE PRA	In IPEEE	In 2.3 SWEL	In Appendix R	Notes
36	FEEDWATER SECONDARY TREATMENT	No	No	No	No	No	
37	GLAND SEAL WATER	Yes	Yes	No	No	No	System screened out
38	INSULATING OIL	No	No	No	No	No	
39	CO2 STORAGE, FIRE PROTECTION & PURGING	No	No	No	No	No	
40	STATION DRAINAGE	No	No	No	No	No	
41	LAYUP WATER TREATMENT	No	No	No	No	No	
42	CHEMICAL CLEANING	No	No	No	No	No	
43	SAMPLING AND WATER QUALITY	No	No	Yes	No	No	
44	BUILDING HEATING	No	No	No	No	No	
46	FEEDWATER CONTROL	No	No	No	No	Yes	
47	TURBOGENERATOR CONTROL	Yes	Yes	No	No	No	System screened out
49	BREATHING AIR	No	No	No	No	No	
50	HYPOCHLORITE	No	No	No	No	No	
51	RAW WATER CHLORINATION	No	No	No	No	No	
52	SYSTEM TEST FACILITY	No	No	No	No	No	
54	INJECTION WATER	No	Yes	No	No	No	System screened out
55	ANNUNCIATOR & SEQUENTIAL EVENTS RECORDING	No	No	No	No	No	
56	TEMPERATURE MONITORING	No	No	No	No	No	
57	ASSOCIATED ELECTRICAL	No	No	No	No	No	
58	GENERATOR (ISOLATED PHASE) BUS COOLING	No	No	No	No	No	
59	DEMINERALIZED WATER & CASK DECONTAMINATION	No	Yes	No	No	No	
61	ICE CONDENSER	Yes	No	Yes	No	No	
62	CHEMICAL AND VOLUME CONTROL	Yes	Yes	Yes	No	Yes	
63	SAFETY INJECTION	Yes	Yes	Yes	Yes	Yes	
64	ICE CONDENSER (PRIMARY) CONTAINMENT	Yes	No	No	No	No	
65	EMERGENCY GAS TREATMENT	No	No	No	No	No	
67	ESSENTIAL RAW COOLING WATER	Yes	Yes	Yes	Yes	Yes	

Table 4.1-1 Systems Considered for the Seismic Equipment List

System Number	System	Mitigation Potential	In IE PRA	In IPEEE	In 2.3 SWEL	In Appendix R	Notes
68	REACTOR COOLANT	Yes	Yes	Yes	Yes	Yes	
69	PLUMBING	No	No	No	No	No	
70	COMPONENT COOLING	Yes	Yes	Yes	No	Yes	
72	CONTAINMENT SPRAY (CS)	Yes	Yes	Yes	Yes	Yes	
74	RESIDUAL HEAT REMOVAL	Yes	Yes	Yes	Yes	Yes	
76	VOLUME REDUCTION & SOLIDIFICATION	No	No	No	No	No	
77	WASTE DISPOSAL	No	Yes	Yes	No	Yes	
78	SPENT FUEL PIT COOLING	No	No	Yes	Yes	No	
79	FUEL HANDLING & STORAGE	No	No	No	No	No	
80	PRIMARY CONTAINMENT COOLING	No	No	No	No	No	
81	PRIMARY MAKEUP WATER	Yes	No	Yes	No	No	
82	STANDBY DIESEL GENERATOR	Yes	Yes	Yes	Yes	No	
83	HYDROGEN RECOMBINATION	No	No	No	No	No	
84	FLOOD MODE BORATION MAKEUP	No	Yes	No	No	No	
85	CONTROL ROD DRIVE	No	No	No	No	No	
86	DIESEL STARTING AIR	Yes	No	No	No	No	This system is included in the standby diesel generator system
87	UPPER HEAD INJECTION	No	No	No	No	No	
88	CONTAINMENT ISOLATION	Yes	No	Yes	No	No	
90	RADIATION MONITORING	No	Yes	Yes	No	No	
92	NEUTRON MONITORING	No	Yes	Yes	No	Yes	
94	IN-CORE FLUX DETECTORS	No	No	No	No	No	
99	REACTOR PROTECTION	Yes	Yes	Yes	No	Yes	
150	MEASURING AND TEST EQUIPMENT	No	No	No	No	No	
200	STATUS MOINTOR SYSTEM	No	No	No	No	No	
201	480-V ELECTRICAL BOARDS AND MOTOR CONTROL CENTER	Yes	Yes	Yes	Yes	Yes	
202	6900-V ELECTRICAL BOARDS (AND LOGIC PANELS)	Yes	Yes	Yes	Yes	Yes	
232	REACTOR VENT POWER	No	No	No	No	No	

Table 4.1-1 Systems Considered for the Seismic Equipment List

System Number	System	Mitigation Potential	In IE PRA	In IPEEE	In 2.3 SWEL	In Appendix R	Notes
233	YARD & STREET LIGHTING	No	No	No	No	No	
234	HEAT TRACE EQUIPMENT	No	No	No	No	No	
235	120V AC VITAL POWER	Yes	No	No	No	No	No SSCs listed under this system number. These SSCs are included under System 250.
236	125V DC VITAL POWER	Yes	No	No	No	No	No SSCs listed under this system number. These SSCs are included under System 250.
237	120V AC INSTRUMENT POWER	Yes	No	No	No	No	No SSCs listed under this system number. These SSCs are included under System 250.
238	120V AC PREFERRED POWER	Yes	No	No	No	No	No SSCs listed under this system number. These SSCs are included under System 250.
239	250V DC POWER SYSTEM	Yes	No	No	No	No	No SSCs listed under this system number. These SSCs are included under System 250.
240	48V DC POWER	No	No	No	No	No	
241	SWITCHYARD AND TRANSFORMERS (INCLUDING 22.5,161, AND 500KV	Yes	Yes	No	No	No	
242	RADIATION MONITOR & SAMPLING POWER, PROCESS & AREA RADIATION MONITOR POWER	No	No	No	No	No	
243	RECORDING INSTRUMENT POWER	No	No	No	No	No	
244	COMMUNICATIONS SYSTEM	No	No	No	No	No	
245	SECURITY SYSTEMS	No	No	No	No	No	
246	MAIN RELAY BOARDS	No	No	No	No	No	

Table 4.1-1 Systems Considered for the Seismic Equipment List

System Number	System	Mitigation Potential	In IE PRA	In IPEEE	In 2.3 SWEL	In Appendix R	Notes
247	LIGHTING SYSTEMS	No	No	No	No	No	
248	ELECTRICAL CONTROL AND RECORDING INSTRUMENTS	No	No	No	No	No	
249	CONDENSATE DEMINERALIZER MOTOR CONTROL CENTER	No	No	No	No	No	
250	AD/DC LOW VOLTAGE POWER SYSTEM	No	Yes	Yes	Yes	Yes	This system includes SSCs previously under Systems 235, 236, 237, 238, and 239
251	SOUND POWERED TELEPHONES	No	No	No	No	No	
252	CAP AND INTERCOM	No	No	No	No	No	
253	VHF RADIO AND MICROWAVE	No	No	No	No	No	
254	CARRIER EQUIPMENT	No	No	No	No	No	
255	DATA ACQUISITION EQUIPMENT	No	No	No	No	No	
256	SHUTDOWN COMMUNICATIONS (SOUND POWERED)	No	No	No	No	No	
257	CLOSED CIRCUIT TELEVISION AND SECURITY	No	No	No	No	No	
258	MISCELLANEOUS AUDIO	No	No	No	No	No	
259	COMMUNICATION ROOM	No	No	No	No	No	
260	CHEMICAL LAB TEST EQUIPMENT	No	No	No	No	No	
261	PLANT COMPUTER	No	No	No	No	No	
262	LOAD SHED LOGIC	No	No	No	No	No	
263	CONDENSER TUBE CLEANING SYSTEM	No	No	No	No	No	
264	TECHNICAL SUPPORT SYSTEM	No	No	No	No	No	
265	HEALTH PHYSICS TEST LAB EQUIPMENT	NO	No	No	No	No	
268	PERMANENT HYDROGEN MITIGATION SYSTEM	Yes	Yes	Yes	No	No	
270	MISCELLANEOUS MOTOR OPERATED DOORS, PLATFORMS, HOISTS, ETC.	No	No	No	No	No	
271	CONTROL, AUXILIARY, AND REACTOR BLDG, MISCELLANEOUS	No	No	No	No	No	

Table 4.1-1 Systems Considered for the Seismic Equipment List

System Number	System	Mitigation Potential	In IE PRA	In IPEEE	In 2.3 SWEL	In Appendix R	Notes
272	WATER TREATMENT PLANT BUILDING CONDUIT AND CABLE TRAYS	No	No	No	No	No	
275	BALANCE OF PLANT INSTRUMENT	No	No	No	No	No	
276	LOCAL INSTRUMENT CONTROL PANEL	No	No	No	No	No	
277	NSSS AUXILIARY INSTRUMENT	No	No	No	No	No	
278	MAIN AND AUXILIARY CONTROL	No	No	No	No	No	
279	FIELD SERVICES FACILITY CONDUIT AND CABLE TRAYS	No	No	No	No	No	
280	CONDENSER TUBE CLEANING SYSTEM	No	No	No	No	No	
281	MAKEUP WATER TREATMENT PLANT ELECTRICAL EQUIPMENT	No	No	No	No	No	
282	FIELD SERVICES FACILITY ELECTRICAL EQUIPMENT	No	No	No	No	No	
283	LOW LEVEL RADWASTE FACILITY	No	No	No	No	No	
284	VOLUME REDUCTION SOLID WASTE FACILITY CONDUIT AND CABLE TRAYS	No	No	No	No	No	
285	SPARE CABLES	No	No	No	No	No	
286	SECURITY BACKUP POWER BUILDING	No	No	No	No	No	
287	ADDITIONAL DIESEL GENERATOR CONDUIT AND CABLE TRAYS	No	No	No	No	No	
288	500-KU SWITCHYARD CONDUIT	No	No	No	No	No	
289	HYPOCHLORITE BUILDING CONDUIT AND CABLE TRAY SYSTEMS	No	No	No	No	No	
290	CB CONDUIT AND CABLE TRAY	No	No	No	No	No	
291	IB CONDUIT AND CABLE TRAY	No	No	No	No	No	
292	AB CONDUIT AND CABLE TRAY	No	No	No	No	No	
293	RB CONDUIT AND CABLE TRAY	No	No	No	No	No	
294	DB CONDUIT AND CABLE TRAY	No	No	No	No	No	
295	SB CONDUIT AND CABLE TRAY	No	No	No	No	No	

Table 4.1-1 Systems Considered for the Seismic Equipment List

System Number	System	Mitigation Potential	In IE PRA	In IPEEE	In 2.3 SWEL	In Appendix R	Notes
296	DGB CONDUIT AND CABLE TRAY	No	No	No	No	No	
297	IPS CONDUIT AND CABLE TRAY	No	No	No	No	No	
298	CCW PS CONDUIT AND CABLE TRAY	No	No	No	No	No	
299	YARD CONDUIT AND CABLE TRAY	No	No	No	No	No	
300	MISCELLANEOUS ELECTRICAL EQUIPMENT	No	No	No	No	No	
301	COMPUTERS AND RECORDS	No	No	No	No	No	
302	PENETRATIONS AND SLEEVES (MECHANICAL AND ELECTRICAL)	No	No	No	No	No	
313	MANUAL DAMPERS	Yes	Yes	No	No	No	Included in AC system
500	ELECTRICAL PANELS	No	Yes	No	Yes	No	
928	MAKEUP WATER TREATMENT PLANT ELECTRICAL EQUIPMENT	No	No	No	No	No	
959	DEMINERALIZER WATER STORAGE & DISTRIBUTION SYSTEM FOR MAKEUP WATER TREATMENT PLA	No	Yes	No	No	No	System screened out
991	JUNCTION BOXES WITH TVA INDEX NUMBERS	No	No	No	No	No	
999	VARIOUS SYSTEMS	No	No	No	No	No	
XXX	ADDITIONAL POTENTIAL SIFF FIRE & FLOODING SOURCES	No	No	Yes	No	No	
XXX	STRUCTURES AND MISCELLANEOUS SSC	Yes	No	Yes	No	No	
XXX	MISCELLANEOUS PANELS	Yes	No	2 SSCs	4 SSCs	No	
XXX	FLEX SYSTEMS	Yes	No	No	No	No	

4.1.2 Relay and Breaker Evaluation

During a seismic event, vibratory ground motion can cause relays and breakers to chatter. The chattering of relays potentially can result in spurious signals to equipment. The chattering of breakers potentially can result in equipment either losing power or starting when it is not desired. Relay/breaker chatter can be acceptable (does not impact the associated equipment), self-correcting, or recovered by operator action. An extensive relay/breaker chatter evaluation was performed for both SQN units [24] in accordance with SPID, Section 6.4.2, and ASME/ANS PRA Standard, Section 5-2.2. The evaluation resulted in many relay/breaker chatter scenarios screened out from further evaluation based on no impact to component function. The 242 relays and breakers that were not screened out are listed in Table 4.1-2, along with their description, type, and disposition in the SPRA with appropriate seismic fragility or operator action.

Table 4.1-2 Summary of Disposition of Unscreened Relays and Breakers

UNID	Description	Relay/Breaker type	Disposition
SQN-1-FCV-001-0051	AFPT TRIP & THROTTLE VALVE	Allen Bradley 202	Modeled with Fragility Group SEIS_0-30-14
SQN-1-BCTA-003-0118-A	BREAKER FOR AUX FEEDWATER PUMP 1A-A	AS04/ASEA, BROWN BOVERI INC.	Modeled with Fragility Group SEIS_0-30-6
SQN-1-BCTA-003-0128-B	BREAKER FOR AUX FEEDWATER PUMP 1B-B	AS04/ASEA, BROWN BOVERI INC.	Modeled with Fragility Group SEIS_0-30-6
SQN-2-BCTA-003-0118-A	BREAKER FOR AUX FEEDWATER PUMP 2A-A	AS04/ASEA, BROWN BOVERI INC.	Modeled with Fragility Group SEIS_0-30-6
SQN-2-BCTA-003-0128-B	BREAKER FOR AUX FEEDWATER PUMP 2B-B	AS04/ASEA, BROWN BOVERI INC.	Modeled with Fragility Group SEIS_0-30-6
SQN-1-FCV-026-0240	HIGH PRESSURE FIRE PROTECTION CONTAINMENT ISOLATION VALVE	Westinghouse AR440R	Not Modeled. Fire protection containment isolation.
SQN-1-FCV-026-0243	HIGH PRESSURE FIRE PROTECTION CONTAINMENT ISOLATION VALVE	Westinghouse AR440R	Not Modeled. Fire protection containment isolation.
SQN-2-FCV-026-0240	HIGH PRESSURE FIRE PROTECTION CONTAINMENT ISOLATION VALVE	Westinghouse AR440R	Not Modeled. Fire protection containment isolation.
SQN-2-FCV-026-0243	HIGH PRESSURE FIRE PROTECTION CONTAINMENT ISOLATION VALVE	Westinghouse AR440R	Not Modeled. Fire protection containment isolation.
SQN-1-FCV-030-0046-A	CNTMT VAC RELF ISOL VLV	Westinghouse AR880AR	Modeled with Fragility Group SEIS_0-30-3-2
SQN-1-FCV-030-0047-A	CNTMT VAC RELF ISOL VLV	Westinghouse AR880AR	Modeled with Fragility Group SEIS_0-30-3-2
SQN-1-FCV-030-0048-A	CNTMT VAC RELF ISOL VLV	Westinghouse AR880AR	Modeled with Fragility Group SEIS_0-30-3-2
SQN-2-FCV-030-0046-A	CNTMT VAC RELF ISOL VLV	Westinghouse AR880AR	Modeled with Fragility Group SEIS_0-30-3-2
SQN-2-FCV-030-0047-A	CNTMT VAC RELF ISOL VLV	Westinghouse AR880AR	Modeled with Fragility Group SEIS_0-30-3-2
SQN-2-FCV-030-0048-A	CNTMT VAC RELF ISOL VLV	Westinghouse AR880AR	Modeled with Fragility Group SEIS_0-30-3-2
SQN-1-FCV-032-0080	CONTROL AIR CONTAINMENT ISOLATION MANUAL ISOLATION, Relay Id K-619	Westinghouse AR440R	Included in Containment Isolation Model.

Table 4.1-2 Summary of Disposition of Unscreened Relays and Breakers

UNID	Description	Relay/Breaker type	Disposition
SQN-1-FCV-032-0102	CONTROL AIR CONTAINMENT ISOLATION MANUAL ISOLATION, Relay Id K-619	Westinghouse AR440R	Included in Containment Isolation Model.
SQN-1-FCV-032-0110	CONTROL AIR CONTAINMENT ISOLATION MANUAL ISOLATION, Relay Id K-619	Westinghouse AR440R	Included in Containment Isolation Model.
SQN-2-FCV-032-0103	CONTROL AIR CONTAINMENT ISOLATION MANUAL ISOLATION, Relay Id K-619	Westinghouse AR440R	Included in Containment Isolation Model.
SQN-2-FCV-032-0111	CONTROL AIR CONTAINMENT ISOLATION MANUAL ISOLATION, Relay Id K-619	Westinghouse AR440R	Included in Containment Isolation Model.
SQN-2-FCV-032-0081	CONTROL AIR CONTAINMENT ISOLATION MANUAL ISOLATION, Relay Id K-619	Westinghouse AR440R	Included in Containment Isolation Model.
SQN-1-FCV-032-0080	CONTROL AIR CONTAINMENT ISOLATION MANUAL ISOLATION, Relay Id K-803	Potter & Burmfield Series MDR 66-4	Included in Containment Isolation Model.
SQN-1-FCV-032-0102	CONTROL AIR CONTAINMENT ISOLATION MANUAL ISOLATION, Relay Id K-803	Potter & Burmfield Series MDR 66-4	Included in Containment Isolation Model.
SQN-1-FCV-032-0110	CONTROL AIR CONTAINMENT ISOLATION MANUAL ISOLATION, Relay Id K-803	Potter & Burmfield Series MDR 66-4	Included in Containment Isolation Model.
SQN-2-FCV-032-0103	CONTROL AIR CONTAINMENT ISOLATION MANUAL ISOLATION, Relay Id K-803	Potter & Burmfield Series MDR 66-4	Included in Containment Isolation Model.
SQN-2-FCV-032-0111	CONTROL AIR CONTAINMENT ISOLATION MANUAL ISOLATION, Relay Id K-803	Potter & Burmfield Series MDR 66-4	Included in Containment Isolation Model.
SQN-2-FCV-032-0081	CONTROL AIR CONTAINMENT ISOLATION MANUAL ISOLATION, Relay Id K-803	Potter & Burmfield Series MDR 66-4	Included in Containment Isolation Model.
SQN-2-FCV-062-0063-A	SEAL FLOW ISO VLV, Relay Id K804	Potter & Brumfield Series MDR 66-4	Modeled with Fragility Group SEIS_0-30-2

Table 4.1-2 Summary of Disposition of Unscreened Relays and Breakers

UNID	Description	Relay/Breaker type	Disposition
SQN-1-FCV-062-0090	CHARGING FLOW ISO VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-1-FCV-062-0091-B	CHARGING FLOW ISO VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-2-FCV-062-0090-A	CHARGING FLOW ISO VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-2-FCV-062-0091-B	CHARGING FLOW ISO VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-1-LCV-062-0135-A	CHARGING PUMP FLOW RWST	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-1-LCV-062-0136-B	CHARGING PUMP FLOW RWST	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-2-LCV-062-0135-A	CHARGING PUMP FLOW RWST	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-2-LCV-062-0136-B	CHARGING PUMP FLOW RWST	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-1-FCV-062-0077	LETDOWN LINE ISO VLV FLOW CONTROL	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-2-FCV-062-0077-B	LETDOWN LINE ISO VLV FLOW CONTROL	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-1-FCV-062-0061-B	SEAL FLOW ISO VLV, Relay Id K614	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-1-FCV-062-0063-A	SEAL FLOW ISO VLV, Relay Id K614	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-2-FCV-062-0061-B	SEAL FLOW ISO VLV, Relay Id K804	Potter & Brumfield Series MDR 66-4	Modeled with Fragility Group SEIS_0-30-2
SQN-1-FCV-062-0061-B	SEAL FLOW ISO VLV, Relay Id K804	Potter & Brumfield Series MDR 66-4	Modeled with Fragility Group SEIS_0-30-2
SQN-1-FCV-062-0063-A	SEAL FLOW ISO VLV, Relay Id K804	Potter & Brumfield Series MDR 66-4	Modeled with Fragility Group SEIS_0-30-2
SQN-2-FCV-062-0061-B	SEAL FLOW ISO VLV, Relay Id K614	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-2-FCV-062-0063-A	SEAL FLOW ISO VLV, Relay Id K614	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1

Table 4.1-2 Summary of Disposition of Unscreened Relays and Breakers

UNID	Description	Relay/Breaker type	Disposition
SQN-1-BCTA-062-0108-A	CHARGING PUMP 1A-A BREAKER	7.5 HK ABB	Modeled with Fragility Group SEIS_0-30-6
SQN-1-BCTA-062-0104-B	CHARGING PUMP 1B-B BREAKER	AS04/ASEA, BROWN BOVERI INC.	Modeled with Fragility Group SEIS_0-30-6
SQN-2-BCTA-062-0104-B	CHARGING PUMP 2B-B BREAKER	AS04/ASEA, BROWN BOVERI INC.	Modeled with Fragility Group SEIS_0-30-6
SQN-2-BCTA-062-0108-A	CHARGING PUMP 2A-A BREAKER	AS04/ASEA, BROWN BOVERI INC.	Modeled with Fragility Group SEIS_0-30-6
SQN-1-FCV-063-0025-B	SIS CCP INJ TANK SHUTOFF VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-1-FCV-063-0026-A	SIS CCP INJ TANK SHUTOFF VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-1-FCV-063-0039-A	SIS CCP INJ TANK INLET SHUTOFF VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-1-FCV-063-0040-B	SIS CCP INJ TANK INLET SHUTOFF VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-2-FCV-063-0025-B	SIS CCP INJ TANK SHUTOFF VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-2-FCV-063-0026-A	SIS CCP INJ TANK SHUTOFF VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-2-FCV-063-0039-A	SIS CCP INJ TANK INLET SHUTOFF VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-2-FCV-063-0040-B	SIS CCP INJ TANK INLET SHUTOFF VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-1-BCTA-063-10-A	BREAKER FOR SAFETY INJECTION PUMP 1A-A	G182/GOULD INC/FORMERLY ITE IMPREIAL	Modeled with Fragility Group SEIS_0-30-6
SQN-1-BCTA-063-15-B	BREAKER FOR SAFETY INJECTION PUMP 1B-B	AS04/ASEA BROWN BOVERI	Modeled with Fragility Group SEIS_0-30-6
SQN-2-BCTA-063-10-A	BREAKER FOR SAFETY INJECTION PUMP 2A-A	ABB BREAKER 7.5 HK-500	Modeled with Fragility Group SEIS_0-30-6
SQN-2-BCTA-063-15-B	BREAKER FOR SAFETY INJECTION PUMP 2B-B	AS04/ASEA BROWN BOVERI	Modeled with Fragility Group SEIS_0-30-6
SQN-0-BCTA-067-0432-A	BREAKER FOR ERCW PUMP J-A	Siemens 8HKR-50-1200-130	Modeled with Fragility Group SEIS_0-30-6

Table 4.1-2 Summary of Disposition of Unscreened Relays and Breakers

UNID	Description	Relay/Breaker type	Disposition
SQN-0-BCTA-067-0436-A	BREAKER FOR ERCW PUMP K-A	G182/GOULD INC/FORMERLY ITE IMPERIAL	Modeled with Fragility Group SEIS_0-30-6
SQN-0-BCTA-067-0440-B	BREAKER FOR ERCW PUMP L-B	ABB 7.5HK	Modeled with Fragility Group SEIS_0-30-6
SQN-0-BCTA-067-0444-B	BREAKER FOR ERCW PUMP M-B	AS04/ASEA, BROWN BOVERI INC.	Modeled with Fragility Group SEIS_0-30-6
SQN-0-BCTA-067-0456-B	BREAKER FOR ERCW PUMP P-B	AS04/ASEA, BROWN BOVERI INC.	Modeled with Fragility Group SEIS_0-30-6
SQN-0-BCTA-067-0452-B	BREAKER FOR ERCW PUMP N-B	AS04/ASEA, BROWN BOVERI INC.	Modeled with Fragility Group SEIS_0-30-6
SQN-0-BCTA-067-0460-A	BREAKER FOR ERCW PUMP Q-A	G182/GOULD INC/FORMERLY ITE IMPERIAL	Modeled with Fragility Group SEIS_0-30-6
SQN-0-BCTA-067-0464-A	BREAKER FOR ERCW PUMP K-A	AS04/ASEA, BROWN BOVERI INC.	Modeled with Fragility Group SEIS_0-30-6
SQN-1-FCV-070-0087-B	RC PMP THRM BARR RET CONTMNT ISOL VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-1-FCV-070-0089-B	RC PMP OIL CLR RET CONTMNT ISOL VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-1-FCV-070-0090-A	RC PMP THRM BARR RET CONTMNT ISOL VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-1-FCV-070-0092-A	RC PMP OIL CLR RET CONTMNT ISOL VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-1-FCV-070-0133-A	RC PMP THRM BAR CONT ISOL VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-1-FCV-070-0134-B	RC PMP THRM BAR CONT ISOL VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-1-FCV-070-0140-B	RC PMP OIL CLR HDR CONT ISOL VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-2-FCV-070-0087-B	RC PMP THRM BARR RET CONTMNT ISOL VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-2-FCV-070-0089-B	RC PMP OIL CLR RET CONTMNT ISOL VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-2-FCV-070-0090-A	RC PMP THRM BARR RET CONTMNT ISOL VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1

Table 4.1-2 Summary of Disposition of Unscreened Relays and Breakers

UNID	Description	Relay/Breaker type	Disposition
SQN-2-FCV-070-0092-A	RC PMP OIL CLR RET CONTMNT ISOL VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-2-FCV-070-0133-A	RC PMP THRM BAR CONT ISOL VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-2-FCV-070-0134-B	RC PMP THRM BAR CONT ISOL VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-2-FCV-070-0139-A	RC PMP OIL CLR HDR CONT ISOL VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-2-FCV-070-0140-B	RC PMP OIL CLR HDR CONT ISOL VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-1-FCV-070-0139-A	RC PMP OIL CLR HDR CONT ISOL VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-1
SQN-1-FCV-070-0143	CCS TO EXCESS LETDOWN HX	Westinghouse AR440R	Not Modelled. No credit taken for excess letdown.
SQN-2-FCV-070-0143	CCS TO EXCESS LETDOWN HX	Westinghouse AR440R	Not Modelled. No credit taken for excess letdown.
SQN-1-FCV-070-0141-A	RC PUMP OIL CLR HDR CONT ISOL VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-3-1
SQN-2-FCV-070-0141-A	RC PUMP OIL CLR HDR CONT ISOL VLV	Westinghouse AR440R	Modeled with Fragility Group SEIS_0-30-3-1
SQN-1-BCTA-074-10-A	RHR PUMP 1A-A breaker	G182/GOULD INC/FORMERLY ITE IMPERIAL	Modeled with Fragility Group SEIS_0-30-6
SQN-1-BCTA-074-20-B	RHR PUMP 1B-B breaker	G182/GOULD INC/FORMERLY ITE IMPERIAL	Modeled with Fragility Group SEIS_0-30-6
SQN-2-BCTA-074-10-B	RHR PUMP 2A-A breaker	G182/GOULD INC/FORMERLY ITE IMPERIAL	Modeled with Fragility Group SEIS_0-30-6
SQN-2-BCTA-074-20-B	RHR PUMP 2B-B breaker	A997/ABB	Modeled with Fragility Group SEIS_0-30-6
SQN-1-GENB-082-0001A-A	DG 1A-A, Relay Id SDR	Square D relay, Class 8501, Type KPD13V63, Series D	Modeled with Fragility Group SEIS_0-30-10
SQN-1-GENB-082-0001B-B	DG 1B-B, Relay Id SDR	Square D relay, Class 8501, Type KPD13V63, Series D	Modeled with Fragility Group SEIS_0-30-10
SQN-2-GENB-082-0002A-A	DG 2A-A, Relay Id SDR	Square D relay, Class 8501, Type KPD13V63, Series D	Modeled with Fragility Group SEIS_0-30-10

Table 4.1-2 Summary of Disposition of Unscreened Relays and Breakers

UNID	Description	Relay/Breaker type	Disposition
SQN-2-GENB-082-0002B-B	DG 2B-B, Relay Id SDR	Square D relay, Class 8501, Type KPD13V63, Series D	Modeled with Fragility Group SEIS_0-30-10
SQN-1-GENB-082-0001A-A	DG 1A-A,Relay Id R2	Square D relay, Class 8501, Type KPD13V63, Series D	Modeled with Fragility Group SEIS_0-30-10
SQN-1-GENB-082-0001B-B	DG 1B-B, Relay Id R2	Square D relay, Class 8501, Type KPD13V63, Series D	Modeled with Fragility Group SEIS_0-30-10
SQN-2-GENB-082-0002A-A	DG 2A-A, Relay Id R2	Square D relay, Class 8501, Type KPD13V63, Series D	Modeled with Fragility Group SEIS_0-30-10
SQN-2-GENB-082-0002B-B	DG 2B-B, Relay Id R2	Square D relay, Class 8501, Type KPD13V63, Series D	Modeled with Fragility Group SEIS_0-30-10
SQN-1-GENB-082-0001A-A	DG 1A-A, Relay Id HJWTS	Square D relay, Class 8501, Type KPD13V63, Series D	Modeled with Fragility Group SEIS_0-30-10
SQN-1-GENB-082-0001B-B	DG 1B-B, Relay Id HJWTS	Square D relay, Class 8501, Type KPD13V63, Series D	Modeled with Fragility Group SEIS_0-30-10
SQN-2-GENB-082-0002A-A	DG 2A-A, Relay Id HJWTS	Square D relay, Class 8501, Type KPD13V63, Series D	Modeled with Fragility Group SEIS_0-30-10
SQN-2-GENB-082-0002B-B	DG 2B-B, Relay Id HJWTS	Square D relay, Class 8501, Type KPD13V63, Series D	Modeled with Fragility Group SEIS_0-30-10
SQN-1-GENB-082-0001A-A	DG 1A-A, Relay Id CP	Square D relay, Class 8501, Type KPD13V63, Series D	Modeled with Fragility Group SEIS_0-30-10
SQN-1-GENB-082-0001B-B	DG 1B-B, Relay Id CP	Square D rely, Class 8501, Type KPD13V63, Series D	Modeled with Fragility Group SEIS_0-30-10
SQN-2-GENB-082-0002A-A	DG 2A-A, Relay Id CP	Square D rely, Class 8501, Type KPD13V63, Series D	Modeled with Fragility Group SEIS_0-30-10
SQN-2-GENB-082-0002B-B	DG 2B-B, Relay Id CP	Square D relay, Class 8501, Type KPD13V63, Series D	Modeled with Fragility Group SEIS_0-30-10
SQN-1-GENB-082-0001A-A	DG 1A-A, Relay Id K4	Kraus & Naimer relay Type R254	Modeled with Fragility Group SEIS_0-30-11
SQN-1-GENB-082-0001B-B	DG 1B-B, Relay Id K4	Kraus & Naimer relay Type R254	Modeled with Fragility Group SEIS_0-30-11
SQN-2-GENB-082-0002A-A	DG 2A-A, Relay Id K4	Kraus & Naimer relay Type R254	Modeled with Fragility Group SEIS_0-30-11
SQN-2-GENB-082-0002B-B	DG 2B-B, Relay Id K4	Kraus & Naimer relay Type R254	Modeled with Fragility Group SEIS_0-30-11

Table 4.1-2 Summary of Disposition of Unscreened Relays and Breakers

UNID	Description	Relay/Breaker type	Disposition
SQN-1-GENB-082-0001A-A	DG 1A-A, Relay Id 86GA	Electro Switch Relay, Type 7803E	Modeled with Fragility Group SEIS_0-30-12
SQN-2-GENB-082-0002B-B	DG 2B-B, Relay Id 86GA	Electro Switch Relay, Type 7803E	Modeled with Fragility Group SEIS_0-30-12
SQN-1-GENB-082-0001B-B	DG 1B-B, Relay Id 86GA	Electro Switch Relay, Type 7803E	Modeled with Fragility Group SEIS_0-30-12
SQN-2-GENB-082-0002A-A	DG 2A-A, Relay Id 86GA	Electro Switch Relay, Type 7803E	Modeled with Fragility Group SEIS_0-30-12
SQN-1-GENB-082-0001A-A	DG 1A-A, Relay Id ESX21A	General Electric (GE) HGA Relay	Modeled with Fragility Group SEIS_0-30-13
SQN-1-GENB-082-0001B-B	DG 1B-B, Relay Id ESX21A	GE HGA Relay	Modeled with Fragility Group SEIS_0-30-13
SQN-2-GENB-082-0002B-B	DG 2B-B, Relay Id ESX21A	GE HGA Relay	Modeled with Fragility Group SEIS_0-30-13
SQN-2-GENB-082-0002A-A	DG 2A-A, Relay Id ESX21A	GE HGA Relay	Modeled with Fragility Group SEIS_0-30-13
SQN-1-GENB-082-0001A-A	DG 1A-A, Relay Id TD1	Agastat Model 7012PD	Modeled with Fragility Group SEIS_0-30-15
SQN-1-GENB-082-0001B-B	DG 1B-B, Relay Id TD1	Agastat Model 7012PD	Modeled with Fragility Group SEIS_0-30-15
SQN-2-GENB-082-0002A-A	DG 2A-A, Relay Id TD1	Agastat Model 7012PD	Modeled with Fragility Group SEIS_0-30-15
SQN-2-GENB-082-0002B-B	DG 2B-B, Relay Id TD1	Agastat Model 7012PD	Modeled with Fragility Group SEIS_0-30-15
SQN-1-BCTB-201-DJ /1B-A	NORMAL SUPPLY BKR FOR 480V SHTDN BD 1A1-A, Relay Id 86N	Type WL lock-out relay	Modeled with Fragility Group SEIS_0-30-8
SQN-1-BCTB-201-DK /1B-A	NORMAL SUPPLY BKR FOR 480V SHDN BD 1A2-A, Relay Id 86N	Type WL lock-out relay	Modeled with Fragility Group SEIS_0-30-8
SQN-1-BCTB-201-DL /1B-B	NORMAL SUPPLY BKR FOR 480V SHTDN BD 1B1-B, Relay Id 86N	Type WL lock-out relay	Modeled with Fragility Group SEIS_0-30-8
SQN-1-BCTB-201-DM /1B-B	NORMAL SUPPLY BKR FOR 480V SHTDN BD 1B2-B, Relay Id 86N	Type WL lock-out relay	Modeled with Fragility Group SEIS_0-30-8
SQN-2-BCTB-201-DN /1B-A	NORMAL SPPLY BKR FOR 480V SHTDN BD 2A1-A, Relay Id 86N	Type WL lock-out relay	Modeled with Fragility Group SEIS_0-30-8

Table 4.1-2 Summary of Disposition of Unscreened Relays and Breakers

UNID	Description	Relay/Breaker type	Disposition
SQN-2-BCTB-201-DO /1B-A	NORMAL SUPPLY BKR FOR 480V SHTDN BD 2A2-A, Relay Id 86N	Type WL lock-out relay	Modeled with Fragility Group SEIS_0-30-8
SQN-2-BCTB-201-DP /1B-B	NORMAL SUPPLY BKR FOR 480V SHTDN BD 2B1-B, Relay Id 86N	Type WL lock-out relay	Modeled with Fragility Group SEIS_0-30-8
SQN-2-BCTB-201-DQ /1B-B	NORMAL SUPPLY BKR FOR 480V SHTDN BD 2B2-B, Relay Id 86N	Type WL lock out relay	Modeled with Fragility Group SEIS_0-30-8
SQN-1-BCTB-201-DJ /9B-A	NOR FDR RVENT BD 1A-A	Breaker, Westinghouse Type DS	Modeled with Fragility Group SEIS_0-30-9
SQN-1-BCTB-201-DJ /1B-A	NORMAL SUPPLY BKR FOR 480V SHTDN BD 1A1-A	Breaker, Westinghouse Type DS	Modeled with Fragility Group SEIS_0-30-9
SQN-1-BCTB-201-DK /10B-A	NOR FDR C&A VENT BD 1A2-A	Breaker, Westinghouse Type DS	Modeled with Fragility Group SEIS_0-30-9
SQN-1-BCTB-201-DK /1B-A	NORMAL SUPPLY BKR FOR 480V SHDN BD 1A2-A	Breaker, Westinghouse Type DS	Modeled with Fragility Group SEIS_0-30-9
SQN-1-BCTB-201-DL /9B-B	NOR FDR RVENT BD 1B-B	Breaker, Westinghouse Type DS	Modeled with Fragility Group SEIS_0-30-9
SQN-1-BCTB-201-DL /1B-B	NORMAL SUPPLY BKR FOR 480V SHTDN BD 1B1-B	Breaker, Westinghouse Type DS	Modeled with Fragility Group SEIS_0-30-9
SQN-1-BCTB-201-DM /1B-B	NORMAL SUPPLY BKR FOR 480V SHTDN BD 1B2-B	Breaker, Westinghouse Type DS	Modeled with Fragility Group SEIS_0-30-9
SQN-1-BCTB-201-DM /10B-B	NOR FDR C&A VENT BD 1B2-B	Breaker, Westinghouse Type DS	Modeled with Fragility Group SEIS_0-30-9
SQN-2-BCTB-201-DN /1B-A	NORMAL SPPLY BKR FOR4 80V SHTDN BD 2A1-A	Breaker, Westinghouse Type DS	Modeled with Fragility Group SEIS_0-30-9
SQN-2-BCTB-201-DN /9B-A	NOR FDR RVENT BD 2A-A	Breaker, Westinghouse Type DS	Modeled with Fragility Group SEIS_0-30-9
SQN-2-BCTB-201-DO /1B-A	NORMAL SUPPLY BKR FOR 480V SHTDN BD 2A2-A	Breaker, Westinghouse Type DS	Modeled with Fragility Group SEIS_0-30-9
SQN-2-BCTB-201-DO /10B-A	NOR FDR C&A VENT BD 2A2-A	Breaker, Westinghouse Type DS	Modeled with Fragility Group SEIS_0-30-9
SQN-2-BCTB-201-DP /1B-B	NORMAL SUPPLY BKR FOR 480V SHTDN BD 2B1-B	Breaker, Westinghouse Type DS	Modeled with Fragility Group SEIS_0-30-9
SQN-2-BCTB-201-DP /9B-B	NOR FDR RVENT BD 2B-B	Breaker, Westinghouse Type DS	Modeled with Fragility Group SEIS_0-30-9

Table 4.1-2 Summary of Disposition of Unscreened Relays and Breakers

UNID	Description	Relay/Breaker type	Disposition
SQN-2-BCTB-201-DQ /1B-B	NORMAL SUPPLY BKR FOR 480V SHTDN BD 2B2-B	Breaker, Westinghouse Type DS	Modeled with Fragility Group SEIS_0-30-9
SQN-2-BCTB-201-DQ /10B-B	NOR FDR C&A VENT BD 2B2-B	Breaker, Westinghouse Type DS	Modeled with Fragility Group SEIS_0-30-9
SQN-1-BCTB-201-DJ /9B-A	NOR FDR RVENT BD 1A-A , Relay Id RV UVX	Westinghouse Type SG 116	Modeled with Fragility Group SEIS_0-30-7
SQN-1-BCTB-201-DK /10B-A	NOR FDR C&A VENT BD 1A2-A, Relay Id RV UVX	Westinghouse Type SG 116	Modeled with Fragility Group SEIS_0-30-7
SQN-1-BCTB-201-DL /9B-B	NOR FDR RVENT BD 1B-B, Relay Id RV UVX	Westinghouse Type SG 116	Modeled with Fragility Group SEIS_0-30-7
SQN-1-BCTB-201-DM /10B-B	NOR FDR C&A VENT BD 1B2-B, Relay Id RV UVX	Westinghouse Type SG 116	Modeled with Fragility Group SEIS_0-30-7
SQN-2-BCTB-201-DN /9B-A	NOR FDR RVENT BD 2A-A, Relay Id RV UVX	Westinghouse Type SG 116	Modeled with Fragility Group SEIS_0-30-7
SQN-2-BCTB-201-DO /10B-A	NOR FDR C&A VENT BD 2A2-A, Relay Id RV UVX	Westinghouse Type SG 116	Modeled with Fragility Group SEIS_0-30-7
SQN-2-BCTB-201-DP /9B-B	NOR FDR RVENT BD 2B-B, Relay Id RV UVX	Westinghouse Type SG 116	Modeled with Fragility Group SEIS_0-30-7
SQN-2-BCTB-201-DQ /10B-B	NOR FDR C&A VENT BD 2B2-B, Relay Id RV UVX	Westinghouse Type SG 116	Modeled with Fragility Group SEIS_0-30-7
SQN-1-BCTB-201-DJ /9B-A	NOR FDR RVENT BD 1A-A, Relay Id RV UVY	Westinghouse Type SG 116	Modeled with Fragility Group SEIS_0-30-7
SQN-1-BCTB-201-DK /10B-A	NOR FDR C&A VENT BD 1A2-A, Relay Id RV UVY	Westinghouse Type SG 116	Modeled with Fragility Group SEIS_0-30-7
SQN-1-BCTB-201-DL /9B-B	NOR FDR RVENT BD 1B-B, Relay Id RV UVY	Westinghouse Type SG 116	Modeled with Fragility Group SEIS_0-30-7
SQN-1-BCTB-201-DM /10B-B	NOR FDR C&A VENT BD 1B2-B, Relay Id RV UVY	Westinghouse Type SG 116	Modeled with Fragility Group SEIS_0-30-7
SQN-2-BCTB-201-DN /9B-A	NOR FDR RVENT BD 2A-A, Relay Id RV UVY	Westinghouse Type SG 116	Modeled with Fragility Group SEIS_0-30-7
SQN-2-BCTB-201-DO /10B-A	NOR FDR C&A VENT BD 2A2-A, Relay Id RV UVY	Westinghouse Type SG 116	Modeled with Fragility Group SEIS_0-30-7
SQN-2-BCTB-201-DP /9B-B	NOR FDR RVENT BD 2B-B, Relay Id RV UVY	Westinghouse Type SG 116	Modeled with Fragility Group SEIS_0-30-7

Table 4.1-2 Summary of Disposition of Unscreened Relays and Breakers

UNID	Description	Relay/Breaker type	Disposition
SQN-2-BCTB-201-DQ /10B-B	NOR FDR C&A VENT BD 2B2-B, Relay Id RV UVY	Westinghouse Type SG 116	Modeled with Fragility Group SEIS_0-30-7
SQN-1-BCTA-202-CM /6-A	EMERGENCY SUPPLY BKR 1912 FOR 6.9KV SHUTDOWN BOARD 1A-A, Relay Id 86-718	GE 12HEA61B235-X2	Modeled with Fragility Group SEIS_0-30-5
SQN-1-BCTA-202-CM /6-A	EMERGENCY SUPPLY BKR 1912 FOR 6.9KV SHUTDOWN BOARD 1A-A, Relay Id 86-716	GE 12HEA61B235-X2	Modeled with Fragility Group SEIS_0-30-5
SQN-1-BCTA-202-CN /6-B	EMERGENCY SUPPLY BKR 1914 FOR 6.9KV SHUTDOWN BOARD 1B-B, Relay Id 86-728	GE 12HEA61B235-X2	Modeled with Fragility Group SEIS_0-30-5
SQN-1-BCTA-202-CN /6-B	EMERGENCY SUPPLY BKR 1914 FOR 6.9KV SHUTDOWN BOARD 1B-B, Relay Id 86-726	GE 12HEA61B235-X2	Modeled with Fragility Group SEIS_0-30-5
SQN-2-BCTA-202-CO /6-A	EMERGENCY SUPPLY BKR 1922 FOR 6.9KV SHUTDOWN BOARD 2A-A, Relay Id 86-818	GE 12HEA61B235-X2	Modeled with Fragility Group SEIS_0-30-5
SQN-2-BCTA-202-CO /6-A	EMERGENCY SUPPLY BKR 1922 FOR 6.9KV SHUTDOWN BOARD 2A-A, Relay Id 86-816	GE 12HEA61B235-X2	Modeled with Fragility Group SEIS_0-30-5
SQN-2-BCTA-202-CP /6-B	EMERGENCY SUPPLY BKR 1924 FOR 6.9KV SHUTDOWN BOARD 2B-B, Relay Id 86-828	GE 12HEA61B235-X2	Modeled with Fragility Group SEIS_0-30-5
SQN-2-BCTA-202-CP /6-B	EMERGENCY SUPPLY BKR 1924 FOR 6.9KV SHUTDOWN BOARD 2B-B, Relay Id 86-826	GE 12HEA61B235-X2	Modeled with Fragility Group SEIS_0-30-5
SQN-1-BCTA-202-CM /6-A	EMERGENCY SUPPLY BKR 1912 FOR 6.9KV SHUTDOWN BOARD 1A-A, Relay Id 86S1A	GE 12HEA61C240-X2	Modeled with Fragility Group SEIS_0-30-5
SQN-1-BCTA-202-CM /3-A	480V SHTDN TRANS 1A1-A FDR 45N765-2, Relay Id 86S1A	GE HEA	Modeled with Fragility Group SEIS_0-30-5
SQN-1-BCTA-202-CM /4-A	480V SD TRANS 1A2-A 45N765-2, Relay Id 86S1A	GE HEA	Modeled with Fragility Group SEIS_0-30-5
SQN-1-BCTA-202-CM /5-A	480V SHTDN TRANS 1A-A FDR 45N765-2, Relay Id 86S1A	GE HEA	Modeled with Fragility Group SEIS_0-30-5

Table 4.1-2 Summary of Disposition of Unscreened Relays and Breakers

UNID	Description	Relay/Breaker type	Disposition
SQN-1-BCTA-202-CN /3-B	480V SHTDN TRANS 1B1-B FDR 45N765-2, Relay Id 86S1A	GE HEA	Modeled with Fragility Group SEIS_0-30-5
SQN-1-BCTA-202-CN /4-B	480V SHTDN TRANS 1B2-B FDR 45N765-2, Relay Id 86S1A	GE HEA	Modeled with Fragility Group SEIS_0-30-5
SQN-1-BCTA-202-CN /5-B	480V SHTDN TRANS 1B-B FDR, Relay Id 86S1A	GE HEA	Modeled with Fragility Group SEIS_0-30-5
SQN-2-BCTA-202-CO /3-A	480V SHTDN TRANS 2A1-A NORMAL 45N765-2, Relay Id 86S1A	GE HEA	Modeled with Fragility Group SEIS_0-30-5
SQN-2-BCTA-202-CO /4-A	480V SHUTDOWN BD 2A2-A NORMAL 45N765-2, Relay Id 86S1A	GE HEA	Modeled with Fragility Group SEIS_0-30-5
SQN-2-BCTA-202-CO /5-A	480V SHTDN TRANS 2A-A ALT FOR 2A2-A & 2A1-A, Relay Id 86S1A	GE HEA	Modeled with Fragility Group SEIS_0-30-5
SQN-2-BCTA-202-CP /4-B	480V SHTDN TRANS 2B2-B 45N765-2 NORM, Relay Id 86S1A	GE HEA	Modeled with Fragility Group SEIS_0-30-5
SQN-2-BCTA-202-CP /5-B	480V SHTDN BD TRANS ALT FOR 2B2-B & 2B1-B, Relay Id 86S1A	GE HEA	Modeled with Fragility Group SEIS_0-30-5
SQN-2-BCTA-202-CP /3-B	480V SHTDN TRANS 2B1-B NORMAL 45N765-2, Relay Id 86S1A	GE HEA	Modeled with Fragility Group SEIS_0-30-5
SQN-1-BCTA-202-CN /6-B	EMERGENCY SUPPLY BKR 1914 FOR 6.9KV SHUTDOWN BOARD 1B-B, Relay Id 86S1A	GE 12HEA61C240-X2	Modeled with Fragility Group SEIS_0-30-5
SQN-2-BCTA-202-CO /6-A	EMERGENCY SUPPLY BKR 1922 FOR 6.9KV SHUTDOWN BOARD 2A-A, Relay Id 86S2A	GE 12HEA61C240-X2	Modeled with Fragility Group SEIS_0-30-5
SQN-2-BCTA-202-CP /6-B	EMERGENCY SUPPLY BKR 1924 FOR 6.9KV SHUTDOWN BOARD 2B-B, Relay Id 86S2A	GE 12HEA61C240-X2	Modeled with Fragility Group SEIS_0-30-5
SQN-1-BCTA-202-CM /6-A	EMERGENCY SUPPLY BKR 1912 FOR 6.9KV SHUTDOWN BOARD 1A-A, Relay Id 51	Westinghouse type CO	Modeled with Fragility Group SEIS_0-30-6
SQN-1-BCTA-202-CN /6-B	EMERGENCY SUPPLY BKR 1914 FOR 6.9KV SHUTDOWN BOARD 1B-B, Relay Id 51	Westinghouse type CO	Modeled with Fragility Group SEIS_0-30-6

Table 4.1-2 Summary of Disposition of Unscreened Relays and Breakers

UNID	Description	Relay/Breaker type	Disposition
SQN-2-BCTA-202-CO /6-A	EMERGENCY SUPPLY BKR 1922 FOR 6.9KV SHUTDOWN BOARD 2A-A, Relay Id 51	Westinghouse type CO	Modeled with Fragility Group SEIS_0-30-6
SQN-2-BCTA-202-CP /6-B	EMERGENCY SUPPLY BKR 1924 FOR 6.9KV SHUTDOWN BOARD 2B-B, Relay Id 51	Westinghouse type CO	Modeled with Fragility Group SEIS_0-30-6
SQN-1-BCTA-202-CM /22-A	480V XFMR 1A-A TO ERCW PUMP STATION, Relay Id 86S1A	GE HEA	Not modeled. Loads of 480 V ERCW Board not credited in the model.
SQN-1-BCTA-202-CN /22-B	480V XFMR 1B-B TO ERCW PUMP STATION, Relay Id 86S1A	GE HEA	Not modeled. Loads of 480 V ERCW Board not credited in the model.
SQN-2-BCTA-202-CO /22-A	480V XFMR 2A-A TO ERCW PUMP STATION, Relay Id 86S1A	GE HEA	Not modeled. Loads of 480 V ERCW Board not credited in the model.
SQN-2-BCTA-202-CP /22-B	480V XFMR 2B-B TO ERCW PUMP STATION, Relay Id 86S1A	GE HEA	Not modeled. Loads of 480 V ERCW Board not credited in the model.
SQN-1-BCTA-202-CM /6-A	Emergency Supply bkr 1912 for 6.9KV Shutdown board 1A-A	AS04/ASEA, BROWN BOVERI INC.	Modeled with Fragility Group SEIS_0-30-6
SQN-1-BCTA-202-CN /6-B	Emergency Supply bkr 1914 for 6.9KV Shutdown board 1B-B	AS04/ASEA, BROWN BOVERI INC.	Modeled with Fragility Group SEIS_0-30-6
SQN-2-BCTA-202-CO /6-A	Emergency Supply bkr 1922 for 6.9KV Shutdown board 2A-A	AS04/ASEA, BROWN BOVERI INC.	Modeled with Fragility Group SEIS_0-30-6
SQN-1-BCTA-202-CM /3-A	480V SHTDN TRANS 1A1-A FDR 45N765-2	AS04/ASEA, BROWN BOVERI INC.	Modeled with Fragility Group SEIS_0-30-6
SQN-1-BCTA-202-CM /4-A	480V SD TRANS 1A2-A 45N765-2	AS04/ASEA, BROWN BOVERI INC.	Modeled with Fragility Group SEIS_0-30-6
SQN-1-BCTA-202-CM /5-A	480V SHTDN TRANS 1A-A FDR 45N765-2	AS04/ASEA, BROWN BOVERI INC.	Modeled with Fragility Group SEIS_0-30-6
SQN-1-BCTA-202-CN /3-B	480V SHTDN TRANS 1B1-B FDR 45N765-2	AS04/ASEA, BROWN BOVERI INC.	Modeled with Fragility Group SEIS_0-30-6
SQN-1-BCTA-202-CN /4-B	480V SHTDN TRANS 1B2-B FDR 45N765-2	AS04/ASEA, BROWN BOVERI INC.	Modeled with Fragility Group SEIS_0-30-6

Table 4.1-2 Summary of Disposition of Unscreened Relays and Breakers

UNID	Description	Relay/Breaker type	Disposition
SQN-1-BCTA-202-CN /5-B	480V SHTDN TRANS 1B-B FDR	AS04/ASEA, BROWN BOVERI INC.	Modeled with Fragility Group SEIS_0-30-6
SQN-2-BCTA-202-CO /3-A	480V SHTDN TRANS 2A1-A NORMAL 45N765-2	AS04/ASEA, BROWN BOVERI INC.	Modeled with Fragility Group SEIS_0-30-6
SQN-2-BCTA-202-CO /4-A	480V SHUTDOWN BD 2A2-A NORMAL 45N765-2	AS04/ASEA, BROWN BOVERI INC.	Modeled with Fragility Group SEIS_0-30-6
SQN-2-BCTA-202-CP /4-B	480V SHTDN TRANS 2B2-B 45N765-2 NORM	AS04/ASEA, BROWN BOVERI INC.	Modeled with Fragility Group SEIS_0-30-6
SQN-2-BCTA-202-CP /5-B	480V SHTDN BD TRANS ALT FOR 2B2-B & 2B1-B	AS04/ASEA, BROWN BOVERI INC.	Modeled with Fragility Group SEIS_0-30-6
SQN-2-BCTA-202-CP /3-B	480V SHTDN TRANS 2B1-B NORMAL 45N765-2	AS04/ASEA, BROWN BOVERI INC.	Modeled with Fragility Group SEIS_0-30-6
SQN-2-BCTA-202-CO /5-A	480V SHTDN TRANS 2A-A ALT FOR 2A2-A & 2A1-A	G182/GOULD INC/FORMERLY ITE IMPERIAL	Modeled with Fragility Group SEIS_0-30-6
SQN-2-BCTA-202-CP /6-B	EMERGENCY SUPPLY BKR 1924 FOR 6.9KV SHUTDOWN BOARD 2B-B	G182/GOULD INC/FORMERLY ITE IMPERIAL	Modeled with Fragility Group SEIS_0-30-6
SQN-2-BCTA-202-CO /22-A	480V XFMR 2A-A TO ERCW PUMP STATION	ABB 7.5 HK-500	Not modeled. Loads of 480 V ERCW Board not credited in the model.
SQN-1-BCTA-202-CM /22-A	480V XFMR 1A-A TO ERCW PUMP STATION	AS04/ASEA, BROWN BOVERI INC.	Not modeled. Loads of 480 V ERCW Board not credited in the model.
SQN-1-BCTA-202-CN /22-B	480V XFMR 1B-B TO ERCW PUMP STATION	AS04/ASEA, BROWN BOVERI INC.	Not modeled. Loads of 480 V ERCW Board not credited in the model.
SQN-2-BCTA-202-CP /22-B	480V XFMR 2B-B TO ERCW PUMP STATION	AS04/ASEA, BROWN BOVERI INC.	Not modeled. Loads of 480 V ERCW Board not credited in the model.
SQN-1-FCV-313-0222	INCORE INSTRUMENT ROOM CHILL WATER	Westinghouse AR440R	Not Modeled. Incore Instrument Room chill water not included.
SQN-1-FCV-313-0223	INCORE INSTRUMENT ROOM CHILL WATER	Westinghouse AR440R	Not Modeled. Incore Instrument Room chill water not included.
SQN-1-FCV-313-0224	INCORE INSTRUMENT ROOM CHILL WATER	Westinghouse AR440R	Not Modeled. Incore Instrument Room chill water not included.

Table 4.1-2 Summary of Disposition of Unscreened Relays and Breakers

UNID	Description	Relay/Breaker type	Disposition
SQN-1-FCV-313-0225	INCORE INSTRUMENT ROOM CHILL WATER	Westinghouse AR440R	Not Modeled. Incore Instrument Room chill water not included.
SQN-1-FCV-313-0229	INCORE INSTRUMENT ROOM CHILL WATER	Westinghouse AR440R	Not Modeled. Incore Instrument Room chill water not included.
SQN-1-FCV-313-0230	INCORE INSTRUMENT ROOM CHILL WATER	Westinghouse AR440R	Not Modeled. Incore Instrument Room chill water not included.
SQN-1-FCV-313-0231	INCORE INSTRUMENT ROOM CHILL WATER	Westinghouse AR440R	Not Modeled. Incore Instrument Room chill water not included.
SQN-1-FCV-313-0232	INCORE INSTRUMENT ROOM CHILL WATER	Westinghouse AR440R	Not Modeled. Incore Instrument Room chill water not included.
SQN-2-FCV-313-0222	INCORE INSTRUMENT ROOM CHILL WATER	Westinghouse AR440R	Not Modeled. Incore Instrument Room chill water not included.
SQN-2-FCV-313-0223	INCORE INSTRUMENT ROOM CHILL WATER	Westinghouse AR440R	Not Modeled. Incore Instrument Room chill water not included.
SQN-2-FCV-313-0224	INCORE INSTRUMENT ROOM CHILL WATER	Westinghouse AR440R	Not Modeled. Incore Instrument Room chill water not included.
SQN-2-FCV-313-0225	INCORE INSTRUMENT ROOM CHILL WATER	Westinghouse AR440R	Not Modeled. Incore Instrument Room chill water not included.
SQN-2-FCV-313-0229	INCORE INSTRUMENT ROOM CHILL WATER	Westinghouse AR440R	Not Modeled. Incore Instrument Room chill water not included.
SQN-2-FCV-313-0230	INCORE INSTRUMENT ROOM CHILL WATER	Westinghouse AR440R	Not Modeled. Incore Instrument Room chill water not included.
SQN-2-FCV-313-0231	INCORE INSTRUMENT ROOM CHILL WATER	Westinghouse AR440R	Not Modeled. Incore Instrument Room chill water not included.
SQN-2-FCV-313-0232	INCORE INSTRUMENT ROOM CHILL WATER	Westinghouse AR440R	Not Modeled. Incore Instrument Room chill water not included.

4.2 Walkdown Approach

This section provides a summary of the methodology and scope of the seismic walkdowns performed for the SPRA. Walkdowns were performed by personnel with appropriate qualifications as defined in the SPID [3]. Walkdowns of those SSCs included on the SEL were performed as part of the development of the SEL to assess the as-installed condition of these SSCs for use in determining their seismic capacity and performing initial screening.

Walkdowns were performed in accordance with guidance in the SPID, Section 6.5, and the associated requirements in the ASME/ANS PRA Standard [8].

Several previous seismic walkdowns for SQN have been documented. The information gathered during these previous walkdowns and the results and conclusions contained in the walkdown information were used, where applicable, to supplement plant drawings and calculations. These previous walkdowns include:

- IPEEE – Performed in 1995 using the guidelines contained in EPRI NP-6041-SL [26].
- NTTF 2.3 Seismic – Performed in 2012 in response to NTTF Recommendation 2.3 Seismic, to identify and address degraded, nonconforming, or unanalyzed conditions, and to verify the current plant configuration with the current seismic licensing basis.
- Expedited Seismic Evaluation Process (ESEP) – Performed in 2013 to focus the initial industry efforts on short-term evaluations 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, including FLEX equipment installations. This included walkdowns and calculations to demonstrate that the high confidence low probability of failure (HCLPF) seismic capacity for the ESEP subset of plant equipment exceeded the Review Level Ground Motion (RLGM). The RLGM was set to 2xSSE (i.e., 0.36g) for this purpose.

Information from these walkdowns was gathered and reviewed to obtain inputs and insights for the development of component fragilities. To ensure that the information remained valid and to include components that had not been previously walked down, all components on the SEL, including those previously walked down were included in the scope of the current SPRA walkdowns. However, for components that had been previously walked down and for which sufficient information was available to permit development of a fragility, the walkdown was limited to a walk-by of the individual components.

Detailed walkdowns were performed for all components that had not been previously walked down. During a detailed walkdown, the caveats from the Seismic Qualification Utility Group (SQUG) Generic Information Procedure (GIP) [25] were verified and sufficient information was gathered to permit development of a fragility. This included information on anchorage, configuration, weight, dimensions, load path, and other structural information. In addition, the walkdown team focused on potential adverse seismic interaction issues, including the potential for seismically induced fire and flood and seismic II/I concerns, such as masonry block walls in the vicinity of the components.

More simplified walk-bys were performed for components that had been previously walked down. During walk-bys, the walkdown team inspected these components to ensure that there were no obvious changes that might adversely impact their seismic capacity. In particular, the walkdown team focused on potential seismic interaction concerns and conditions. In general, walk-bys were less detailed and less intrusive than walkdowns.

Components that were not accessible during plant operation were walked down during plant outages. Separate walkdowns were performed to assess operator pathways used to perform operator actions, to assess implementation of FLEX, to obtain detailed information related to in-cabinet amplification factors for relays, and to provide specific inputs to the fragility team such as nozzle loads. In addition, even though the walkdown team focused on the potential for seismically induced fire and flood during the walkdowns, a separate walkdown was conducted to specifically evaluate the potential for seismically induced fires due to electrical faults.

Walkdown documentation for equipment and structures consisted of noting the existing conditions, taking photographs, and recording any findings.

4.2.1 Significant Walkdown Results and Insights

Consistent with the guidance from EPRI NP-6041-SL [26], no significant findings or adverse conditions were noted during the SQN seismic walkdowns. Observations made during the walkdowns are documented in the walkdown report [28].

Components on the SEL were evaluated for seismic anchorage, interaction effects (including block walls and other items that might cause a reduction in seismic capacity), and effects of component degradation, such as corrosion and concrete cracking, for consideration in the development of SEL fragilities. In addition, walkdowns were performed on operator pathways, and the potential for seismic-induced fire and flooding scenarios was assessed. Potential internal flood scenarios were incorporated into the SQN SPRA model. The walkdown observations were judged to be adequate for use in developing the SSC fragilities for the SPRA.

4.2.2 Seismic Equipment List and Seismic Walkdowns Technical Adequacy

The SQN SPRA SEL development [23] and walkdowns [28] were subjected to an independent peer review against the pertinent requirements in the ASME/ANS PRA Standard [8]. The SEL development and walkdowns were peer reviewed relative to Capability Category II for the full set of supporting requirements in the Standard. After completion of the subsequent independent assessment, the full set of supporting requirements was met, and the SEL and walkdowns were determined to be acceptable for use in the SPRA.

The peer review assessment [7], and subsequent disposition of peer review findings through an independent assessment [20], is further described in Appendix A and establishes that the SQN SPRA SEL and seismic walkdowns are suitable for this SPRA application.

4.3 Dynamic Analysis of Structures

This section summarizes the dynamic analyses of structures that contain systems and components important to achieving a safe shutdown, using fixed-base and/or soil-structure interaction (SSI) analyses (as applicable). The section describes the methodologies used, responses at various locations within the structures and relevant outputs, important assumptions, and sources of uncertainty. A list of structures and description of relevant parameters is provided in Table 4.3-1.

4.3.1 Fixed-base Analysis

SQN is a firm rock site; SSI was performed for each of the major structures analyzed for the SPRA. Note that fixed-base analyses were performed as a verification step in development of some of the SSI models.

4.3.2 SSI Analysis

Probabilistic SSI analyses considering ground motion incoherence were performed for the RB coupled with ESVR, ACB Complex, AEB, DGB, ADGB and ERCW Pumping Station. The SSI between the structures and the surrounding soil medium is considered by SC-SASSI computer model at defined interaction nodes. Cutoff frequency for the SSI analyses was chosen to be 50 Hz, and the SSI models were sufficiently refined to transmit frequencies of at least 50 Hz through the soil/rock-foundation interface. All SSI analyses utilized the SASSI Direct Method (DM), where all soil layer interface and excavated soil nodes are defined as interaction nodes and the analyses in the three spatial directions are performed simultaneously. The DM calculates the impedance for all interaction nodes present in the soil volume and is deemed the most accurate method for solving SSI problems in SASSI.

The site conditions in the SSI models are represented by uniform horizontal soil layers with equivalent linear soil properties and by an underlying half-space layer.

Median soil profiles are defined with hazard-compatible soil properties based on those from the SQN PSHA report [21]. The soil properties include shear-wave velocity (V_S), compression-wave velocity (V_P), corresponding damping (D_S and D_P), and unit weight. These properties and values are provided by the PSHA for a range of hazard levels. The properties and values are then interpolated between AFE of 1E-4 and 1E-5 to the GMRS level to represent the hazard consistent median soil profile for each structure based on its applicable FIRS. The soil layering profiles (i.e., layer thicknesses) for SSI analysis are refined from that of the PSHA to meet passing frequency requirements.

Probabilistic seismic response analysis was performed following the approach similar to that documented in NUREG/CR-2015 “Seismic Safety Margins Research Programs Phase I Final Report”, which implements Latin Hypercube Simulation (LHS). Variables in the LHS include the earthquake acceleration time histories, structure stiffness and damping, and rock stiffness and damping. Thirty simulations were developed by randomly selecting from each of these variables. The following is a summary of the main steps.

1. Generate 30 sets of ground motions by spectrally matching 30 seed motions to the FIRS and applying directional variability to the spectrally matched motions (SMMs).
2. Develop median structural model.
3. Develop median soil strain compatible soil profiles interpolated to the GMRS.
4. Develop median SSI models using the median structural models with effective stiffness and damping consistent with GMRS-level ground motion.
5. Generate 30 unique model variations of each structural model and uniquely pair them to the 30 SMMs for SSI analysis. Perform probabilistic SSI analysis for the 30 probabilistic cases.
6. Extract results from the 30 SSI analyses and generate results, including the median and 84% In-Structure Response Spectra (ISRS) as well as their variability.

For each simulation, structural and soil properties were defined consistent with their response at a representative acceleration hazard range of interest selected via coordination with fragility and PRA analysts. This hazard range of interest was selected to be the GMRS level based on insights from incremental risk quantifications, especially regarding the relative risk-significance of different acceleration intervals and individual components. A list of structures and description of relevant parameters are provided in Table 4.3-1.

4.3.3 Structure Response Models

The purpose of the mathematical models, which are the Finite Element Model (FEM) or the Lumped Mass Stick Model (LMSM), is to adequately determine the response of the structure in the frequency range of interest consistent with the seismic hazard. The mathematical models include structural elements that form the load-resisting system and appropriately represent the locations of mass and stiffness, thereby accounting for eccentric torsional effects. Dynamic analysis for both LMSMs and FEMs is performed in SC-SASSI to capture structural response due to both horizontal and vertical motions.

The following subsections provide the modeling approach and general input properties used for the development of the FEMs and LMSMs.

4.3.3.1 Lumped Mass Stick Models

The only LMSMs used in this project are the RB portion (Internal Concrete Structure (ICS), Steel Containment Vessel (SCV), Concrete Shield Building (CSB), and NSSS)) of the RB+ESVR FEM and the ADGB. Except for the NSSS, the rest of the LMSMs were modeled using conventional beam elements. The NSSS was modeled using matrix elements. Due to the symmetrical shape of the RB, LMSMs can adequately represent the dynamic properties of the RB structure. Additionally, the existing ADGB LMSM is judged as sufficient to “drive” the foundation slab FEM where the large mass credited equipment is mounted.

The LMSMs of the RB and ADGB meet or exceed the seven criteria listed in Section 6.3.1 of the SPID [3] as minimum requirements, which are paraphrased as follows:

1. The structural models should be capable of capturing the overall structural responses for both the horizontal and vertical components of ground motion.
2. One combined model should be used if there is significant coupling between the horizontal and vertical responses.
3. The structural mass should be lumped so that the total mass, as well as the center of gravity (CG), is preserved.
4. The number of nodal or dynamic degrees of freedom should be sufficient to represent significant structural modes up to 20 Hz.
5. The torsional effects resulting from eccentricities between the CG and the center of rigidity (CR) should be included.
6. The multi-stick model should be used if the “one-stick” model is insufficient to represent the structure.
7. The in-plane floor flexibility (and subsequent amplified seismic response) should be captured appropriately for developing ISRS accurate up to 15 Hz.

4.3.3.2 3D Finite Element Models

As shown in Table 4.3-1, the ESVR, ACB, ERCW Pumping Station, and DGB SSI models were developed using detailed 3D FEMs. Additionally, the substructure portions of all the models, regardless of being LMSM or 3D FEM, were developed using 3D solid or shell elements.

4.3.3.2.1 Reinforced Concrete Walls and Slabs

Shell elements representing the floors were modeled at the center of the slab thickness. However, for the foundation slabs that were modeled with shell elements, the shell elements were placed at the bottom of the slab to be consistent with the soil profile layering elevations and to maintain consistency with the FIRS definition elevations.

The walls were also explicitly modeled with shell elements. The walls were modeled from CG to CG of the slabs. Openings in walls and slabs that were judged to not influence dynamic behavior were neglected. Typically, an opening smaller than about 10% of the wall is considered to have insignificant influence on the overall dynamic characteristics of the structure and, therefore, these small openings could be neglected in the FEMs. Most of the floor slabs and walls were modeled with 4-node shell elements, although 3-node shell elements were used for mesh compatibility.

4.3.3.2.2 Substructure

The RB foundation, ESVR backfill, DGB backfill, ACB spent fuel pit massive slab, and the ERCW Pumping Station tremie concrete cells were modeled with 3D solid elements. All other foundations are modeled using shell elements. The solid elements consist of 8-node elements primarily, although 6-node elements were sometimes used for mesh compatibility.

4.3.3.2.3 Concrete Block Walls

Block walls in the DGB structure were explicitly modeled. However, these walls are considered to crack before the concrete walls and, therefore, not contribute any stiffness to the structural system. Therefore, the modulus of elasticity of these walls is considered as 1% of the value for concrete.

4.3.3.3 Structural Damping

Material damping is considered using the guidance of Sections 3.1.2.2 of ASCE 4-98 [29] as well as Section 3.2.2 of ASCE 4-16 [30], consistent with the damping ratios used in other SPRAs.

Damping is a function of strain response (i.e., the larger the strain, the bigger the damping gets). This is reflected in the ASCE 4-98 [29] Table 3.1-1, which provides damping values for different response levels. For reinforced concrete elements, the median damping ratios are 4% and 7% of critical damping for response level 1 and response level 2, respectively.

For steel structures, the median damping ratios for response level 1 is considered as 2% of critical damping, and as 4% of critical damping for response level 2 based on ASCE 4-98 [29] Table 3.1-1. The justification for the response level used is provided in the SSI model documentation of the applicable structures.

For the reinforced-concrete shear walls in the median models, the response levels and corresponding damping ratios are selected based on the in-plane shear stress and out-of-plane bending stress of the wall. If the average shear and/or bending stresses in the walls at any given time step exceed the stress limits provided in Section C3.2.2 of ASCE 4-16 [30], response level 2 is considered and 7% damping ratio is assigned. If the average shear and/or bending stress in the walls at any given time step do not exceed the stress limits that are provided in section C3.2.2 of ASCE 4-16 [30], response level 1 is considered and 4% damping ratio is assigned. The concrete stress limits for response level and damping determination are $3\sqrt{f'_C}$ for shear, and $7.5\sqrt{f'_C}$ for bending.

For reinforced-concrete slabs and beams, they are considered as cracked due to addition of dead and live load bending stresses to the seismic bending stresses, and response level 2 (7% damping) is assigned.

For the steel beams and columns (both steel and concrete), response level 1 is considered without further investigation. This is because these members are secondary members, and the selection of their damping through detailed evaluation of the stress is not expected to significantly change the overall response.

4.3.3.4 Concrete cracking

The best estimate (median) stiffnesses of concrete structures are consistent with the stress state in the structure. This is accomplished by verifying that the stress state in the main load carrying elements (i.e., concrete shear walls) is consistent with the expected response level, as documented for each structure in its corresponding SSI model documentation. Determination of the effective stiffness of the reinforced concrete members follows the guidance of ASCE 43-05, “Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities.” The adjustment of the stiffness is achieved by changing the cross-section properties (i.e. thicknesses) rather than the elastic and shear moduli. The changes in cross section thicknesses are applied to the specific direction that is cracked (i.e. membrane vs. bending). The reduced section thicknesses are not considered in mass calculation and are only used in stiffness calculations.

Table 4.3-1 Description of Structures and Analysis Methods for SQN SPRA

Structure	Foundation Condition	Type of Model	Analysis Method	Comments/Other Information
Auxiliary-Control Building (ACB)	Rock	3D FEM	Probabilistic (30 THs)	Shear-wave velocity > about 4,500 ft/sec. SSI analysis performed with incoherence, 30 SSI input profiles used.
Reactor Building (RB)	Rock	LMSM	Probabilistic (30 THs)	Shear-wave velocity > about 4,500 ft/sec. SSI analysis performed with incoherence, 30 SSI input profiles used. ESVR and RB are combined into one SSI model since they are connected at the foundation level, and RB response drives the soil supporting the ESVR such that SSSI effects may be influential.
Essential Raw Cooling Water (ERCW) Pumping Station	Rock	3D FEM	Probabilistic (30 THs)	Shear-wave velocity > about 4,500 ft/sec. SSI analysis performed with incoherence, 30 SSI input profiles used.
Additional Equipment Buildings (AEBs)	Rock	3D FEM	Probabilistic (30 THs)	Shear-wave velocity > about 4,500 ft/sec. SSI analysis performed with incoherence, 30 SSI input profiles used.

Table 4.3-1 Description of Structures and Analysis Methods for SQN SPRA

Structure	Foundation Condition	Type of Model	Analysis Method	Comments/Other Information
East Steam Valve Room (ESVR)	Caissons socketed to bedrock	3D FEM	Probabilistic (30 THs)	Shear-wave velocity > about 4,500 ft/sec. SSI analysis performed with incoherence, 30 SSI input profiles used. ESVR and RB are combined into one SSI model since they are connected at the foundation level, and RB response drives the soil supporting the ESVR such that SSSI effects may be influential.
Diesel Generator Building (DGB)	Soil (in-situ)	3D FEM	Probabilistic (30 THs)	Shear-wave velocity > about 300 ft/sec. SSI analysis performed with 30 SSI input profiles used.
Additional Diesel Generator Building (ADGB)	Soil (in-situ)	LMSM	Probabilistic (30 THs)	Shear-wave velocity > about 600 ft/sec. SSI analysis performed with 30 SSI input profiles used.

4.3.4 Seismic Structure Response Analysis Technical Adequacy

The SQN Structural Response Analysis Report [31] was subjected to an independent peer review against the pertinent requirements in the ASME/ANS PRA Standard [8]. The seismic structure response and SSI was peer reviewed relative to Capability Category II for the full set of requirements in the Standard. After completion of the subsequent independent assessment, the full set of requirements was met, and the seismic structure response and SSI were determined to be acceptable for use in the SPRA.

The peer review assessment [7], and subsequent disposition of peer review findings through an independent assessment [20], is further described in Appendix A.

4.4 SSC Fragility Analysis

The SSC seismic fragility analysis considers the impact of seismic events on the probability of SSC failures at a given value of a seismic motion parameter defined as PGA. The fragilities of the SSCs that participate in the SPRA accident sequences (i.e., those included on the SEL) are addressed in the model. Seismic fragilities for the significant risk contributors, i.e., those that have an important contribution to plant risk, are realistic and plant-specific based on actual current conditions of the SSCs in the plant, as confirmed through the detailed walkdown of the plant.

This section summarizes the fragility analysis methodology and presents a tabulation of the fragilities with appropriate parameters for those SSCs determined to be sufficiently risk important based on the final SPRA quantification (as summarized in Section 5). This section also discusses important assumptions and important sources of uncertainty, and any fragility-related insights identified.

4.4.1 SSC Screening Approach

The SQN SEL, consisting of approximately 5,300 components, was reviewed, analyzed, and then reduced to about 1,350 components after walkdowns. The process of reducing the SEL is an iterative and multi-step process as summarized below.

First, the SEL provided to the Seismic Review Team (SRT) was reduced by removing components judged to be non-contributors to the overall response of the SPRA. It was identified that all components that are not in a Category I Building (not counting tanks in the Yard) are not contributors to the SPRA and can be screened as not necessary. These components include anything not within the ACB, RB, ERCW, DGB, or Yard. No fragility value is required for these components.

Components that are judged inherently rugged were also screened out from needing a walkdown. These items included check valves, manual valves, control valves, stop valves, in-line dampers, filters, and safety heads. These components are driven by the system they are mounted on as they are typically more rugged. Passive valves are small, lightweight, robust, and are typically mounted in line with piping. They do not need to change state during or after an event and have no external vulnerabilities. While the failure of one of these valves can contribute to the results of the SPRA, they will be bound by the fragility of the distribution system to which they are attached. No fragility value is specifically developed for passive valves, but fragility for piping is developed. Piping is walked by as part of the distribution system walkdown. This same methodology applies to filters. Dampers are made of robust steel and are typically thick in gauge compared to the duct system to which they are mounted. While they may have to change state after an earthquake, they do not need to change during the seismic event. As was the case with passive valves, the fragility of the damper will be driven by the duct system to which they are mounted. While duct systems were walked down as part of the distribution system walkdown, it is understood that, in general, the failure mode of ducting is usually the supports of that duct. Ducts are either designed to handle tornado vacuum loads, which create more stress in the duct than earthquake loads, or are protected by tornado dampers. The dampers that were not in-line dampers were part of the SEL. These include the fire dampers in the ACB and exhaust dampers in the DGB. These dampers were checked for interaction concern during the walkdowns.

Active valves (Air-Operated Valves (AOVs), Motor-Operated Valves (MOVs), and Solenoid-Operated Valves (SOVs)) are not walked down in their entirety. The majority of SQN safety-related AOVs, MOVs and SOVs pass the 3g horizontal and 2g vertical seismic qualification criteria documented in SQN-DC-V-46.0 [32]. This demonstrates that the SQN valves have sufficient seismic adequacy. Since these valves do contribute to the response of the SPRA and are an important part of the model, walkdown efforts were made to locate these valves, examine for any vertical vulnerability, and measure the operator height. If a valve was not easily located, especially in the high-dose area, walk-bys and walkthroughs were performed of the entire area looking for interaction or proximity issues. The valves that change state or are required to change state during or after a seismic event are addressed in the fragility analysis. The active valves with vertical vulnerability were noted during the walkdowns and considered as special cases for fragility evaluation. The valves that do not satisfy the 3g/2g criteria but are qualified to lower accelerations based on piping analysis were addressed on an as-needed basis during the fragility analysis. In general, SOVs are seismically rugged and screened out. However, the SOVs that change state following an accident were included in the walkdown list and evaluated for initial risk quantification.

The components that reside inside other components are screened by the rule-of-the-box. Examples include level indicators inside tanks and switches inside a panel. Like active valves, these components are still addressed in the fragility analysis, but a walkdown of the box component is all that is necessary. These devices were modeled in the SPRA with the fragility value of their box assigned to them. It was assured that boxes containing devices are included in the SEL.

4.4.2 SSC Fragility Analysis Methodology

For the SQN SPRA, the following methods were used to determine seismic fragilities for SSCs included in the SPRA:

> Consistent with the requirements in ASME/ANS PRA Standard [8], the fragility analysis for the selected SSCs is based on the methodology in EPRI guidelines. The strategy for developing the fragilities for the complete set of SSCs on the SPRA SEL follows the recommendations of EPRI NP-6041-SL [26], EPRI 1019200 [33], EPRI 103959 [34] and EPRI 3002000709 [22] and proceeds progressively from using experienced-based capacities to component-specific-evaluations. Regardless of the method, the development of fragility estimates uses plant-specific information based on SSC conditions, as confirmed through detailed walkdowns.
>
> Components are first binned into equipment classes, e.g., EPRI classes presented in Appendix F of EPRI NP-6041-SL, and then grouped according to similarity and location. Representative samples in each equipment group are then evaluated to obtain fragility estimates for all the items in the group.

The SPRA approach used at SQN initially utilized three quantifications. In addition to these formal quantifications, various sensitivity studies were performed during the effort to help identify important risk contributors. After each quantification and completion of the sensitivity studies, components identified as risk-significant were selected and evaluated further to improve their calculated fragilities in order to reduce their risk significance. This approach has been successfully implemented at several plants and complies with the ASME Standard [8] and the SPID [3]. All three quantifications and numerous sensitivity studies were performed prior to the peer review. Subsequent to the peer review and to address peer review findings, additional quantifications were performed. After each quantification, the results were reviewed to determine if additional insights were obtained and to determine if further refinement of fragilities associated with top risk contributors would improve the results and yield a more realistic model.

For the first quantification, site-specific representative fragilities (referred to as ‘representative’ throughout) were typically developed by scaling existing design basis calculations to account for available margins in the design. This is the margin between allowable values associated with design requirements and values associated with HCLPF evaluations. These margins were used to develop a Safety Factor, which is anchored to the PGA of the GMRS to estimate a HCLPF fragility value. The generic values of aleatory variability and epistemic uncertainty from the SPID [3] were applied to the HCLPF to obtain the median fragility value.

For the second quantification, "enhanced" fragilities were provided for top risk contributors to both SCDF and SLERF. The top risk contributors were determined based on the F-V numbers from the initial quantification and subsequent sensitivity studies. The cutoff F-V value for selecting components from the first quantification was 5E-05 for both SCDF and SLERF. This is well below the threshold from the ASME Standard of 5E-03. The fragilities were calculated using the Conservative Deterministic Failure Margin (CDFM) method to determine the HCLPF. The generic uncertainty values, as recommended in Table 6.2 of the SPID for various SSCs, were used to estimate the median fragility value, with the generic uncertainty values adjusted if needed to account for specific conditions. Site-specific information obtained from walkdowns and plant documentation, including actual anchorage and configuration details, were used along with ISRS at the location of the individual components.

Fragilities for the third quantification were developed for the dominant risk contributors as identified during the second SPRA quantification. When beneficial, the fragilities for the final quantification were computed using the Separation of Variable (SoV) approach, where the median capacity and the associated variabilities are calculated rigorously, and then the HCLPF capacity is back-calculated using the median capacity and the variabilities. The SoV approach provides more realistic fragilities.

Critical failure modes, such as structure/anchorage or functionality or block wall, were identified and fragility calculations were performed for the median capacity A_m for each of the failure modes. The lowest, governing A_m was selected and when two or more failure modes were close (i.e., their median capacities within 20% of each other), the governing median capacity was computed for combined failure.

The NSSS was evaluated for fragility variables. The NSSS includes the reactor vessel, the steam generators, the reactor coolant pumps (RCPs), a pressurizer, and the piping that connects these components to the reactor vessel. The fragility evaluation of these components was based on scaling of the existing safety analysis results, in accordance with SPID guidance.

Subsequent to the peer review, additional quantifications were performed to further refine the SPRA model and to respond to peer review findings. These quantifications are described in Section 5 of this report. To support these quantifications, additional refined fragilities were developed using either the CDFM or SoV approach as appropriate. Table 4.4-1 provides a summary of the number of components for which fragilities were developed for each quantification. Note that the number of SSCs included in the SPRA model was not reduced to the numbers shown in this table for the second quantification onward. Fragilities that were not improved were carried over from one quantification to the next. In some cases, refined fragilities were provided for certain SSCs for use in various sensitivity studies. These refined fragilities were developed based on estimates and maximum potential improvements to determine the impact and benefit of developing more detailed fragilities for these items based on the results of the sensitivity studies.

Table 4.4-1 Approximate Numbers of Refined SSC and Relay Fragilities for Each Risk Quantification

Quantification	Count of SSC
Q1	~1200
Q2	~450
Q3	~70
Post Peer Review	~200

The Watts Bar Nuclear Power Plant (WBN) is a sister plant of SQN operated by TVA. The general outline and equipment layout at SQN are essentially the same as the general outline and equipment layout at WBN. A comparison review of the reinforcement drawings reveals that the reinforcement is very similar between WBN and SQN in the walls and floor slabs of the safety-related structures. Structure fragility for the RB, including its sub-structures ICS, SCV and CSB), ACB Complex and DGB, was estimated by scaling the corresponding WBN structural fragilities. Where applicable, the screening method outlined in EPRI NP-6041-SL [26] was used to estimate the structure fragility for other structures.

4.4.3 SSC Fragility Analysis Results and Insights

The final set of fragilities for the risk-important contributors to SCDF and SLERF are summarized in Section 5. Refer to Tables 5.4-3 and 5.4-4 for SCDF and Tables 5.5-3 and 5.5-4 for SLERF. Detailed SoV calculations have been performed for selected highest risk-significant SSCs, as well as for other components.

Consistent with the three-step graded approach for risk quantification, components for refinement were selected based on interim sensitivity studies and previously completed risk quantifications. The fragilities of selected components that were identified to be risk-significant were previously refined using the CDFM-based Hybrid Method, and several were refined using the SoV Approach. Using the refined fragilities in the subsequent risk quantifications resulted in either the refined fragility group becoming less risk-significant or new fragility groups (with CDFM-based fragility) becoming more risk-significant.

As stated in Section 6.4.1, EPRI SPID [3],

"*The CDFM approach for developing fragilities is a simpler method that can be performed consistently by more analysts and is an acceptable approach for generating fragilities within an SPRA for the majority of components for which a less detailed assessment is necessary. Because only a handful of components are risk-significant enough to justify the additional effort required by the separation of variables method, the CDFM method can provide efficiencies in the overall effort. Therefore, use of the CDFM approach is useful and beneficial for calculating fragilities of SSCs for use in seismic PRAs conducted to address the 50.54(f) letter.*"

After the final risk quantification, as previously described, many of the SSCs with refined fragilities based on the SoV approach dropped off the risk-significant list, and other SSCs with refined fragilities based on CDFM approach appeared on the risk-significant list. Sensitivity studies were conducted after the final risk quantification by varying the fragilities of risk-significant SSCs to ensure that the overall risk profile remains stable. Those sensitivity studies are discussed in section 5.7.

4.4.4 SSC Fragility Analysis Technical Adequacy

The SQN SPRA SSC Fragility Analysis [36] was subjected to an independent peer review against the pertinent requirements in the ASME/ANS PRA Standard [8]. The SSC fragility analysis was peer reviewed relative to Capability Category II for the full set of supporting requirements in the standard. After completion of the subsequent independent assessment [20], the full set of supporting requirements were met, and the SSC fragility analysis was determined to be acceptable for use in the SPRA.

The peer review assessment [7], and subsequent disposition and closure of peer review findings through an independent assessment [20], is further described in Appendix A.

5.0 Plant Seismic Logic Model

The seismic plant response analysis models the various combinations of structural, equipment, and human failures given the occurrence of a seismic event that could initiate and propagate a seismic core damage or large early release sequence. This model is quantified to determine the overall SCDF and SLERF and to identify the important contributors, e.g., important accident sequences, SSC failures, and human actions. The quantification process also includes an evaluation of sources of uncertainty and provides a perspective on how such sources of uncertainty affect SPRA insights.

5.1 Development of the SPRA Plant Seismic Logic Model

The SQN seismic response model was developed by starting with the SQN internal events at-power Level 1/Level 2 PRA model of record as of August 2014 [47], and adapting the model in accordance with guidance in the SPID [3] and ASME/ANS PRA Standard [8], including addition of seismic initiating events (IEs) based on the plant-specific seismic hazard curve and seismic fragility-related basic events to the appropriate portions of the IEPRA, eliminating some parts of the internal events model that do not apply, and adjusting the IEPRA model human reliability analysis to account for response during and following a seismic event. This modeling approach leaves the IEPRA system logic intact while incorporating the necessary additions required for the SPRA. The SQN internal events at-power PRA model of record as of August 2014 was subjected to an independent assessment per the guidelines of NEI 12-13 Appendix X [9] as described in Section A.6 of Appendix A.

The SQN SPRA model was developed using the EPRI Risk and Reliability Workstation software suite (CAFTA, FRANX, HRA Calculator, ACUBE, SYSIMP and UNCERT). The permanently installed 480V Flexible and Diverse Coping Strategies (FLEX) diesel generators and the 6.9kV FLEX diesel generators are credited in the model. Both random and seismic-induced failures of modeled SSCs were included. The seismic-induced fire and flooding were evaluated as well.

5.1.1 Seismic Initiating Event

The seismic IE was modeled using nine discrete hazard bins based on increasing PGA. The seismic hazard bins are listed in Table 5.1-1. Each bin is treated as a seismic initiator, and the SCDF and SLERF results are summed over all the bins to obtain the total CDF and SLERF.

The bin ranges were chosen such that the first bin covers the PGA range from the Operating Basis Earthquake (OBE) to the Safe Shutdown Earthquake (SSE), while the second covers the range from the SSE to a common review level earthquake (RLE) of 0.3g.

The OBE, the strongest earthquake at which the plant is designed to be able to continue normal operation, is defined as 0.09g. Below 0.09g, no significant seismic impacts are expected. The SSE is defined as an acceleration of 0.18g. The plant is seismically designed such that safety-related equipment should not fail given an SSE.

Table 5.1-1 Seismic Hazard Bins

Seismic Bin	Lower Bound (g)	Upper Bound (g)	Bin Mean PGA (g)	Bin Mean Frequency (1/y)	Notes
%G01	0.09	0.18	0.13	4.0E-04	OBE to SSE
%G02	0.18	0.3	0.23	1.2E-04	SSE to 0.3g RLE
%G03	0.3	0.5	0.39	5.4E-05	0.3g RLE to 0.5g RLE
%G04	0.5	0.7	0.59	1.5E-05	
%G05	0.7	0.9	0.79	5.5E-06	
%G06	0.9	1.1	1.00	2.4E-06	
%G07	1.1	1.5	1.28	1.8E-06	
%G08	1.5	3.0	2.12	9.7E-07	
%G09	3.0	Unbounded	3.3	9.2E-08	
				Total=6.0E-04	

Note: For %G09, FRANX calculates the representative ground motion as the addition of 10% to the lowest PGA of the bin, 1.1* 3.0g = 3.3g.

5.1.2 Accident Sequences

The IEPRA uses event trees (ETs) to model the potential plant responses to IEs. The SPRA uses the same approach. The SPRA uses a seismic initiating event tree (SIET) to partition the seismic IE into accident sequence types typically modeled in the IEPRA. Transfers can then be made from the SIET to the corresponding IEPRA ETs to model plant response.

The SIET top events include the recommended minimum set of IEs listed in NUREG/CR-4840 [44] except for the initial status of the power conversion system. No credit is taken for non-safety-related equipment such as the power conversion system in the SQN SPRA base case.

An additional top event involving seismically induced direct core damage is included in the SIET. The sequence leads directly to core damage and, therefore, does not transfer to an IEPRA ET. Structural failures of the RB, ACB, or DGB combined with a loss of offsite power (LOSP) are assumed to lead directly to core damage. Reactor vessel ruptures or other excessive loss of coolant accidents (LOCAs) are also assumed to lead to core damage. Structural support failures of the reactor pressure vessel, pressurizer, or steam generator are assumed to lead directly to core damage. Finally, seismic failure of the control room ceiling resulting in operator abandonment and failure to shut down the plant remotely is assumed to lead to core damage.

5.1.3 Loss of Offsite Power

The fragility of seismically induced LOSP resulting from switchyard or grid failures was obtained from Table 6-1 in NUREG/CR-6544 [37]. Seismic-induced LOSP is predicted to occur with a median magnitude 0.3g. The predicted failure mode is failure of ceramic insulators in the switchyard. Use of this fragility for seismically induced LOSP is a standard industry practice for plants in the eastern portion of the United States. The path for transmission of offsite power to safety-related equipment and non-safety-related equipment within the plant was considered to be governed by the fragility for seismically induced offsite power, including any paths through the Turbine Building (TB). Note that seismically induced LOSP is assumed to fail both switchyards (complete seismic correlation). The SPRA takes no credit for recovery of offsite power.

5.1.4 Very Small LOCA

SPRAs need to consider whether a coincident very small LOCA (VSLOCA) needs to be modeled for other SIET sequences. For other LOCA sequences, which are small LOCA (SLOCA), medium LOCA (MLOCA), large LOCA (LLOCA) and interfacing system LOCA (ISLOCA), the addition of a coincident very small LOCA would have no impact because the other LOCA modeled is already larger than a very small LOCA. Also, the direct core damage events modeled are not impacted by a very small LOCA because they are assumed to go directly to core damage and early release. Inclusion of a coincident very small LOCA might potentially impact accident progression and success for SIET sequences of general transient, steam generator tube rupture, secondary side break inside containment, and secondary side break outside containment. However, the plant specific fragility analysis determined that the seismic fragility for the very small LOCA was high (A_m=2.04 g).

5.1.5 Seismic Level 2 Analysis

The SPRA Level 2 analysis provides a method for estimating the SQN seismically induced LERF. The seismic Level 2 PRA analysis includes an accident event progression following core melt that is similar to the event progression initiated by an internal events initiator. The SQN IEPRA developed a complete Level 2 model consisting of Containment Event Trees (CETs) and supporting Level 2 fault trees. Those sequences of the complete Level 2 model that resulted in a large early release were designated as SLERF sequences. The SPRA used the complete Level 2 model and incorporated the impact of seismic events into it. As a result, the SPRA can be used to determine the frequencies of each of the Level 2 accident progression end states, as well as being used to determine SLERF.

The process of performing the containment analysis begins with an evaluation of the SQN SPRA Level 1 sequences. These sequences are categorized in terms of the type of challenge to containment posed by each sequence and the operability of systems that could mitigate these effects. These Plant Damage States (PDSs) are used to assist in the linking of seismic Level 1 sequences to the appropriate Level 2 sequences. While each seismic Level 1 accident sequence is explicitly treated in the CAFTA computer model of the SQN plant, the Level 1 sequence logic is transferred into the developed Level 2 CETs to take advantage of the similarities in accident challenges from the Level 1 analysis and to streamline the quantification of the core melt progression CETs. The PDS grouping and the CETs identify the general course of the accident sequence, including which systems are operating and the specific phenomena that may occur. For the SPRA, releases classified as LATE in the internal events model were considered to be early for seismic bins %G04 and above with the conservative assumption of no seismic impacts on evacuation timing for earthquakes up to 0.5g.

Development of the Level 2/SLERF model for the seismic sequences was performed in the same manner as for the IEPRA. In the IEPRA, each sequence in the ET that results in core damage is sufficiently subdivided to indicate the type of event, the state of the primary system, and the state of containment protection systems. Each Level 1 end state that results in core damage is classified into PDSs, which specify whether containment has been bypassed, whether the Reactor Coolant System (RCS) pressure is high or low, and whether the steam generator tubes are wet or dry. The state of these PDSs for each Level 1 core damage sequence determines which CET the sequence is input to in the Level 2 analysis. For sequences that can result in a large early release, the SLERF or the Conditional Large Early Release Probability (CLERP) can be determined.

5.1.6 Summary of Resulting Correlated Component Groupings

Correlation of components (or common cause failure) is considered in accordance with the ASME/ANS PRA Standard [8]. There are insufficient data on partial or full correlation of seismic failures of similar components in similar locations and alignments to perform sophisticated seismic correlations in SPRAs. Instead, a common practice is to assume complete seismic correlation for these groups of similar components, locations, and alignments. The SQN SPRA results involve complete seismic correlation within fragility groups.

5.1.7 Summary of HRA methodology

Operator actions that are modeled in the SPRA are either pre-initiator or post-initiator. Pre-initiator Human Failure Events (HFEs) are events that represent the impact of human failures committed prior to the initiation of an accident sequence (e.g., during test or maintenance or the use of calibration procedures). Pre-initiator actions are latent and not affected by seismic events, so their assessments are not changed from the IEPRA model.

The list of post-initiator human actions for the internal events model is the starting point of the seismic HRA, and all existing HRAs are analyzed for modification due to seismic effects. The HFEs associated with the existing accident sequence models were retained in the SPRA model. The model was also examined for any potential human actions unique to the seismic analysis, and any new operator actions identified were added to the SPRA. Any new operator actions added to the seismic model are discussed further in the SPRA HRA Notebook [42].

Since the potential earthquakes examined vary in magnitude, as does the on-site acceleration, the level of plant damage varies accordingly due to the impacts of the different seismic events. Post-initiator HFEs retained in the SPRA model were evaluated for seismic impacts. The degree of impact is dependent on the seismic acceleration level. The seismic impacts on every post-initiator HFE in the SPRA models were accounted for by the HFE-specific performance shaping factors and selected minimal values that increase with acceleration as a function of the PDS. Following EPRI

Report 3002008093 [41], the seismically-adjusted HFEs use the internal events HFE nomenclature, with a suffix of "_Sn," where n ranges from 1 to 4; i.e., four separate seismic acceleration ranges were evaluated for varying seismic impacts. The SPRA HRA Notebook discusses which HFE bins correspond to which seismic acceleration levels. For bin S4 (which includes the highest acceleration seismic initiators), it was conservatively assumed that all post-trip actions are set to failed (1.0).

The use of the same method from the internal events model for the HRA dependency analysis is valid for the SPRA HRA. The SPRA HRA Notebook discusses the method used to assess HFE dependency. The SPRA Quantification Notebook [1] also has details of how the HRA dependency analysis was performed.

Accessibility for HFEs performed outside the control room was addressed by walkdowns.

5.1.8 Seismic-Fire

Seismic-fire interaction events have the potential to contribute significantly to core damage or large early release. The guidelines in Appendix G of EPRI 3002000709 [22] were followed in the identification and assessment of potential seismic-fire interaction events. That effort included an assessment of fire ignition sources categorized as medium or higher and additional sources identified in the IPEEE/Final Safety Analysis Report (FSAR).

Seismic-fire interaction event identification and assessment is included in the SQN SPRA Seismic-Fire Interaction calculation [38]. The results of the assessments indicated some of those events needed to be included in the SPRA. The other ignition sources were screened because: (1) such events would not impact modeled equipment; (2) impacts are already covered by fragility assignments; or (3) EPRI [22] assessed such events as having a low potential for seismically induced fire.

The unscreened fire ignition sources due to seismic impact were assumed to go directly to core damage since the SQN Fire PRA is not complete. The seismic-induced fire sources were modeled with a fragility of 3.21g (A_m) and, therefore, were not a significant contributor to SCDF.

5.1.9 Seismic-Flood

Seismic-flood interaction events have the potential to contribute significantly to core damage or large early release. A two-step process was used to identify such events at SQN. The first step was to review internal flood scenarios modeled in the internal flooding portion of the IEPRA [39]. All scenarios from the IEPRA were identified and were subject to further evaluation, including the scenarios that were screened out in the internal events model. The screening in the IEPRA is based on the frequency (CDF or LERF), which is the product of the pipe-break frequency and the conditional core damage probability (CCDP) or CLERP. For a seismic event, the pipe-break frequencies directly depend on the occurrence frequency of a certain level of earthquake and the pipe fragility. The flooding scenarios that were screened out due to the very low internal event pipe-break frequency may have considerably high CCDP or CLERP and become seismically risk-significant in combination with the potential seismic failure of other equipment. Therefore, all internal flooding scenarios, screened or not, are included in the seismic-flood interaction evaluation. Second, the scenario with the highest CCDP is chosen as the seismic flooding scenario for a flooding source that may be from various piping or tanks.

There are three types of scenarios in the IEPRA internal flooding analysis: spray, flood, and major flood. The major flood scenario typically has the highest failure probability of the three and is kept for the seismic-flood interaction model. Risk evaluation SQN-0-17-095 [40] documents the basis for selecting the flooding scenarios. The seismic flooding model is built in FRANX, and no operator flooding recovery actions are credited for the seismic-induced flooding. Finally, the seismic flooding model is injected into the CAFTA fault tree via the FRANX tool XINIT for the SQN SPRA quantification.

5.2 SPRA Plant Seismic Logic Model Technical Adequacy

The SQN SPRA seismic plant response methodology and analysis were subjected to an independent peer review against the pertinent requirements in the ASME/ANS PRA Standard [8]. The seismic plant response methodology and analysis were peer reviewed relative to Capability Category II for the full set of supporting requirements in the Standard. After completion of the subsequent independent assessment, the full set of supporting requirements was met, and the seismic plant response methodology and analysis were determined acceptable for use in the SPRA.

The peer review assessment [7], and subsequent disposition of peer review findings through an independent closure peer review assessment [20], are further described in Appendix A.

5.3 Seismic Risk Quantification

In the SPRA risk quantification, the seismic hazard is integrated with the seismic response analysis model to calculate the frequencies of core damage and large early release of radioactivity to the environment. This section describes the SPRA quantification methodology and important modeling assumptions.

5.3.1 SPRA Quantification Methodology

Once the SQN SPRA single top logic was developed [1], the model can be quantified for core damage and large early release. The FRANX 4.2 software was used to perform this quantification. Several ACCESS tables within FRANX are used to define the seismic hazard bins, assign seismic fragilities to basic events with the logic model, calculate fragilities associated with each of the seismic hazard bins, and assign a Human Error Probability (HEP) by seismic bin for each HFE.

The following steps were used to perform the SPRA model quantification for both SCDF and SLERF for each unit:

(1) Obtain CCDP or CLERP cutsets for each seismic bin using FRANX 4.2 and ACUBE with initial fragility and HEP values and generally assuming complete seismic correlation within fragility groups.

(2) Identify fragilities and HEPs to be refined.

(3) Refine fragility groups for complete seismic correlation modeling.

(4) Identify final set of fragilities to be inserted into the model (because of model size limitations and software constraints).

(5) Perform truncation sensitivity to determine final truncation level; quantify the models (four FRANX files) with initial HEP values.

(6) Assemble bin cutsets into combined cutset files (one for SCDF and one for SLERF).

(7) Perform HFE detailed HRA analysis and HFE dependency analysis; incorporate new HRA values into the model.

(8) Finalize quantification of SCDF and SLERF (ACUBE analysis).

(9) Evaluate basic event importances (SYSIMP/ACUBE analysis supplemented by selected sensitivity analyses).

(10) Perform uncertainty analysis (UNCERT).

(11) Evaluate sensitivity cases.

Specific issues related to quantification are discussed in the following sections addressing SCDF and SLERF results.

5.3.2 SPRA Model and Quantification Assumptions

Hazard analysis assumptions:

> Refer to Section 3.1 of this submittal for a discussion of assumptions and uncertainties associated with the hazard analysis.

Structures/fragilities analyses assumptions.

> Most of the structure/fragilities analyses uses the CDFM method. The CDFM method predicts a slightly conservative-biased fragility.

Key plant response modeling assumptions:

1. Structural failures of the RB, ACB, or DGB (combined with LOSP) are assumed to fail sufficient equipment within the structure to lead directly to core damage and large early release.
2. In addition to these large structure failures, seismic failures of the reactor vessel and its supports and structural failures of the pressurizer and steam generator supports are also considered to lead directly to core damage and large early release.
3. Finally, the combination of a seismically induced failure of the control room (ceiling collapse) and failure of the operators to safely shut down the plant remotely is also assumed to lead directly to core damage and large early release.

   These are potentially conservative assumptions.

5.4 SCDF Results

5.4.1 Overall SCDF

The SPRA shows that the point estimate SCDF is 4.1E-06 per ry for Unit 1 and is 4.9E-06 per ry for Unit 2 [1].

5.4.2 SCDF as a Function of Hazard Interval

A summary of the SCDF results for each seismic hazard interval is presented in Table 5.4-1 for Unit 1 SCDF and Table 5.4-2 for Unit 2 SCDF. The maximum CCDP is 0.908 for Unit 1 and 0.932 for Unit 2. These are equal to the plant availability factors for each unit, respectively.

Table 5.4-1 Unit 1 SCDF Contribution by Initiating Event

Truncation	Scenario	Description	Earthquake Frequency	CCDP	SCDF	Percent Contribution
1.0E-14	%G01	Seismic Initiating Event (0.09g to <0.18g)	4.0E-04	7.53E-06	2.99E-09	0.1%
1.0E-14	%G02	Seismic Initiating Event (0.18g to <0.3g)	1.2E-04	2.29E-04	2.83E-08	0.7%
1.0E-12	%G03	Seismic Initiating Event (0.3g to <0.5g)	5.4E-05	6.54E-03	3.53E-07	8.6%
1.0E-12	%G04	Seismic Initiating Event (0.5g to <0.7g)	1.5E-05	5.10E-02	7.59E-07	18.4%
2.0E-09	%G05	Seismic Initiating Event (0.7g to <0.9g)	5.5E-06	1.07E-01	5.90E-07	14.3%
5.0E-09	%G06	Seismic Initiating Event (0.9g to <1.1g)	2.4E-06	2.36E-01	5.65E-07	13.7%
5.0E-08	%G07	Seismic Initiating Event (1.1g to <1.5g)	1.8E-06	4.72E-01	8.55E-07	20.8%
7.0E-07	%G08	Seismic Initiating Event (1.5g to <3g)	9.7E-07	9.08E-01	8.78E-07	21.3%
1.0E-08	%G09	Seismic Initiating Event (>3g)	9.2E-08	9.08E-01	8.39E-08	2.0%
				Total SCDF=	**4.1E-06**	

Table 5.4-2 Unit 2 SCDF Contribution by Initiating Event

Truncation	Scenario	Description	Earthquake Frequency	CCDP	SCDF	Percent Contribution
1.0E-14	%G01	Seismic Initiating Event (0.09g to <0.18g)	4.0E-04	9.19E-06	3.65E-09	0.1%
1.0E-14	%G02	Seismic Initiating Event (0.18g to <0.3g)	1.2E-04	2.71E-04	3.36E-08	0.7%
1.0E-12	%G03	Seismic Initiating Event (0.3g to <0.5g)	5.4E-05	7.00E-03	3.77E-07	7.7%
1.0E-12	%G04	Seismic Initiating Event (0.5g to <0.7g)	1.5E-05	6.26E-02	9.33E-07	19.2%
2.0E-09	%G05	Seismic Initiating Event (0.7g to <0.9g)	5.5E-06	1.49E-01	8.21E-07	16.9%
5.0E-09	%G06	Seismic Initiating Event (0.9g to <1.1g)	2.4E-06	3.00E-01	7.20E-07	14.8%
5.0E-08	%G07	Seismic Initiating Event (1.1g to <1.5g)	1.8E-06	5.47E-01	9.90E-07	20.3%
7.0E-07	%G08	Seismic Initiating Event (1.5g to <3g)	9.7E-07	9.32E-01	9.01E-07	18.5%
1.0E-08	%G09	Seismic Initiating Event (>3g)	9.2E-08	9.32E-01	8.61E-08	1.8%
				Total SCDF=	**4.9E-06**	

5.4.3 Significant Systems, Structures, and Components for SCDF

The SSCs with the most significant seismic failure contributions to SCDF for Unit 1 are listed in Table 5.4-3, sorted by F-V. The seismic fragilities for each of the significant contributors are also provided in Table 5.4-3, along with the corresponding limiting seismic failure mode and method of fragility calculation. The corresponding measures for Unit 2 are presented in Table 5.4-4.

Table 5.4-3 Unit 1 SCDF Importance Measures Ranked by F-V[1]

Fragility Group	F-V	Fragility					
		A_m (g)	βu	βr	Description	Failure Mode	Fragility Method
SEIS_LOSP	0.597	0.3	0.43	0.19	LOSP (Loss of Offsite Power)	Ceramic insulators	Table 6-1 NUREG/CR-6544
SEIS_0-30-5	0.382	0.84 [2]	0.32	0.24	Relay Chatter - 480V/6.9kV SD BD GE HEA Breakers	Functionality	CDFM
SEIS_23-5	0.078	1.79	0.26	0.24	NSSS Reactor Pressure Vessel	Anchorage	CDFM
SEIS_0-16-2	0.068	2.04	0.32	0.24	Instrument Line Very Small LOCA	Anchorage	CDFM
SEIS_VSLOCA	0.038	2.04	0.32	0.24	Seismic Very Small LOCA	Anchorage	CDFM
SEIS_3-4-1	0.038	0.8	0.38	0.24	120V AC Vital Inverter in Aux. sub 1	Anchorage	CDFM
SEIS_23-6	0.024	2.24	0.26	0.24	NSSS Steam Generator	Anchorage	CDFM
SEIS_23-6A	0.019	2.24	0.26	0.24	Steam Generator Support Failure	Anchorage	CDFM
SEIS_23-4	0.016	2.54	0.32	0.24	NSSS Pressurizer	Anchorage	CDFM
SEIS_HINST (see Note)	0.016	1.32	0.32	0.24	Seismically induced failure of HRA Main Control Room (MCR) Instrumentation	Functionality	CDFM
SEIS_19-1	0.014	0.79	0.26	0.24	Condensate Storage Tank (CST)	Anchorage	CDFM
SEIS_1A-4	0.013	1.02 [2]	0.32	0.24	480V DG Aux. Board	Block Wall	CDFM
SEIS_19-9	0.013	1.04	0.26	0.24	Residual Heat Removal (RHR) Heat Exchanger	Anchorage	CDFM
SEIS_0-30-11	0.012	0.71	0.32	0.24	Relay Chatter - DGB Control Panel Kraus & Naimer relays	Functionality	CDFM
SEIS_3-4-2	0.011	0.8	0.38	0.24	120V AC Vital Inverter in Aux. sub 2	Anchorage	CDFM

Table 5.4-3 Unit 1 SCDF Importance Measures Ranked by F-V[1]

Fragility Group	**F-V**	**Fragility**					
		A_m (g)	**βu**	**βr**	**Description**	**Failure Mode**	**Fragility Method**
SEIS_0-12	0.01	1.07	0.32	0.24	Block Walls in Diesel Generator Building	Block Wall	CDFM
SEIS_0-30-6	0.009	0.63	0.32	0.24	Relay Chatter - 480V/6.9kV SD BD, Pumps, Misc. Breakers	Functionality	CDFM
SEIS_0-1	0.008	1.73	0.43 [2]	0.19 [2]	Buried ERCW piping	Soil (Buried Piping) Failure	SoV
SEIS_1C-1-2	0.007	1.5	0.32	0.24	U2 6.9kV Shutdown Board	Functionality	CDFM
SEIS_0-30-10	0.007	0.84	0.32	0.24	Relay Chatter - DGB Control Panel Square D Relays	Functionality	CDFM

1. Importance rankings obtained from SYSIMP/ACUBE output. The fragility group SEIS_HINST is a combination of fragilities rather than a single fragility. See the SHR Notebook for details.
2. The fragility parameters for SEIS_0-30-5, SEIS_0-30-15, SEIS_1A-4, and SEIS_0-1 were revised following the final quantification. These changes are addressed in sensitivity study 31 in Table 5.7-1.

Table 5.4-4 Unit 2 SCDF Importance Measures Ranked by F-V [1]

Fragility Group	F-V	Fragility					
		A_m (g)	βu	βr	Description	Failure Mode	Fragility Method
SEIS_LOSP	0.667	0.3	0.43	0.19	LOSP (Loss of Offsite Power)	Ceramic insulators	Table 6-1 NUREG/CR-6544
SEIS_0-30-5	0.479	0.84 [2]	0.32	0.24	Relay Chatter - 480V/6.9kV SD BD GE HEA Breakers	Functionality	CDFM
SEIS_3-4-1	0.085	0.8	0.38	0.24	120V AC Vital Inverter in Aux. sub 1	Anchorage	CDFM
SEIS_0-16-2	0.076	2.04	0.32	0.24	Instrument Line Very Small LOCA	Anchorage	CDFM
SEIS_23-5	0.059	1.79	0.26	0.24	NSSS Reactor Pressure Vessel	Anchorage	CDFM
SEIS_VSLOCA	0.04	2.04	0.32	0.24	Seismic Very Small LOCA	Anchorage	CDFM
SEIS_3-4-2	0.038	0.8	0.38	0.24	120V AC Vital Inverter in Aux. sub 2	Anchorage	CDFM
SEIS_23-6	0.024	2.24	0.26	0.24	NSSS Steam Generator	Anchorage	CDFM
SEIS_0-30-11	0.021	0.71	0.32	0.24	Relay Chatter - DGB Control Panel Kraus & Naimer relays	Functionality	CDFM
SEIS_23-6A	0.016	2.24	0.26	0.24	Steam Generator Support Failure	Anchorage	CDFM
SEIS_HINST	0.015	1.32	0.32	0.24	Seismically-induced failure of HRA MCR Instrumentation	Functionality	CDFM
SEIS_23-4	0.014	2.54	0.32	0.24	NSSS Pressurizer	Anchorage	CDFM
SEIS_0-30-6	0.014	0.63	0.32	0.24	Relay Chatter - 480V/6.9kV SD BD, Pumps, Misc. Breakers	Functionality	CDFM
SEIS_1A-4	0.013	1.02 [2]	0.32	0.24	480V DG Aux. Board	Block Wall	CDFM

Table 5.4-4 Unit 2 SCDF Importance Measures Ranked by F-V [1]

Fragility Group	F-V	Fragility					
		A_m (g)	βu	βr	Description	Failure Mode	Fragility Method
SEIS_0-30-10	0.012	0.84	0.32	0.24	Relay Chatter - DGB Control Panel Square D Relays	Functionality	CDFM
SEIS_19-9	0.012	1.04	0.26	0.24	RHR Heat Exchanger	Anchorage	CDFM
SEIS_0-12	0.011	1.07	0.32	0.24	Block Walls in Diesel Generator Building	Block Wall	CDFM
SEIS_0-1	0.009	1.73	0.43 [2]	0.19 [2]	Buried ERCW piping	Soil (Buried Piping) Failure	SoV
SEIS_FLEX_BUS480	0.008	1.45	0.38	0.24	FLEX 480V DG Bus Panel	Functionality	CDFM
SEIS_16-1	0.008	1.71	0.38	0.24	Aux. Control Air Compressor	Functionality	CDFM
SEIS_19-1	0.008	0.79	0.26	0.24	Condensate Storage Tank (CST)	Anchorage	CDFM
SEIS_0-30-9	0.007	0.88	0.32	0.24	Relay Chatter - 480V SD BD Supply Breakers	Functionality	CDFM
SEIS_1B-1	0.005	1.4	0.38	0.24	480V Shutdown Board	Anchorage	CDFM
SEIS_12-2	0.005	1.47	0.32	0.24	ERCW Pump	Functionality	CDFM
SEIS_IF	0.005	1.32	0.32	0.24	Seismic Induced Flood	Block Wall	CDFM

1. Importance rankings obtained from SYSIMP/ACUBE output. The fragility group SEIS_HRAINSTR is a combination of fragilities rather than a single fragility. See the SHR Notebook for details.
2. The fragility parameters for SEIS_0-30-5, SEIS_0-30-15, SEIS_1A-4, and SEIS_0-1 were revised following the final quantification. These changes are addressed in sensitivity study 31 in Table 5.7-1.

The EPRI SYSIMP software was used to calculate the importance measure of each fragility group, considering the combined F-V importance across all the seismic initiator bins.

For Unit 1, the most important fragility group in Table 5.4-3 is SEIS_LOSP, which represents seismically induced LOSP. The fragility for this event is A_m = 0.3 g, which is very low compared with other events in the table. The use of this representative fragility for seismically induced LOSP is a standard industry practice for US plants. Refinement of this fragility is typically not attempted because both the switchyard and the grid outside the plant boundary would need to be considered.

The second most important fragility group is SEIS_0-30-5, which is used to model seismic relay chatter of the 480V/6.9kV shutdown board breakers. This F-V importance is high because the relay chatter prevents the emergency diesel generators (EDGs) from tying on to the shutdown boards. The importance of this group decreased with the improved fragility discussed in Sensitivity Study 31.

The third most important fragility group is SEIS_23-5, which represents the supports for the reactor pressure vessel. Failure of this group leads directly to core damage.

The fourth most important fragility group is SEIS_0-16-2, which represents a VSLOCA in an instrument line. This failure is mapped to both VSLOCA and SLOCA because multiple VSLOCAs would have the same effect as a SLOCA.

The fifth most important fragility group is SEIS_VSLOCA, which represents a VSLOCA due to small leakage in components other than instrument lines due to the seismic event.

The sixth most important fragility group is SEIS_3-4-1, which represents subgroup 1 of the 120V AC vital inverters. Failure of this group results in the failure of vital inverters for divisions 1-I, 1-II, 1-IV, 2-I, 2-II, and 2-III.

The seventh most important fragility group is SEIS_23-6. This event is the seismic failure of the steam generators. Failure of this group leads directly to core damage.

Although the F-V values of the fragility groups are slightly different for Unit 2, the same fragility groups are dominant. Fragility group SEIS_3-4-2, which represents subgroup 2 of the 120V AC vital inverters is important for Unit 2. Failure of this group results in the failure of vital inverters for divisions 1-III and 2-IV.

5.4.4 Significant Human Failure Events

The most important HFEs with respect to F-V are listed in Table 5.4-5 for Unit 1 and 5.4-6 for Unit 2. Note that the importance rankings obtained from SYSIMP/ACUBE output reflect combined importances for HFE events that vary by bin. The importance of the HFEs includes all events associated with a given HFE, including recovery and combination events. For Unit 1, there are thirteen HFEs with F-V >0.005. Failure to reset the 6.9kV Shutdown Board lockout relays and hand wheel operation of Atmospheric Relief Valves (ARVs) for Steam Generators 1&4 are the top operator actions. Actions to align the 480V FLEX diesels and to supply makeup water to the RWST via the containment spray (CS) pumps are also important actions.

For Unit 2, there are thirteen HFEs with F-V > 0.005. Many of the same actions that are important for Unit 1 are important for Unit 2, just with slightly different importance rankings.

Table 5.4-5 Risk-Significant Operator Actions for Unit 1 SCDF

Operator Action	Description	F-V
OP-LOCKOUT_69KSDB_S	Operator reset of 6900V Shutdown Board lockout relays (seismic)	3.88E-01
HAMARV	Handwheel Operation of the Steam Generator Atmospheric Relief Valves S/G 1&4	3.47E-01
HAESBODG1_S	Align 225kVA 480V Diesel Generators (seismic)	5.74E-02
HACSMU	Makeup to RWST using CS pump test recirc line from sump	2.01E-02
HASP1	Locally operate TD AFW pump after battery depletion	1.82E-02
OP-LOCKOUT_EDG_S	Operator reset of EDG start lockout relays (seismic)	1.80E-02
HINST*	Seismically induced failure of HRA MCR instrumentation	1.57E-02
HAFR2	Restore TDAFWP speed control following initiator and loss of air	1.50E-02
HAAF2	Align ERCW supply to AFW pumps	1.43E-02
HAOB3	Re-establish AFW Cooling following Loss of Vital Instrument Power Board on Unit 1	1.36E-02
HAESBO3MW_S	Align 6.9kV Diesel Generators (seismic)	1.27E-02
HARR1	Align high-pressure recirculation, given auto swap over works	6.00E-03
AFWOP1	Depressurize/cooldown to low-pressure injection following MLOCA	5.50E-03
*Although this is not an operator action but an equipment failure, it has an effect similar to that of multiple operator action failures.		

Table 5.4-6 Risk-Significant Operator Actions for Unit 2 SCDF

Operator Action	Description	F-V
OP-LOCKOUT_69KSDB_S	Operator reset of 6900V Shutdown Board lockout relays (seismic)	4.82E-01
HAMARV	Handwheel Operation of the Steam Generator Atmospheric Relief Valves S/G 1&4	4.40E-01
HAESBODG1_S	Align 225kVA 480V Diesel Generators (seismic)	2.88E-01
OP-LOCKOUT_EDG_S	Operator reset of EDG start lockout relays (seismic)	3.26E-02
HAFR2	Restore TDAFWP speed control following initiator and loss of air	2.46E-02
HAESBO3MW_S	Align 6.9kV Diesel Generators (seismic)	1.98E-02
HASP1	Locally operate TDAFW pump after battery depletion	1.85E-02
HACSMU	Makeup to RWST using CS pump test recirc line from sump	1.56E-02
HINST*	Seismically induced failure of HRA MCR instrumentation	1.51E-02
HAAF2	Align ERCW supply to AFW pumps	1.30E-02
AFWOP1	Depressurize/cooldown to low-pressure injection following MLOCA	9.90E-03

Operator Action	Description	F-V
OP-LOCKOUT_480VSDBD_S	Operator reset of 480V Shutdown Board lockout relays (seismic)	7.20E-03
HARR1	Align high-pressure recirculation, given auto swap over works	5.50E-03
*Although this is not an operator action but an equipment failure, it has an effect similar to that of multiple operator action failures.		

5.4.4.1 Summary of the Approach used to Evaluate Human Error Probabilities

The approach used to evaluate HEPs is based on EPRI 3002008093 [41]. The HFEs were identified, followed by a screening analysis and, if required, a detailed analysis to evaluate HEPs.

5.4.4.2 Screening Analysis for HEPs

EPRI 3002008093 [41] addresses the basis for developing increased HEPs due to seismic events. The choice of seismic acceleration levels for binning and applying performance shaping factors was evaluated using this basis.

Screening quantification used the analysis previously performed and applies a multiplier to the internal events HEP. The screening process produced a set of HEPs for the initial SPRA model quantification. Risk rankings based on the results of the initial quantification were used to identify risk-significant HEPs, defined as having a F-V > 0.005 or a Risk Achievement Worth (RAW) > 2.

5.4.4.3 Detailed Analysis for HEPs

Risk-significant HFEs were analyzed with detailed HRA [42], in accordance with the guidance in EPRI 3002008093 [41].

The EPRI approach for seismic HRA directs the detailed analysis of HFEs to be done in two parts: qualitative and quantitative analysis. In practice, these are done in tandem for each HFE, and the starting point for the SQN seismic HRA is the IEPRA HRA. Detailed analysis was performed for EPRI Bins 1 through 3. No detailed analysis was performed for EPRI Bin 4, as all HFEs are considered infeasible due to the damage state of this bin and the uncertainty of instrumentation availability.

5.4.4.4 Operator action credit for FLEX

FLEX actions were credited in the SPRA. However, only actions associated with the activation of the permanently installed 6.9kV and the 480V diesels were included in the model. All actions associated with portable FLEX equipment were assumed to be failed and were not modeled.

5.4.5 Significant SCDF Accident Sequences

Significant accident sequences for Unit 1 SCDF are discussed in Table 5.4-7. Table 5.4-8 discusses the significant accident sequences in the SPRA for Unit 2 SCDF. The accident sequences described account for 90% or more of the total SCDF across both units [1].

Table 5.4-7 Unit 1 SCDF Significant Accident Sequences

Accident Sequence	Description	Discussion
GTRAN-04	High-pressure recirculation with charging fails along with failure to refill the RWST with the CS pumps. Long-term makeup and cooldown with the steam generators fails.	High-pressure recirculation fails because the Chemical Volume Control System (CVCS) in recirculation mode fails. This can be due to several reasons. Either the RHR sump valves can fail due to loss of the shutdown boards due to relay chatter and failure to recover them, or the RHR pump room cooling can fail due to loss of the shutdown boards due to relay chatter and failure to recover them in the most dominant sequences. The CS pumps also fail because of loss of the shutdown boards and failure to recover them. Long-term cooldown with the steam generators via AFW fails because makeup from the CST or ERCW is not successful. This is because of flow path failures due to valves not opening because of loss of control power from the shutdown boards. Long-term makeup to the CSTs also fails, either because several valves in the TB are assumed to be failed, or because the demineralized water booster pump is assumed to be failed by the seismic event.
GTRAN-14	There is a failure to establish an RCS bleed pathway for charging pump bleed and feed. Auxiliary feedwater (AFW) fails and there is a failure to restore feedwater.	Failure of the RCS bleed pathway could be due to failure of the manual action to establish it or failure of the power-operated relief valve (PORV) bleed pathways because of seismic failure of the PORVs or loss of vital DC battery boards for control power. Feedwater is assumed to be failed for all seismic sequences so feedwater cannot be restored. Flow to the steam generators from both turbine-driven AFW (TDAFW) and motor-driven AFW (MDAFW) is lost. Actuation signals for MDAFW fail due to seismically induced logic panel failures, and the action to manually start AFW also fails. TDAFW also fails, primarily due to loss of water supply from the CST and from the ERCW headers. Failure of automatic swap over to ERCW fails, and the manual action to swap over also fails. The automatic swap over fails due to seismic failure of the vital instrument power boards due to loss of the 120V AC vital power inverters.

Table 5.4-7 Unit 1 SCDF Significant Accident Sequences

Accident Sequence	Description	Discussion
GTRAN-13	High-pressure recirculation with charging fails along with failure to refill the RWST with the CS pumps. AFW fails, and there is a failure to restore feedwater.	High-pressure recirculation fails because the CVCS in recirculation mode fails. This can be due to several reasons. Either the RHR sump valves can fail due to loss of the shutdown boards due to relay chatter and failure to recover them, or the RHR pump room cooling can fail due to loss of the shutdown boards due to relay chatter and failure to recover them in the most dominant sequences. The CS pumps also fail because of loss of the shutdown boards and failure to recover them. This could cause loss of the actuation signal due to loss of the vital inverters or loss of control power to the MOV flow path through loss of the reactor MOV (RMOV) boards. Feedwater is assumed to be failed for all seismic sequences so feedwater cannot be restored. Flow to the steam generators from both TDAFW and MDAFW is lost. Actuation signals for MDAFW fail due to seismically induced logic panel failures, and the action to manually start AFW also fails. TDAFW also fails, primarily due to loss of water supply from the CST and from the ERCW headers. Failure of automatic swap over to ERCW fails, and the manual action to swap over also fails. The automatic swap over fails due to seismic failure of the vital instrument power boards due to loss of the 120V AC vital power inverters.
GTRAN-23	The pressurizer PORVs fail. AFW fails and there is a failure to restore feedwater.	The PORVs fail because they do not receive a signal to open. This is due to a failure of the 120V AC vital instrument power boards, due to a seismic loss of the 120V AC vital inverters Feedwater is assumed to be failed for all seismic sequences so feedwater cannot be restored. Actuation signals for MDAFW fail due to seismically induced logic panel failures and the action to manually start AFW also fails. The TDAFW can fail similar to the way that it failed in sequence GTRAN-13. It can also fail because the operator fails to locally control TDAFW after battery depletion and loss of the shutdown board and backup electrical supply from the 480V FLEX diesel generator to the vital battery boards.

Table 5.4-7 Unit 1 SCDF Significant Accident Sequences

Accident Sequence	Description	Discussion
SLOCAV-19	There is a failure of cold leg injection from the centrifugal charging pumps (CCPs). Safety Injection (SI) fails. AFW fails and there is a failure to restore feedwater. The initiator is a very small LOCA (SLOCAV).	Cold leg injection fails either because the pump trains fail, or the injection pathway is lost. The injection pathway is lost because both boron injection tank isolation valves fail due to seismic loss of the 480V shutdown boards. The pump trains can also fail because of a loss of the 480V shutdown boards seismically. Both trains of SI fail because SI pump room cooling fails as a result of loss of the ventilation boards because the 480V shutdown boards fail seismically. Feedwater is assumed to be failed for all seismic sequences so feedwater cannot be restored. MDAFW fails because of inadequate cooling the MDAFW pumps. This is due to loss of the vent boards as a result of seismic failure of the 480V shutdown boards. The TDAFW system fails either because of loss of ventilation due to loss of the battery boards or because the vital instrument power boards fail due to loss of the 480V shutdown boards. Some sequences also have the vital instrument power boards failing because the vital inverters themselves fail seismically.
GTRAN-19	There is a failure of cold leg injection from the CCPs. SI fails. AFW fails and there is a failure to restore feedwater. This is similar to sequence SLOCAV-19 except that the initiator is a GTRAN rather than a SLOCAV.	Cold leg injection fails either because the pump trains fail, or the injection pathway is lost. The injection pathway is lost because both boron injection tank isolation valves fail due to seismic loss of the 480V shutdown boards. The pump trains can also fail because of a loss of the 480V shutdown boards seismically. Both trains of SI fail because SI pump room cooling fails as a result of loss of the ventilation boards because the 480V shutdown boards fail seismically. Feedwater is assumed to be failed for all seismic sequences so feedwater cannot be restored. MDAFW fails because of inadequate cooling the MDAFW pumps. This is due to loss of the vent boards as a result of seismic failure of the 480V shutdown boards. The TDAFW system fails either because of loss of ventilation due to loss of the battery boards or because the vital instrument power boards fail due

Table 5.4-7 Unit 1 SCDF Significant Accident Sequences

Accident Sequence	Description	Discussion
		to loss of the 480V shutdown boards. Some sequences also have the vital instrument power boards failing because the vital inverters themselves fail seismically.
GTRAN-18	There is a failure of cold leg injection from the CCPs. There is a failure to establish an RCS bleed path for SI bleed & feed. AFW fails and there is a failure to restore feedwater.	Cold leg injection fails either because the pump trains fail, or the injection pathway is lost. The injection pathway fails because of seismic fail of the suction valves from the RWST. The injection path could also fail in the same way as sequence SLOCAV-19. The pump trains can also fail because both the CCP pump trains fail due to a loss of the 480V shutdown boards. The RCS bleed path for SI bleed & feed fails either most likely because of failure of operator action to establish bleed & feed cooling, or because either of the PORVs fails to open. Feedwater is assumed to be failed for all seismic sequences so feedwater cannot be restored. Flow to the steam generators from both TDAFW and MDAFW is lost. Actuation signals for MDAFW fail due to seismically induced logic panel failures and the action to manually start AFW also fails. TDAFW also fails, primarily due to loss of water supply from the CST and from the ERCW headers. Failure of automatic swap over to ERCW fails, or the manual action to swap over also can fail. The automatic swap over fails due to seismic failure of the vital instrument power boards due to loss of the 120V AC vital power inverters.
GTRAN-22	The pressurizer PORVs fail. Long-term makeup and cooldown with the steam generators fails.	The PORVs fail because they do not receive a signal to open. This is due to a failure of the 120V AC vital instrument power boards, due to a seismic loss of the 120V AC vital inverters. Long-term cooldown with the steam generators via AFW fails because makeup from the CST or ERCW is not successful, or because long-term heat removal via the ARVs fails. Failure of makeup from the CST or ERCW fails because of flow path failures due to valves not opening because of loss of control power from the shutdown boards. Long-term makeup to the CSTs also fails, either

Table 5.4-7 Unit 1 SCDF Significant Accident Sequences

Accident Sequence	Description	Discussion
		because several valves in the TB are assumed to be failed, or because the demineralized water booster pump is assumed to be failed by the seismic event. Long-term heat removal with the ARVs fails because of loss of the 120V vital inverters along with operator backup action failure to manually open the ARVs.
SLOCA-27	There is a failure of cold leg injection from the CCPs. Safety injection fails. RHR low-pressure injection fails.	Cold leg injection fails either because the pump trains fail, or the injection pathway is lost. The injection pathway fails because of seismic fail of the suction valves from the RWST. The injection path could also fail in the same way as with sequence SLOCAV-19. The pump trains can also fail because both the CCP pump trains fail due to a loss of the 480V shutdown boards. Both trains of SI fail because SI pump room cooling fails as a result of loss of the ventilation boards because the 480V shutdown boards fail seismically. The SI pumps could also fail in this sequence because the shutdown boards fail due to a seismic flood event, or because ERCW fails due to a flood event. The SI pumps can also fail in this sequence similar to the way they do in sequence SLOCAV-19 although this is not one of the dominant failure modes for this sequence. RHR low-pressure injection fails because room cooling to the RHR pumps fails, either because ERCW fails or the ventilation board fails due to loss of the shutdown boards. In the dominant cutsets this is due to seismically induced flooding but could also be due to direct seismically induced loss of the shutdown boards.
SLOCA-05	High-pressure recirculation with charging fails. Low-pressure recirculation with RHR fails. There is a failure to refill the RWST with the CS pumps.	High-pressure recirculation with charging fails because of failure of the CVCS. In the dominant sequences, this is due to loss of RHR flow to the CVCS pumps. In the dominant sequence, RHR pump A is lost when DC control power from the vital battery or charger is lost. The vital battery is lost due to battery depletion while the charger is lost due to seismic failures. RHR pump B is lost because RHR heat exchanger cooling is lost ultimately due to seismic failure of the vital inverter, which causes failure of solid-state protection system (SSPS)

Table 5.4-7 Unit 1 SCDF Significant Accident Sequences

Accident Sequence	Description	Discussion
		actuation signals, which causes the Common Spare (C-S) component cooling pump not to start. The ERCW flow path to the discharge header is also lost due to loss of SSPS from failure of the vital inverters Low-pressure recirculation with RHR fails for the same reasons discussed above for failure of the RHR support to the CVCS. Failure to refill the RWST occurs because of loss of both CS trains. CS train A is lost because the battery or charger fails for the same reasons as those discussed for loss of high-pressure recirculation above. Train B is lost because the C-S component cooling pump does not start for the same reasons as those discussed above for loss of high-pressure recirculation. Train B can also be lost because the component cooling system (CCS) heat exchangers are lost, which is ultimately caused by loss of the SSPS signals as discussed above. Failure to refill the RWST can also occur because of operator failure to align makeup to the RWST or through seismic failure of the RWST itself.
SLOCAV-09	There is failure of cold leg injection from the CCPs. Safety Injection fails. AFW depressurization using the ARVs fails.	Some of the cutsets in this sequence are caused by a flood in the ACB. Cold leg injection fails either because the pump trains fail, or the injection pathway is lost. The injection path could fail in the same way as with sequence SLOCAV-19, or both valves could fail to open because of loss of control power from ventilation boards and the shutdown boards due to a flood. The pump trains can also fail because both fail due to a loss of the 480V shutdown boards directly from the seismic event. In the dominant cutsets, the SI trains fail either due to pump support system failure due to a seismic loss of the shutdown boards, or loss of pump cooling when the ventilation boards power supply from the

Table 5.4-7 Unit 1 SCDF Significant Accident Sequences

Accident Sequence	Description	Discussion
		shutdown boards is lost, either directly from the seismic event or due to seismically induced flooding. AFW depressurization fails because the steam generator ARVs fail. This can be related to flood events or failure of operator manual action to operate the valves. The auxiliary control air subsystem (ACAS), which supplies control air to the ARVs, can fail through loss of the shutdown boards by seismic flooding or directly from the seismic event

Table 5.4-8 Unit 2 SCDF Significant Accident Sequences

Accident Sequence	Description	Discussion
GTRAN-04	High-pressure recirculation with charging fails along with failure to refill the RWST with the CS pumps. Long-term makeup and cooldown with the steam generators fails.	High-pressure recirculation fails because the CVCS in recirculation mode fails. This can be due to several reasons. Either the RHR sump valves can fail due to loss of the shutdown boards due to relay chatter and failure to recover them, or the RHR pump room cooling can fail due to loss of the shutdown boards due to relay chatter and failure to recover them in the most dominant sequences. The CS pumps also fail because of loss of the shutdown boards and failure to recover them. Long-term cooldown with the steam generators via AFW fails because makeup from the CST or ERCW is not successful. This is because of flow path failures due to valves not opening because of loss of control power from the shutdown boards. Long-term makeup to the CSTs also fails, either because several valves in the TB are assumed to be failed, or because the demineralized water booster pump is assumed to be failed by the seismic event.
GTRAN-13	High-pressure recirculation with charging fails along with failure to refill the RWST with the CS pumps. AFW fails and there is a failure to restore feedwater.	High-pressure recirculation fails because the CVCS in recirculation mode fails. This can be due to several reasons. Either the RHR sump valves can fail due to loss of the shutdown boards due to relay chatter and failure to recover them, or the RHR pump room cooling can fail due to loss of the shutdown boards due to relay chatter and failure to recover them in the most dominant sequences. The CS pumps also fail because of loss of the shutdown boards and failure to recover them. This could cause loss of the actuation signal due to loss of the vital inverters or loss of control power to the MOV flow path through loss of the RMOV boards. Feedwater is assumed to be failed for all seismic sequences so feedwater cannot be restored. Flow to the steam generators from both TDAFW and MDAFW is lost. Actuation signals for MDAFW fail due to seismically induced logic panel failures and the action to manually start AFW also fails. TDAFW also fails, primarily due to loss

Table 5.4-8 Unit 2 SCDF Significant Accident Sequences

Accident Sequence	Description	Discussion
		of water supply from the CST and from the ERCW headers. Failure of automatic swap over to ERCW fails, and the manual action to swap over also fails. The automatic swap over fails due to seismic failure of the vital instrument power boards due to loss of the 120V AC vital power inverters.
GTRAN-22	The pressurizer PORVs fail. Long-term makeup and cooldown with the steam generators fails.	The PORVs fail because they do not receive a signal to open. This is due to a failure of the 120V AC vital instrument power boards, due to a seismic loss of the 120V AC vital inverters. Long-term cooldown with the steam generators via AFW fails because makeup from the CST or ERCW is not successful, or because long-term heat removal via the ARVs fails. Failure of makeup from the CST or ERCW fails because of flow path failures due to valves not opening because of loss of control power from the shutdown boards. Long-term makeup to the CSTs also fails, either because several valves in the turbine building are assumed to be failed, or because the demineralized water booster pump is assumed to be failed by the seismic event. Long-term heat removal with the ARVs can also fail because of loss of the 120V vital inverters along with operator backup action failure to manually open the ARVs.
SLOCAV-19	There is a failure of cold leg injection from the CCPs. SI fails. AFW fails and there is a failure to restore feedwater.	Cold leg injection fails either because the pump trains fail, or the injection pathway is lost. The injection pathway is lost because both boron injection tank isolation valves fail due to seismic loss of the 480V shutdown boards. The pump trains can also fail due to a loss of the 480V shutdown boards. Both trains of SI fail because SI pump room cooling fails as a result of loss of the ventilation boards because the 480V shutdown boards fail seismically. Feedwater is assumed to be failed for all seismic sequences so feedwater cannot be restored. MDAFW fails because of inadequate cooling the MDAFW pumps. This is due to loss of the vent boards as a result of seismic failure of the 480V shutdown boards. The TDAFW system fails either because of loss of ventilation due to loss of the

Table 5.4-8 Unit 2 SCDF Significant Accident Sequences

Accident Sequence	Description	Discussion
		battery boards or because the vital instrument power boards fail due to loss of the 480V shutdown boards. Some sequences also have the vital instrument power boards failing because the vital inverters fail seismically.
GTRAN-10	There is a failure of cold leg injection from the CCPs. SI fails. Long-term makeup and cooldown with the steam generators fails.	Cold leg injection fails because both CCP pump trains fail. Seismic loss of the 480V shutdown boards can affect the pump trains either through loss of room cooling via ERCW or the pump trains can also fail due to loss of the injection pathway similar to the way they fail for the SLOCAV-19 sequence. The pump trains can also fail due to a loss of the 480V shutdown boards. Both trains of SI fail because SI pump room cooling fails as a result of loss of the ventilation boards because the 480V shutdown boards fail directly seismically or due to relay chatter. Loss of the 480V shutdown boards along with failure of the FLEX diesels either seismically or through failure of the operator action to align them can also fail the SI pumps due to loss of ERCW cooling to them. Long-term cooldown with the steam generators via AFW fails because makeup from the CST or ERCW is not successful, or because long-term heat removal via the ARVs fails. Failure of makeup from the CST or ERCW fails because of flow path failures due to valves not opening because of loss of control power from the shutdown boards. Long-term makeup to the CSTs also fails, either because several valves in the turbine building are assumed to be failed, or because the demineralized water booster pump is assumed to be failed by the seismic event. Long-term heat removal with the ARVs can also fail because of loss of the 120V vital inverters along with operator backup action failure to manually open the ARVs.
GTRAN-23	The pressurizer PORVs fail. AFW fails and there is a failure to restore feedwater.	The PORVs fail because they do not receive a signal to open. This is due to a failure of the 120V AC vital instrument power boards due to a seismic loss of the 120V AC vital inverters Feedwater is assumed to be failed for all seismic sequences so feedwater cannot be restored. Actuation signals for MDAFW fail due

Table 5.4-8 Unit 2 SCDF Significant Accident Sequences

Accident Sequence	Description	Discussion
		to seismically induced logic panel failures and the action to manually start AFW also fails. The TDAFW can fail similar to the way that it failed in sequence GTRAN-13. It can also fail because the operator fails to locally control TDAFW after battery depletion and loss of the shutdown board and backup electrical supply from the 480V FLEX DG to the vital battery boards.
GTRAN-19	There is a failure of cold leg injection from the CCPs. SI fails. AFW fails and there is a failure to restore feedwater.	Cold leg injection fails because both CCP pump trains fail. Seismic loss of the 480V shutdown boards can affect the pump trains either through loss of room cooling via ERCW or the pump trains can also fail due to loss of the injection pathway similar to the way they fail for the SLOCAV-19 sequence. The pump trains can also fail due to a loss of the 480V shutdown boards. SI can fail either through seismic loss of the RWST seismically or through the other ways described (for example in SLOCAV-19 or GTRAN-10). Feedwater is assumed to be failed for all seismic sequences so feedwater cannot be restored. MDAFW can fail similar to the way it does in GTRAN-13 or GTRAN-23. TDAFW can also fail similar to the way it does in those sequences.
GTRAN-14	There is a failure to establish an RCS bleed pathway for charging pump bleed and feed. AFW fails and there is a failure to restore feedwater.	Failure of the RCS bleed pathway could be due to failure of the manual action to establish it or failure of the PORV bleed pathways because of seismic failure of the PORVs or loss of vital DC battery boards for control power. Feedwater is assumed to be failed for all seismic sequences so feedwater cannot be restored. Flow to the steam generators from both TDAFW and MDAFW is lost. Actuation signals for MDAFW fail due to seismically induced logic panel failures and the action to manually start AFW also fails. TDAFW also fails, primarily due to loss of water supply from the CST and from the ERCW headers. Failure of automatic swap over to ERCW fails, and the manual action to swap over also fails. The automatic swap over fails due to seismic

Table 5.4-8 Unit 2 SCDF Significant Accident Sequences

Accident Sequence	Description	Discussion
		failure of the vital instrument power boards due to loss of the 120V AC vital power inverters.
GTRAN-18	There is a failure of cold leg injection from the CCPs. There is a failure to establish an RCS bleed path for SI bleed & feed. AFW fails and there is a failure to restore feedwater.	Cold leg injection fails either because the pump trains fail, or the injection pathway is lost. The injection pathway fails because of seismic fail of the suction valves from the RWST. The injection path could also fail in the same way as with sequence SLOCAV-19. The pump trains can also fail because both the CCP pump trains fail due to a loss of the 480V shutdown boards. The RCS bleed path for SI bleed & feed fails either most likely because of failure of operator action to establish bleed and feed cooling, or because either of the PORVs fails to open. Feedwater is assumed to be failed for all seismic sequences so feedwater cannot be restored. Flow to the steam generators from both TDAFW and MDAFW is lost. Actuation signals for MDAFW fail due to seismically induced logic panel failures and the action to manually start AFW also fails. TDAFW also fails, primarily due to loss of water supply from the CST and from the ERCW headers. Failure of automatic swap over to ERCW fails, or the manual action to swap over also can fail. The automatic swap over fails due to seismic failure of the vital instrument power boards due to loss of the 120V AC vital power inverters.
SLOCA-27	There is a failure of cold leg injection from the CCPs. Safety injection fails. RHR low-pressure injection fails.	Cold leg injection fails either because the pump trains fail, or the injection pathway is lost. The injection pathway fails because of seismic fail of the suction valves from the RWST. The injection path could also fail in the same way as with sequence SLOCAV-19. The pump trains can also fail because both the CCP pump trains fail due to a loss of the 480V shutdown boards. Both trains of SI fail because SI pump room cooling fails as a result of loss of the ventilation boards because the 480V shutdown boards fail seismically. The SI pumps could also fail in this sequence because the shutdown boards fail due to a seismic flood event, or because ERCW fails due to a flood event. The SI pumps can also fail

Table 5.4-8 Unit 2 SCDF Significant Accident Sequences

Accident Sequence	Description	Discussion
		in this sequence similar to the way they do in sequence SLOCAV-19 although this is not one of the dominant failure modes for this sequence. RHR low-pressure injection fails because room cooling to the RHR pumps fails, either because ERCW fails or the ventilation board fails due to loss of the shutdown boards. In the dominant cutsets this is due to seismically induced flooding but could also be due to direct seismically induced loss of the shutdown boards.
SLOCA-05	High-pressure recirculation with charging fails. Low-pressure recirculation with RHR fails. There is a failure to refill the RWST with the CS pumps.	High-pressure recirculation with charging fails because of failure of the CVCS. In the dominant sequences, this is due to loss of RHR flow to the CVCS pumps. In the dominant sequence RHR pump A is lost when DC control power from the vital battery or charger is lost. The vital battery is lost due to battery depletion while the charger is lost due to seismic failures. RHR pump B is lost because RHR heat exchanger cooling is lost ultimately due to seismic failure of the vital inverter, which causes failure of SSPS actuation signals, which causes the C-S component cooling pump not to start. The ERCW flow path to the discharge header is also lost due to loss of SSPS from failure of the vital inverters Low-pressure recirculation with RHR fails for the same reasons discussed above for failure of the RHR support to the CVCS. Failure to refill the RWST occurs because of loss of both CS trains. CS train A is lost because the battery or charger fails for the same reasons as those discussed for loss of high-pressure recirculation above. Train B is lost because the C-S component cooling pump does not start for the same reasons as those discussed above for loss of high-pressure recirculation. Train B can also be lost because the CCS heat exchangers are lost, which is ultimately caused by loss of the SSPS signals as discussed above. Failure to refill the RWST can also occur because of operator failure to align makeup to the RWST.

Table 5.4-8 Unit 2 SCDF Significant Accident Sequences

Accident Sequence	Description	Discussion
SLOCAV-09	There is failure of cold leg injection from the CCPs. Safety Injection fails. AFW depressurization using the ARVs fails.	Some of the cutsets in this sequence are caused by a flood in the ACB. Cold leg injection fails either because the pump trains fail, or the injection pathway is lost. The injection path could fail in the same way as with sequence SLOCAV-19, or both valves could fail to open because of loss of control power from ventilation boards and the shutdown boards due to a flood. The pump trains can also fail because both fail due to a loss of the 480V shutdown boards directly from the seismic event. In the dominant cutsets, the SI trains fail either due to pump support system failure due to a seismic loss of the shutdown boards, or loss of pump cooling when the ventilation boards power supply from the shutdown boards is lost either directly from the seismic event or due to seismically induced flooding. AFW depressurization fails because the steam generator ARVs fail. This can be related to flood events or failure of operator manual action to operate the valves. The ACAS system, which supplies control air to the ARVs, can fail through loss of the shutdown boards by seismic flooding or directly from the seismic event

5.5 SLERF Results

5.5.1 Overall SLERF

The SPRA performed for SQN shows that the point-estimate mean SLERF is 2.6E-06 per ry for Unit 1 and 2.4E-06 per ry for Unit 2 [1].

5.5.2 SLERF as a Function of Hazard Interval

A summary of the SLERF results for each seismic hazard interval is presented in Table 5.5-1 for Unit 1 SLERF and Table 5.5-2 for Unit 2 SLERF. The maximum CCDP is 0.908 for Unit 1 and 0.932 for Unit 2. These are equal to the plant availability factors for each unit, respectively.

Table 5.5-1 Unit 1 SLERF Contribution by Initiating Event

Truncation	Scenario	Description	Earthquake Frequency	CLERP	SLERF	Percent Contribution
5.E-15	%G01	Seismic Initiating Event (0.09g to <0.18g)	3.97E-04	1.44E-06	5.71E-10	0.0%
1.E-14	%G02	Seismic Initiating Event (0.18g to <0.3g)	1.24E-04	2.76E-05	3.42E-09	0.1%
1.E-12	%G03	Seismic Initiating Event (0.3g to <0.5g)	5.39E-05	1.85E-03	9.96E-08	3.9%
1.E-11	%G04	Seismic Initiating Event (0.5g to <0.7g)	1.49E-05	2.42E-02	3.61E-07	14.1%
5.E-09	%G05	Seismic Initiating Event (0.7g to <0.9g)	5.50E-06	4.28E-02	2.35E-07	9.2%
8.E-09	%G06	Seismic Initiating Event (0.9g to <1.1g)	2.40E-06	1.46E-01	3.51E-07	13.8%
5.E-08	%G07	Seismic Initiating Event (1.1g to <1.5g)	1.81E-06	3.04E-01	5.50E-07	21.5%
5.E-07	%G08	Seismic Initiating Event (1.5g to <3g)	9.67E-07	8.97E-01	8.68E-07	34.0%
1.E-08	%G09	Seismic Initiating Event (>3g)	9.24E-08	9.08E-01	8.39E-08	3.3%
				Total SLERF=	**2.6E-06**	

Table 5.5-2 Unit 2 SLERF Contribution by Initiating Event

Truncation	Scenario	Description	Earthquake Frequency	CLERP	SLERF	Percent Contribution
5.E-15	%G01	Seismic Initiating Event (0.09g to <0.18g)	3.97E-04	8.22E-07	3.26E-10	0.0%
1.E-14	%G02	Seismic Initiating Event (0.18g to <0.3g)	1.24E-04	2.63E-05	3.26E-09	0.1%
1.E-12	%G03	Seismic Initiating Event (0.3g to <0.5g)	5.39E-05	9.63E-04	5.19E-08	2.1%
1.E-11	%G04	Seismic Initiating Event (0.5g to <0.7g)	1.49E-05	2.48E-02	3.69E-07	15.1%
5.E-09	%G05	Seismic Initiating Event (0.7g to <0.9g)	5.50E-06	4.35E-02	2.39E-07	9.8%
2.E-08	%G06	Seismic Initiating Event (0.9g to <1.1g)	2.40E-06	8.26E-02	1.98E-07	8.1%
5.E-08	%G07	Seismic Initiating Event (1.1g to <1.5g)	1.81E-06	3.32E-01	6.01E-07	24.6%
5.E-07	%G08	Seismic Initiating Event (1.5g to <3g)	9.67E-07	9.21E-01	8.90E-07	36.5%
1.E-08	%G09	Seismic Initiating Event (>3g)	9.24E-08	9.32E-01	8.61E-08	3.5%
				Total SLERF=	**2.4E-06**	

5.5.3 Significant Systems, Structures, and Components for SLERF

The SSCs with the most significant seismic failure contributions to SLERF for Unit 1 are listed in Table 5.5-3, sorted by F-V. The seismic fragilities for each of the significant contributors are also provided in Table 5.5-3, along with the corresponding limiting seismic failure mode and method of fragility calculation. The corresponding measures for Unit 2 are presented in Table 5.5-4.

Table 5.5-3 Unit 1 SLERF Importance Measures Ranked by F-V [1]

Fragility Group	F-V	Fragility					
		A_m (g)	βu	βr	Description	Failure Mode	Fragility Method
SEIS_LOSP	0.397	0.3	0.43	0.19	LOSP (Loss of Offsite Power)	Ceramic insulators	Table 6-1 NUREG/CR-6544
SEIS_0-30-5	0.242	0.84 [2]	0.32	0.24	Relay Chatter - 480V/6.9kV SD BD GE HEA Breakers	Functionality	CDFM
SEIS_1A-5	0.133	1.4	0.32	0.24	480V RMOV Board	Block wall	CDFM
SEIS_23-5	0.118	1.79	0.26	0.24	NSSS Reactor Pressure Vessel	Anchorage	CDFM
SEIS_HINST	0.093	1.32	0.32	0.24	Seismically induced failure of HRA MCR Instrumentation	Functionality	CDFM
SEIS_0-16-2	0.077	2.04	0.32	0.24	Instrument Line Very Small LOCA	Anchorage	CDFM
SEIS_3-4-1	0.061	0.8	0.38	0.24	120V AC Vital Inverter in Aux. sub 1	Anchorage	CDFM
SEIS_23-6A	0.041	2.24	0.26	0.24	Steam Generator Support Failure	Anchorage	CDFM
SEIS_4-3	0.04	1	0.38	0.24	PHMS Power Transformer	Anchorage	CDFM
SEIS_0-30-1	0.032	1.3	0.3	0.23	Relay Chatter - SSPS Cabinets (R-048, R-051)	Functionality	SoV
SEIS_19-8	0.018	1.24	0.26	0.24	EDG Starting Air Tank	Anchorage	CDFM
SEIS_0-30-6	0.018	0.63	0.32	0.24	Relay Chatter - 480V/6.9kV SD BD, Pumps, Misc. Breakers	Functionality	CDFM
SEIS_VSLOCA	0.016	2.04	0.32	0.24	Seismic Very Small LOCA	Anchorage	CDFM
SEIS_0-30-11	0.008	0.71	0.32	0.24	Relay Chatter - DGB Control Panel Kraus & Naimer relays	Functionality	CDFM
SEIS_0-1	0.007	1.73	0.43 [2]	0.19 [2]	Buried ERCW piping	Soil (Buried Piping) Failure	SoV
SEIS_19-9	0.006	1.04	0.26	0.24	RHR Heat Exchanger	Anchorage	CDFM
SEIS_19-5	0.006	1.11	0.24	0.12	Refueling Water Storage Tank (RWST)	Tank Fragility	SoV

1. Importance rankings obtained from SYSIMP/ACUBE output. The fragility group SEIS_HINST is a combination of fragilities rather than a single fragility. See the SHR Notebook for details.
2. The fragility parameters for SEIS_0-30-5, SEIS_0-30-15, SEIS_1A-4, and SEIS_0-1 were revised following the final quantification. These changes are addressed in sensitivity study 31 in Table 5.7-1.

Table 5.5-4 Unit 2 SLERF Importance Measures Ranked by F-V [1]

Fragility Group	F-V	Fragility					
		A_m (g)	βu	βr	Description	Failure Mode	Fragility Method
SEIS_LOSP	0.413	0.3	0.43	0.19	LOSP (Loss of Offsite Power)	Ceramic insulators	Table 6-1 NUREG/CR-6544
SEIS_0-30-5	0.289	0.84 [2]	0.32	0.24	Relay Chatter - 480V/6.9kV SD BD GE HEA Breakers	Functionality	CDFM
SEIS_1A-5	0.106	1.4	0.32	0.24	480V RMOV Board	Block Wall	CDFM
SEIS_23-5	0.103	1.79	0.26	0.24	NSSS Reactor Pressure Vessel	Anchorage	CDFM
SEIS_HINST	0.091	1.32	0.32	0.24	Seismically induced failure of HRA MCR Instrumentation	Functionality	CDFM
SEIS_3-4-1	0.065	0.8	0.38	0.24	120V AC Vital Inverter in Aux. sub 1	Anchorage	CDFM
SEIS_0-16-2	0.059	2.04	0.32	0.24	Instrument Line Very Small LOCA	Anchorage	CDFM
SEIS_0-30-1	0.045	1.3	0.3	0.23	Relay Chatter - SSPS Cabinets (R-048, R-051)	Functionality	SoV
SEIS_23-6A	0.043	2.24	0.26	0.24	Steam Generator Support Failure	Anchorage	CDFM
SEIS_3-4-2	0.033	0.8	0.38	0.24	120V AC Vital Inverter in Aux. sub 2	Anchorage	CDFM
SEIS_4-3	0.033	1	0.38	0.24	PHMS Power Transformer	Anchorage	CDFM
SEIS_VSLOCA	0.023	2.04	0.32	0.24	Seismic Very Small LOCA	Anchorage	CDFM
SEIS_0-30-6	0.013	0.63	0.32	0.24	Relay Chatter - 480V/6.9kV SD BD, Pumps, Misc. Breakers	Functionality	CDFM
SEIS_19-8	0.010	1.24	0.26	0.24	EDG Starting Air Tank	Anchorage	CDFM
SEIS_19-9	0.006	1.04	0.26	0.24	RHR Heat Exchanger	Anchorage	CDFM
SEIS_19-5	0.006	1.11	0.24	0.12	Refueling Water Storage Tank (RWST)	Tank Fragility	SoV
SEIS_0-30-11	0.005	0.71	0.32	0.24	Relay Chatter - DGB Control Panel Kraus & Naimer relays	Functionality	CDFM

1. Importance rankings obtained from SYSIMP/ACUBE output. The fragility group SEIS_HINST is a combination of fragilities rather than a single fragility. See the SHR Notebook for details.
2. The fragility parameters for SEIS_0-30-5, SEIS_0-30-15, SEIS_1A-4, and SEIS_0-1 were revised following the final quantification. These changes are addressed in sensitivity study 31 in Table 5.7-1.

LOSP represents the most significant contributor, which is consistent with the results of previous SPRA studies across the nuclear industry that have found that extended LOSP events are dominant for seismic risk. The two fragility groups with the highest F-V values excluding LOSP are 1A-5 (480V RX MOV BD), SEIS_0-30-5 (Relay Chatter - 480V/6.9kV SD BD GE HEA Breakers) for Unit 1, and SEIS_0-30-5 and SEIS_1A-5 for Unit 2. SEIS_1A-5 is an important fragility group because it supports many other electrical dependencies associated with ensuring closure of containment isolation valves. SEIS_0-30-5 is a relay-chatter-based failure that is assumed to fail the EDGs. The importance of SEIS_0-30-5 decreased slightly with the improved fragility discussed in Sensitivity Study 31 (See Table 5.7-1). The 6.9kV shutdown boards are an important source of power to numerous systems throughout the plant, and their loss has wide-ranging plant effects. The EDGs are important because the offsite power fragility has a 0.1g HCLPF; about 40% to 41% of SLERFs involve scenarios where all offsite power is lost. In addition to losing power for all injection pumps, failure of the EDGs results in battery depletion and loss of AFW due to steam generator overfilling (flow control valves fail open). AFW is important because seismic events typically fail all other steam generator decay heat removal options due to LOSP.

5.5.4 Significant Human Failure Events

According to the ASME/ANS PRA Standard [8], significant post-initiator operator actions are defined as those operator action basic events that have a F-V value greater than 0.005 or a RAW greater than 2. Note that the common methods of calculating RAW for basic events will not yield useful results for the HRA events, due to the processing of combination events. This is because the events are set to one during the quantification process and a recovery event representing the combination or single event is appended to the cutset. Therefore, setting the event to one to determine the RAW value has no effect on SLERF. The F-V values of each operator action were determined in SYSIMP by defining groups where each operator action appearing in each seismic HRA bin (S1, S2, S3 and S4) were simultaneously set to false in the combined cutset file to determine the combined importance across all seismic bins. The most important HFEs with respect to SLERF based on F-V are listed in Table 5.5-5 (Unit 1) and Table 5.5-6 (Unit 2) [1]. The importance of the HFEs includes all events associated with a given HFE, including recovery and combination events.

Table 5.5-5 Risk-Significant Operator Actions for Unit 1 SLERF

Operator Action	Description	F-V
HAMARV	Handwheel Operation of the Steam Generator Atmospheric Relief Valves S/G 1&4	2.90E-01
OP-LOCKOUT_69KSDB_S	Operator reset of 6900V Shutdown Board lockout relays (seismic)	2.65E-01
HACIV	Isolate RCP seal water and thermal barrier injection and return lines on a Station Blackout (SBO)	1.90E-01
HAESBODG1_S	Align 225kVA 480V Diesel Generators (seismic)	1.84E-01
HINST*	Seismically induced failure of HRA MCR instrumentation	9.33E-02
HAESBO3MW_S	Align 6.9kV Diesel Generators (seismic)	1.41E-02
OP-LOCKOUT_EDG_S	Operator reset of EDG start lockout relays (seismic)	1.19E-02
HART1	Manually trip reactor, given SSPS fails	6.90E-03
AFWOP1	Depressurize/cooldown to low-pressure injection following MLOCA	6.40E-03
HAAF2	Align ERCW supply to AFW pumps	5.20E-03
HAOB3	Re-establish AFW Cooling following Loss of Vital Instrument Power Board on Unit 1	5.00E-03
*Although this is not an operator action but an equipment failure, it has an effect similar to that of multiple operator action failures.		

Table 5.5-6 Risk-Significant Operator Actions for Unit 2 SLERF

Operator Action	Description	F-V
OP-LOCKOUT_69KSDB_S	Operator reset of 6900V Shutdown Board lockout relays (seismic)	3.02E-01
HAMARV	Handwheel Operation of the Steam Generator Atmospheric Relief Valves S/G 1&4	2.88E-01
HAESBODG1_S	Align 225kVA 480V Diesel Generators (seismic)	2.54E-01
HACIV	Isolate RCP seal water and thermal barrier injection and return lines on an SBO	1.50E-01
HINST*	Seismically induced failure of HRA MCR instrumentation	9.10E-02
HAPRZ	Depressurization of the RCS using pressurizer PORVs (Level II only)	1.92E-02
HAESBO3MW_S	Align 6.9kV Diesel Generators (seismic)	8.00E-03
OP-LOCKOUT_EDG_S	Operator reset of EDG start lockout relays (seismic)	7.90E-03
HASP1	Locally operate TD AFW pump after battery depletion	5.90E-03
HAFR2	Restore TDAFWP speed control following initiator and loss of air	5.80E-03
HAAF2	Align ERCW supply to AFW pumps	5.40E-03
*Although this is not an operator action but an equipment failure, it has an effect similar to that of multiple operator action failures.		

5.5.5 Significant SLERF Accident Sequences

Significant accident sequences for Unit 1 SLERF are discussed in Table 5.5-7. Table 5.5-8 discusses the significant accident sequences in the SPRA for Unit 2 SLERF. The accident sequences described account for 90% or more of the total SLERF across both units [1].

Table 5.5-7 Unit 1 SLERF Significant Accident Sequences

Accident Sequence	Description	Discussion
ILERF-001	This sequence involves a containment isolation failure of >=2". It is fed from one of the following sequences core damage sequences from the Level 1 model: 1) no containment bypass, high pressure, dry steam generator 2) no containment bypass, low pressure, wet steam generator or 3) no bypass, high pressure, wet steam generator).	The dominant scenario for this accident sequence involves failure of the RCP seal return isolation penetration (X-44). For this to occur, both valve FCV-62-63 and either valve FCV-62-61 or check valve VLV-62-639 must fail. FCV-62-63 fails primarily because of failure of the 480V RMOV boards and failure of operator action to isolate the RCP seal water and thermal barrier injection and return lines. The 480V RMOV board fails either by seismic functional failure, block wall impact or seismic anchorage failure. The 480V RMOV board can also be lost if the 480V shutdown board is lost. This can happen due to loss of power from the 6.9kV shutdown boards or seismically induced floods. The backup 480V FLEX DG must also fail in order to cause this scenario with the most predominant 480V FLEX DG failure being failure to align it. The most dominant failure of FCV-62-61 is caused by loss of the 480V RMOV boards due to reasons similar to those discussed above for the FCV-62-63 failures. Since 62-639 is a check valve, it is rugged and is not likely to experience any seismic impacts.
LATE-034	This is a late-release sequence that is considered early release for 0.5g earthquakes and above. In this sequence, the Containment Air Return Fans (CARFs) and hydrogen igniters are available. Containment Heat Removal (CHR) fails. It is fed by all the NHD core damage sequences from the Level 1 model.	The SQN IEPRA Level 2 model does not explicitly identify which large, late releases occur beyond 24 hours. As discussed in the Methodology, Modelling and Inputs notebook, to estimate the probability that containment fails before 24 hours, a conditional probability of containment failure before 24 hours assuming the ice condensers are available but CHR fails was derived for the seismic model that is included in the fault tree for this sequence. In order to fail CHR, either the containment sump must fail due to plugging or operator failure prior to the initiating event, which is unlikely, or CS in injection mode along with ice condenser failure must occur, or CS in recirculation mode along with RHR spray failure must happen. CS can fail either through failure of the operator action to transfer CS to the sump or by mechanical failure of the CS pumps or their flow paths. Direct seismic failure of the pumps or their coolers is unlikely because these components are fairly robust seismically. Loss of the 480V ventilation boards (because of loss of the 480V shutdown boards as a result of relay chatter of the 6.9kV shutdown boards that support them) will cause room cooling for the pumps to fail, which is one mechanism

Table 5.5-7 Unit 1 SLERF Significant Accident Sequences

Accident Sequence	Description	Discussion
		that can fail the pumps. The pumps can also fail because of other support system failures, such as loss of the component cooling heat exchangers because of MOV failures in the flow path of the cooling to the heat exchangers caused by loss of the 480V RMOV boards. RHR spray can fail through loss of the RHR heat exchangers, which can happen in the same way as previously discussed for cooling to the CS pumps. It can also occur through operator failure to start RHR spray or seismically induced flooding.
BLERF-001	This sequence involves a thermally-induced steam generator tube rupture (TI-SGTR) and failure to depressurize the RCS. It is fed by all the NHD core damage sequences from the level 1 model.	A TI-SGTR occurs in this sequence. The RCS is not depressurized because one of the two PORVs does not open. This could be due to seismically induced failure of the PORVs to open or failure of operator to depressurize. Seismic failure of the PORVs themselves could occur but is unlikely due their high fragility. The most dominant cause of PORV failure is loss of DC control power because the 125V batteries and the chargers fail. The batteries could fail seismically, or they could fail due to battery depletion. They could fail due to a seismically induced flood, but the piping is rugged, so this is not likely. The chargers could fail if normal power from the 480V shutdown board fails because of loss of the board due to relay chatter and the backup power from the 480V FLEX DG is lost. The most likely reason the power from the 480V FLEX DG is lost is because of operator alignment failure.
LLERF-017	In this sequence, the CARFs fail, the hydrogen igniters fail, and the operator is successful in RCS depressurization. It is fed by all the NHD core damage sequences from the Level 1 model.	Both trains of the CARFs fail and both trains of hydrogen igniters fail. The hydrogen igniters are rugged so they are not likely to fail seismically; however, they could both fail due to operator failure to place them in service, although this is a fairly low-probability event. The primary way the hydrogen igniters fail is either though seismic loss of their power transformers or through loss of the 480V ventilation boards due to loss of the shutdown board itself or through relay chatter due to the seismic event. The CARFs could fail due to seismic failure of the fans themselves or through loss of power from the 480V shutdown boards caused by direct seismic failure of the board or more likely relay chatter. They could also fail because of failure of a containment isolation signal because the shutdown boards fail due to relay chatter along with failure of the 480V FLEX DGs.
LLERF-014	In this sequence, the CARFs are available, the hydrogen igniters fail, and the operator is	The hydrogen igniters are rugged so they are not likely to fail seismically; however, they could both fail due to operator failure to place them in service, although this is

Table 5.5-7 Unit 1 SLERF Significant Accident Sequences

Accident Sequence	Description	Discussion
	successful in RCS depressurization. It is fed by all the NHD core damage sequences from the Level 1 model.	a fairly low-probability event. The primary way the hydrogen igniters fail is either though seismic loss of their power transformers or through loss of the 480V ventilation boards due to loss of the shutdown board itself or through relay chatter due to the seismic event.

Table 5.5-8 Unit 2 SLERF Significant Accident Sequences

Accident Sequence	Description	Discussion
ILERF-001	This sequence involves a containment isolation failure of >=2". It is fed by all the NHD (no containment bypass, high pressure, dry steam generator) NLW (no containment bypass, low pressure, wet steam generator) or NHW (no bypass, high pressure, wet steam generator) core damage sequences from the Level 1 model.	The dominant scenario for this accident sequence involves failure of the RCP seal return isolation penetration (X-44). For this to occur, both valve FCV-62-63 and either valve FCV-62-61 or check valve VLV-62-639 must fail. FCV-62-63 fails primarily because of failure of the 480V RMOV boards and failure of operator action to isolate the RCP seal water and thermal barrier injection and return lines. The 480V RMOV board fails either by seismic functional failure, block wall impact or seismic anchorage failure. The 480V RMOV board can also be lost if the 480V shutdown board is lost. This can happen due to loss of power from the 6.9kV shutdown boards or seismically induced floods. The backup 480V FLEX DG must also fail in order to cause this scenario with the most predominant 480V FLEX DG failure being failure to align it. The most dominant failure of FCV-62-61 is caused by loss of the 480V RMOV boards due to reasons similar to those discussed above for the FCV-62-63 failures. Since 62-639 is a check valve, it is rugged and is not likely to experience any seismic impacts.
LATE-038	This is a late-release sequence that is considered early release for 0.5g earthquakes and above. In this sequence, the CARFs are unavailable, the hydrogen igniters are available, and the operator is successful in RCS depressurization. CHR has failed. It is fed by	The CARFs could fail due to seismic failure of the fans themselves or through loss of power from the 480V shutdown boards caused by direct seismic failure of the board or more likely relay chatter. They could also fail because of failure of a

Table 5.5-8 Unit 2 SLERF Significant Accident Sequences

Accident Sequence	Description	Discussion
	all the NHD core damage sequences from the Level 1 model.	containment isolation signal because the shutdown boards fail due to relay chatter or direct seismic board failure, along with failure of the 480V FLEX DGs. In order to fail CHR, either the containment sump must fail due to plugging or operator failure prior to the initiating event, which is unlikely, or CS in injection mode along with ice condenser failure must occur, or CS in recirculation mode along with RHR spray failure must happen. CS can fail either through failure of the operator action to transfer CS to the sump or by mechanical failure of the CS pumps or their flow paths. Direct seismic failure of the pumps or their coolers is unlikely because these components are fairly robust seismically. Loss of the 480V ventilation boards (because of loss of the 480V shutdown boards as a result of relay chatter of the 6.9kV shutdown boards which support them) will cause room cooling for the pumps to fail which is one mechanism that can fail the pumps. The pumps can also fail because of other support system failures such as loss of the CCS heat exchangers because of MOV failures in the flow path of the cooling to the heat exchangers caused by loss of the 480V RMOV boards. RHR spray can fail through loss of the RHR heat exchangers which can happen in the same way as previously discussed for cooling to the CS pumps. It can also occur through operator failure to start RHR spray or seismically induced flooding.
BLERF-001	This sequence involves a TI-SGTR and failure to depressurize the RCS. It is fed by all the NHD core damage sequences from the level 1 model.	A TI-SGTR occurs in this sequence. The RCS is not depressurized because one of the two PORVs does not open. This could be due to seismically induced failure of the PORVs to open or failure of operator to depressurize. Seismic failure of the PORVs themselves could occur but is unlikely due their high fragility. The most dominant cause of PORV failure is loss of DC control power because the 125V batteries and the chargers fail. The batteries could fail seismically, or they could fail due to battery depletion. They could fail due to a seismically induced flood, but the piping is rugged, so this is not likely. The chargers could fail if normal power from the 480V shutdown board fails because of loss of the board due to relay chatter and the backup power from the 480V FLEX DG is lost. The most likely reason the power from the 480V FLEX DG is lost is because of operator alignment failure.

Table 5.5-8 Unit 2 SLERF Significant Accident Sequences

Accident Sequence	Description	Discussion
LLERF-017	In this sequence, the CARFs fail, the hydrogen igniters fail, and the operator is successful in RCS depressurization. It is fed by all the NHD core damage sequences from the Level 1 model.	Both trains of the CARFs fail and both trains of hydrogen igniters fail. The hydrogen igniters are rugged so they are not likely to fail seismically; however, they could both fail due to operator failure to place them in service, although this is a fairly low-probability event. The primary way the hydrogen igniters fail is either though seismic loss of their power transformers or through loss of the 480V ventilation boards due to loss of the shutdown board itself or through relay chatter due to the seismic event.
LATE-034	This is a late release sequence that is considered early release for 0.5g earthquakes and above. In this sequence the CARFs and hydrogen igniters are available. Containment Heat Removal fails. It is fed by all the NHD core damage sequences from the Level 1 model.	The SQN IEPRA Level 2 model does not explicitly identify which large, late releases occur beyond 24 hours. As discussed in the Methodology, Modelling and Inputs notebook, to estimate the probability that containment fails before 24 hours, a conditional probability of containment failure before 24 hours assuming the ice condensers are available but CHR fails was derived for the seismic model that is included in the fault tree for this sequence. In order to fail CHR, either the containment sump must fail due to plugging or operator failure prior to the initiating event, which is unlikely, or CS in injection mode along with ice condenser failure must occur, or CS in recirculation mode along with RHR spray failure must happen. CS can fail either through failure of the operator action to transfer CS to the sump or by mechanical failure of the CS pumps or their flow paths. Direct seismic failure of the pumps or their coolers is unlikely because these components are fairly robust seismically. Loss of the 480V ventilation boards (because of loss of the 480V shutdown boards as a result of relay chatter of the 6.9kV shutdown boards which support them) will cause room cooling for the pumps to fail, which is one mechanism that can fail the pumps. The pumps can also fail because of other support system failures, such as loss of the CCS heat exchangers because of MOV failures in the flow path of the cooling to the heat exchangers caused by loss of the 480V RMOV boards. RHR spray can fail through loss of the RHR heat exchangers, which can happen in the same way as previously discussed for cooling to the CS pumps. It can also occur through operator failure to start RHR spray or seismically induced flooding.

5.6 SPRA Quantification Uncertainty Analysis

The nature of a PRA is such that the results have inherent uncertainty; these uncertainties must be understood and appreciated when using PRA results. In addition, exploration of the models, inputs, and results promotes an improved understanding of the analysis, and aids in identifying areas for refinement to reduce uncertainty.

NRC RG 1.200 [27] states that an important aspect in understanding the PRA results is knowing the sources of uncertainty and assumptions and understanding their potential impact. They include: (1) parameter uncertainties; (2) model uncertainties and related assumptions; (3) completeness uncertainties; and (4) assumptions related to scope and level of detail.

The scope of the SPRA was limited to the base PRA results and sources of uncertainty for the at-power, Level 1 PRA plus LERF for seismic events. The focus was also on *epistemic uncertainty* that results from incompleteness; it is noted that a PRA also includes *aleatory uncertainty* that results from randomness. The requirements of PRA applications will be evaluated separately for each application to determine whether sources of uncertainties and assumptions are acceptable. Uncertainties and sensitivities in the IEPRA base model are documented in the Quantification, Uncertainty and Sensitivity Analysis Notebook [1].

5.6.1 Parameter Uncertainty

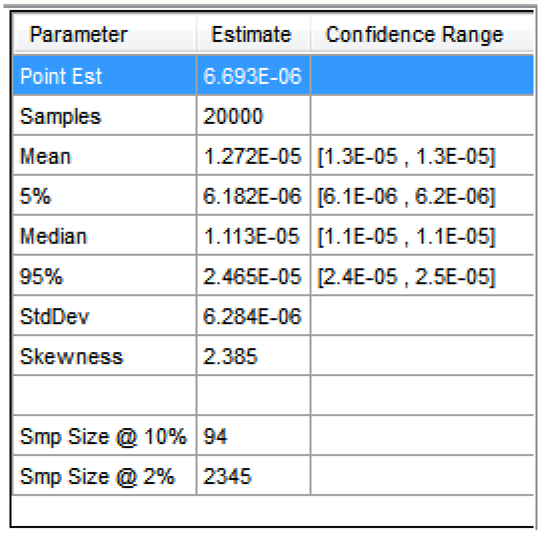
Parameter uncertainty relates to the uncertainty in the computation of the input parameter values used to quantify the model (i.e., initiating event frequencies, component failure probabilities and HEPs). These uncertainties can be characterized by probability distributions that relate to the degree of belief in their values. A formal propagation of uncertainty is the best way to correctly account for this, and the PRA software UNCERT has the capability to propagate these uncertainties.

SCDF uncertainty analysis results are summarized in Table 5.6-1 and presented in Figure 5.6-1 for Unit 1 and are summarized in Table 5.6-2 and presented in Figure 5.6-2 for Unit 2. The uncertainty analysis was performed with UNCERT 4.0, using Monte Carlo sampling with 20,000 samples and ACUBE processing of 1,000 cutsets

The UNCERT analysis included distributions for seismic bin frequencies, fragility estimates, seismic HEPs, and IEPRA basic events. The seismic bin frequency distributions are presented in Table 5.6-5. Those distributions were generated by the FRANX code assuming lognormal distributions and estimating error factors (EFs) from the various seismic hazard curves inputted to the code (16th, median, and 84th). See the FRANX manual for information on this topic.

Sampling of the individual seismic bin frequencies was performed using the correlated approach described in the FRANX manual. Seismic failure probability distributions are determined automatically by FRANX given the fragility parameter estimates (A_m, β_R, and β_U). Distributions for HEPs and combination factors were calculated in the HRA Calculator Version 5.2. Distributions for IEPRA basic events were left unchanged from the IEPRA model.

Table 5.6-1 Unit 1 SCDF Uncertainty Results



Parameter	Estimate	Confidence Range
Point Est	6.693E-06	
Samples	20000	
Mean	1.272E-05	[1.3E-05 , 1.3E-05]
5%	6.182E-06	[6.1E-06 , 6.2E-06]
Median	1.113E-05	[1.1E-05 , 1.1E-05]
95%	2.465E-05	[2.4E-05 , 2.5E-05]
StdDev	6.284E-06	
Skewness	2.385	
Smp Size @ 10%	94	
Smp Size @ 2%	2345	

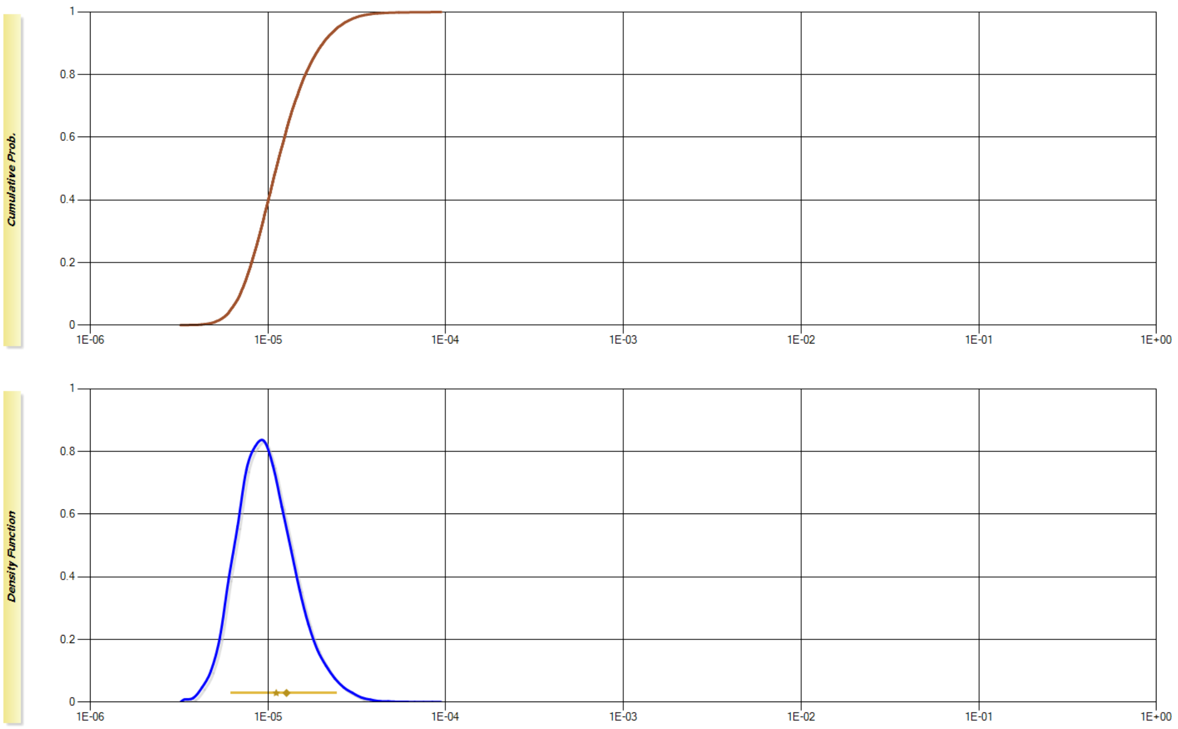
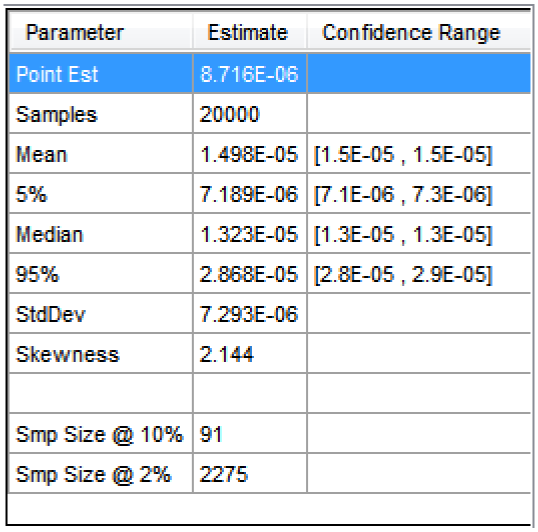


Figure 5.6-1: Unit 1 SCDF Uncertainty Results

Table 5.6-2 Unit 2 SCDF Uncertainty Results



Parameter	Estimate	Confidence Range
Point Est	8.716E-06	
Samples	20000	
Mean	1.498E-05	[1.5E-05 , 1.5E-05]
5%	7.189E-06	[7.1E-06 , 7.3E-06]
Median	1.323E-05	[1.3E-05 , 1.3E-05]
95%	2.868E-05	[2.8E-05 , 2.9E-05]
StdDev	7.293E-06	
Skewness	2.144	
Smp Size @ 10%	91	
Smp Size @ 2%	2275	

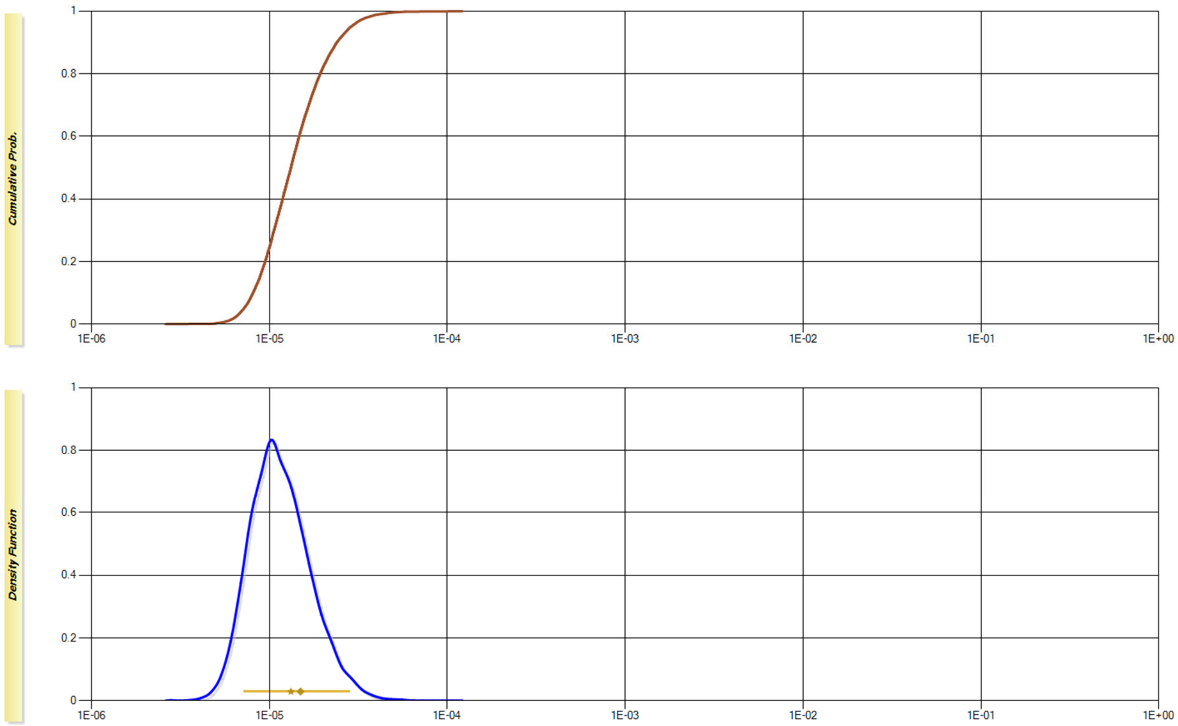
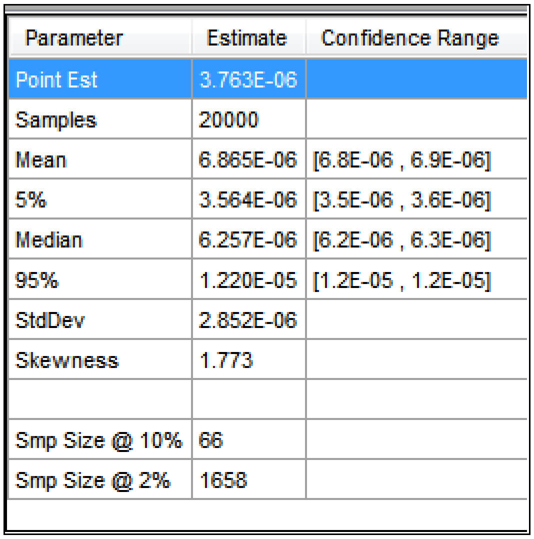


Figure 5.6-2: Unit 2 SCDF Uncertainty Results

Table 5.6-3 Unit 1 SLERF Uncertainty Results



Parameter	Estimate	Confidence Range
Point Est	3.763E-06	
Samples	20000	
Mean	6.865E-06	[6.8E-06 , 6.9E-06]
5%	3.564E-06	[3.5E-06 , 3.6E-06]
Median	6.257E-06	[6.2E-06 , 6.3E-06]
95%	1.220E-05	[1.2E-05 , 1.2E-05]
StdDev	2.852E-06	
Skewness	1.773	
Smp Size @ 10%	66	
Smp Size @ 2%	1658	

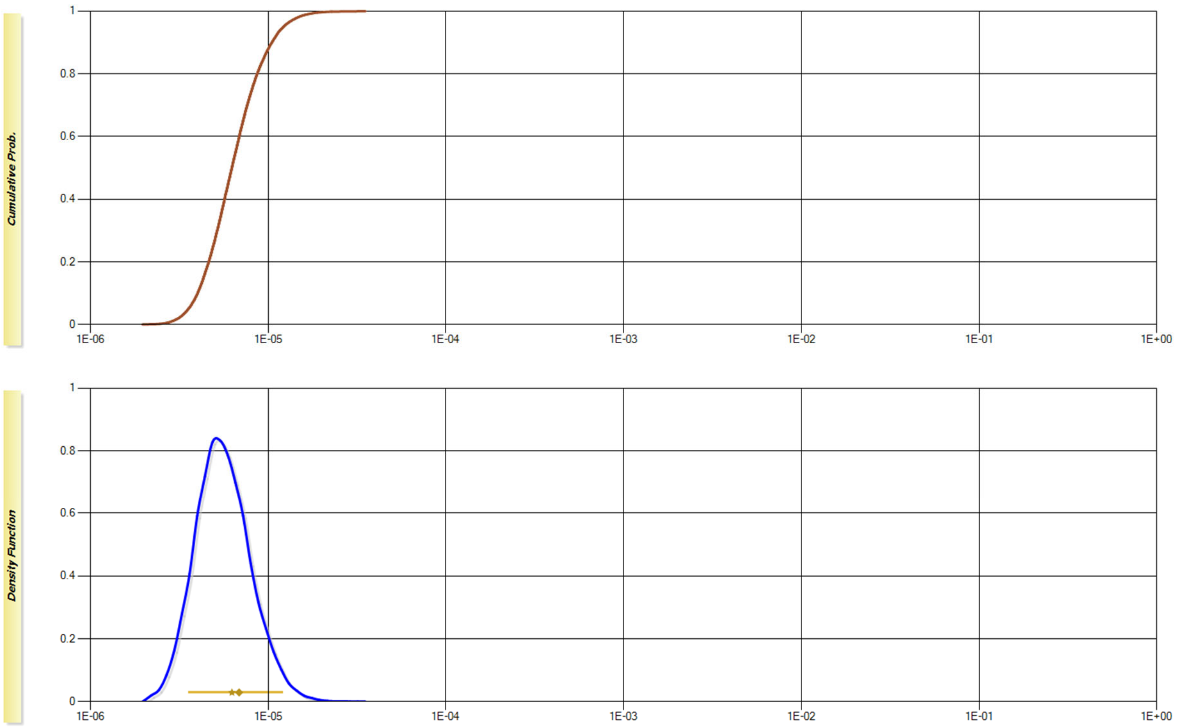
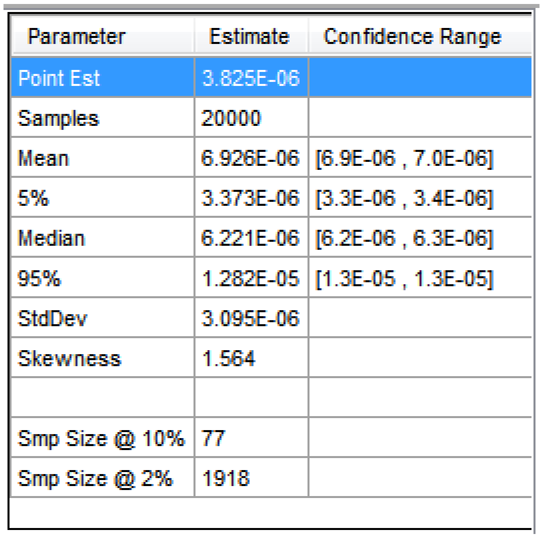


Figure 5.6-3: Unit 1 SLERF Uncertainty Results

Table 5.6-4 Unit 2 SLERF Uncertainty Results



Parameter	Estimate	Confidence Range
Point Est	3.825E-06	
Samples	20000	
Mean	6.926E-06	[6.9E-06 , 7.0E-06]
5%	3.373E-06	[3.3E-06 , 3.4E-06]
Median	6.221E-06	[6.2E-06 , 6.3E-06]
95%	1.282E-05	[1.3E-05 , 1.3E-05]
StdDev	3.095E-06	
Skewness	1.564	
Smp Size @ 10%	77	
Smp Size @ 2%	1918	

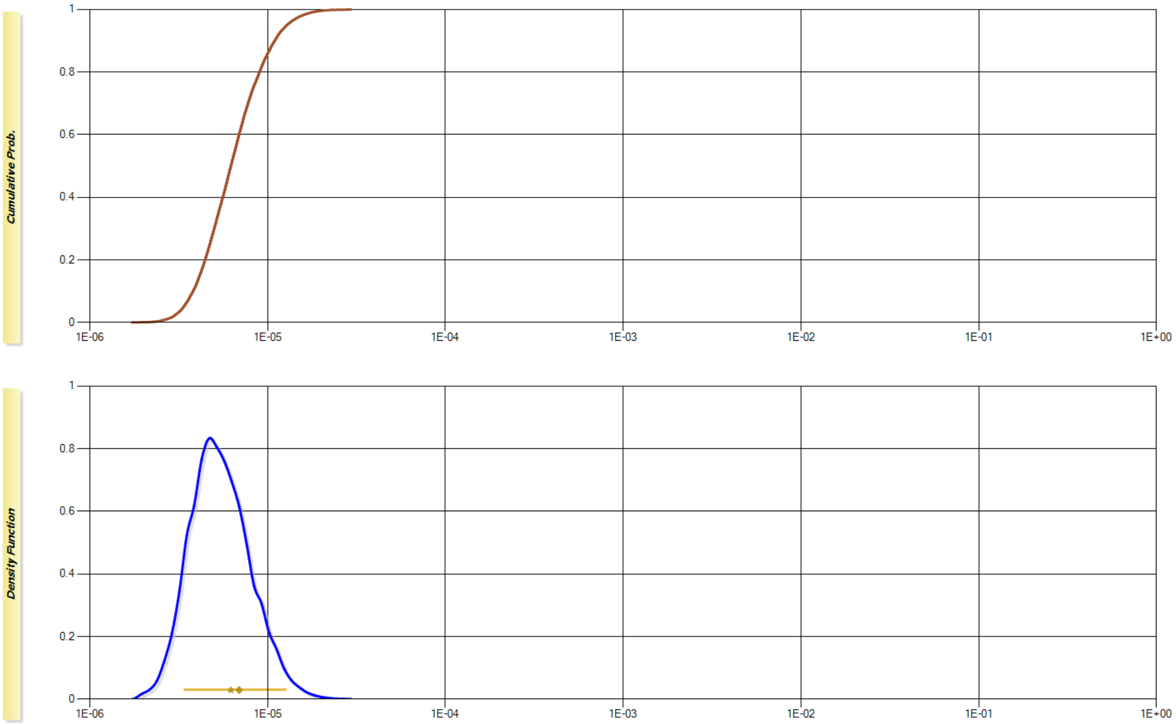


Figure 5.6-4: Unit 2 SLERF Uncertainty Results

Table 5.6-5 Seismic Bin Frequency Distributions

Seismic Bin	Bin PGA (g)	Mean Frequency (1/yr)	Error Factor (EF)
%G01	0.13	3.97E-04	4.96
%G02	0.23	1.24E-04	4.49
%G03	0.39	5.39E-05	4.49
%G04	0.59	1.49E-05	4.96
%G05	0.79	5.50E-06	5.30
%G06	1.00	2.40E-06	5.92
%G07	1.28	1.81E-06	7.26
%G08	2.12	9.67E-07	10.7
%G09	3.3	9.24E-08	39.3
Note: Uncertainty in the bin PGA is assumed to be covered in the bin frequency distribution.			

5.6.2 Model Uncertainty

Model uncertainty arises because different approaches exist to represent plant response. A source of model uncertainty is one related to an issue in which no consensus approach or model exists, and where the choice of approach or model is known to effect the SPRA. These uncertainties are typically dealt with by making assumptions; e.g., the approach to address common-cause failure, how an RCP would fail following a loss of seal cooling, the approach to identify and quantify HFEs. In general, model uncertainties are addressed through sensitivity studies using different models or assumptions.

The guidance provided in EPRI 1016737 [43], was used to address sources of model uncertainty and related assumptions. It provides a framework for the pragmatic treatment of uncertainty characterization to support risk-informed applications and decision making. The process includes identification and characterization of sources of model uncertainty and related assumptions. Appendix G of the Quantification Notebook [1] shows a listing and disposition of the assessed sources of model uncertainty.

5.6.3 Completeness Uncertainty

Completeness uncertainty relates to risk contributors that are not in the SPRA model. These include known types such as the scope of the PRA, which does not include some classes of initiating events, hazards, and operating modes; and the level of analysis, which may have omitted phenomena, failure mechanisms, or other factors because their relative contribution is believed to be negligible. They also include ones that are not known such as the effects on risk from aging or organizational changes; and omitted phenomena and failure mechanisms that are unknown. Both can have a significant impact on risk.

No completeness uncertainties were identified for the SQN SPRA based on the ASME/ANS PRA Standard.

5.6.4 Truncation Study

A truncation study was performed on the Unit 1 and Unit 2 SPRA models (SCDF and SLERF) to ensure that sufficient cutsets were generated to result in an accurate estimate for SCDF and SLERF. The truncation study is more complex than typically performed for the IEPRA CDF because of several reasons:

1. Quantification of the SPRA SCDF and SLERF is performed separately by seismic bin, and the results are then combined to obtain a total SCDF and SLERF estimate.
2. ACUBE post-processing of bin cutsets is performed to obtain more accurate cutset summation estimates, and the number of cutsets that can be processed by ACUBE is limited.
3. The number of fragility events included in the model may be limited by software and hardware constraints.

Therefore, the truncation study is multi-dimensional. In order to complete the truncation study, a simplified modeled was developed to limit the fragility events included in the model. Fragility events with A_m above 3g are assumed to not be vulnerable to seismic failure while events with A_m below 0.3g are assumed to fail seismically. Results of the truncation study are summarized in Tables 5.6-6 through 5.6-9. The truncation study was performed by using ACUBE processing for each seismic bin.

The results of the Unit 1 SCDF truncation study are shown in Table 5.6-6. The increase in total SCDF is 3% for a decade reduction in truncation, therefore, Unit 1 SCDF is converged.

Table 5.6-6 Unit 1 SCDF Truncation Study

Seismic Bin	Bin Frequency	Base Truncation	SCDF at Base Truncation	Minimum Truncation	SCDF at Min Truncation
%G01	3.97E-04	1.00E-12	5.04E-09	1.00E-13	5.50E-09
%G02	1.24E-04	1.00E-12	1.88E-09	1.00E-13	2.06E-09
%G03	5.39E-05	5.00E-11	1.25E-08	5.00E-12	1.30E-08
%G04	1.49E-05	5.00E-11	8.66E-08	5.00E-12	8.86E-08
%G05	5.50E-06	5.00E-11	2.94E-07	5.00E-12	3.02E-07
%G06	2.40E-06	5.00E-11	5.67E-07	5.00E-12	6.23E-07
%G07	1.81E-06	5.00E-11	1.71E-06	5.00E-12	1.74E-06
%G08	9.67E-07	5.00E-08	9.67E-07	5.00E-09	9.67E-07
%G09	9.24E-08	5.00E-08	8.39E-08	5.00E-09	8.39E-08
Total CDF			3.73E-06		3.83E-06

The results of the Unit 2 SCDF truncation study are shown in Table 5.6-7. The increase in total SCDF for a decade reduction in truncation is 3%, therefore, Unit 2 SCDF is converged.

Table 5.6-7 Unit 2 SCDF Truncation Study

Seismic Bin	Bin Frequency	Base Truncation	SCDF at Base Truncation	Minimum Truncation	SCDF at Min Truncation
%G01	3.97E-04	1.00E-12	7.09E-09	1.00E-13	8.20E-09
%G02	1.24E-04	1.00E-12	2.26E-09	1.00E-13	2.66E-09
%G03	5.39E-05	5.00E-11	1.57E-08	5.00E-12	1.63E-08
%G04	1.49E-05	5.00E-11	9.52E-08	5.00E-12	9.74E-08
%G05	5.50E-06	5.00E-11	2.99E-07	5.00E-12	3.06E-07
%G06	2.40E-06	5.00E-11	5.60E-07	5.00E-12	6.14E-07
%G07	1.81E-06	5.00E-11	1.65E-06	5.00E-12	1.69E-06
%G08	9.67E-07	5.00E-08	9.67E-07	5.00E-09	9.67E-07
%G09	9.24E-08	5.00E-08	8.61E-08	5.00E-09	8.61E-08
Total SCDF			3.68E-06		3.79E-06

The results of the Unit 1 SLERF truncation study are shown in Table 5.6-8. The increase in total SLERF for a decade reduction in truncation is 5%, therefore, Unit 1 SLERF is converged.

Table 5.6-8 Unit 1 SLERF Truncation Study

Seismic Bin	Bin Frequency	Base Truncation	LERF at Base Truncation	Minimum Truncation	LERF at Min Truncation
%G01	3.97E-04	1.00E-12	4.41E-10	1.00E-13	5.05E-10
%G02	1.24E-04	1.00E-12	1.38E-10	1.00E-13	1.51E-10
%G03	5.39E-05	5.00E-11	1.23E-09	5.00E-12	1.31E-09
%G04	1.49E-05	5.00E-11	3.52E-08	5.00E-12	3.69E-08
%G05	5.50E-06	5.00E-11	1.79E-07	5.00E-12	1.90E-07
%G06	2.40E-06	5.00E-11	4.51E-07	5.00E-12	5.78E-07
%G07	1.81E-06	5.00E-11	1.71E-06	5.00E-12	1.74E-06
%G08	9.67E-07	5.00E-08	9.67E-07	5.00E-09	9.67E-07
%G09	9.24E-08	5.00E-08	8.39E-08	5.00E-09	8.39E-08
Total SLERF			3.43E-06		3.60E-06

The results of the Unit 2 SLERF truncation study are shown in Table 5.6-9. The increase in total SLERF for a decade reduction in truncation is 5%, therefore, Unit 2 SLERF is converged.

Table 5.6-9 Unit 2 SLERF Truncation Study

Seismic Bin	Bin Frequency	Base Truncation	LERF at Base Truncation	Minimum Truncation	LERF at Min Truncation
%G01	3.97E-04	1.00E-12	4.82E-10	1.00E-13	5.79E-10
%G02	1.24E-04	1.00E-12	1.44E-10	1.00E-13	1.68E-10
%G03	5.39E-05	5.00E-11	1.25E-09	5.00E-12	1.41E-09
%G04	1.49E-05	5.00E-11	3.65E-08	5.00E-12	3.83E-08
%G05	5.50E-06	5.00E-11	1.80E-07	5.00E-12	1.90E-07
%G06	2.40E-06	5.00E-11	4.58E-07	5.00E-12	5.73E-07
%G07	1.81E-06	5.00E-11	1.65E-06	5.00E-12	1.69E-06
%G08	9.67E-07	5.00E-08	9.67E-07	5.00E-09	9.67E-07
%G09	9.24E-08	5.00E-08	8.61E-08	5.00E-09	8.61E-08
Total SLERF			3.38E-06		3.55E-06

5.7 SPRA Quantification Sensitivity Analysis

The sensitivity studies described below are used to investigate other sources of uncertainty that affect the modeling of seismic impacts and the quantification methods used.

The following areas were investigated:

- Modeling of Seismic Impacts
- Correlation of Fragilities
- Relay Chatter
- Human Reliability Analysis

The results for each of the seismic-related sensitivity cases are provided in Table 5.7-1. The sensitivity cases, their definition, manner of implementation, and their SCDF and SLERF results are discussed below. All the basic event importance, HEP importance, and fragility importance tables for the base case results throughout the rest of the document reflect the values after the correction was made. Sensitivity cases 1 through 26C are based on the model in place at the time of the peer review. Sensitivity cases 27 through 31 were performed on the updated model after peer review F&Os and other related model changes were made.

Since the reference earthquake is higher than the GMRS, a non-linear soil study was conducted (F&O 3-4) and the effects of nonlinearities in soil failure and building response were studied (F&O 5-1). Based on the recommendations in F&Os 3-4 and 5-1, sensitivity studies 27-30 were performed to address the fact that the reference earthquake is higher than the GMRS.

Table 5.7-1 Sensitivity Study Results Summary

Case #	Description	Risk Measure	Sensitivity SCDF/SLERF	Percent Change from Base Mode	Discussion
1	HEPs set at 5%	U1 SCDF	5.11E-06	-3.4%	Due to the way the uncertainty analysis was performed, the uncertainty of the HEP values was accounted for in the propagation of uncertainty by UNCERT, making the first two sensitivity studies somewhat redundant. The results show that the model is not especially sensitive to the uncertainties in HEP analyses. In the case of the HEP analyses, the most important HEP events are assumed failed in the higher seismic bins, giving them a constant value of one.
		U2 SCDF	5.65E-06	-2.3%	
		U1 SLERF	2.97E-06	-0.7%	
		U2 SLERF	3.08E-06	-0.9%	
2	HEPs set at 95%	U1 SCDF	6.37E-06	20.3%	
		U2 SCDF	6.79E-06	17.4%	
		U1 SLERF	3.12E-06	4.3%	
		U2 SLERF	3.23E-06	3.9%	
3	Fail FLEX diesels	U1 SCDF	5.49E-06	3.6%	The FLEX diesels were an SSC added to the model for the SPRA that are currently not modeled in the internal events PRA. This sensitivity was performed by setting the FLEX diesel components to the plant-level fragility value because preliminary results obtained from just removing the logic from the model showed no change in SCDF/SLERF. The results show a small increase in SCDF and SLERF; however, the results for Unit 2 SLERF are believed to be misleading in showing a decrease in SLERF. This is because several of the seismic bins had to be quantified at a higher truncation than the base case.
		U2 SCDF	6.43E-06	11.1%	
		U1 SLERF	3.33E-06	11.3%	
		U2 SLERF	2.80E-06	-9.9%	
4	RWST fragilities increased 25%	U1 SCDF	5.30E-06	0.1%	This sensitivity shows that increasing the fragility of the RWST has a negligible effect on plant risk. The positive increase in SCDF/SLERF for U2_CDF, U2_CDF and U2_LERF can be attributed to rounding error since the results are so similar to the base case.
		U2 SCDF	5.79E-06	0.1%	
		U1 SLERF	2.99E-06	-0.1%	
		U2 SLERF	3.11E-06	0.0%	
5	72-hour mission time	U1 SCDF	5.71E-06	7.8%	This sensitivity involved increasing the mission time of all events having a mission time of 24 hours to 72 hours to account for the possibility of the plant safety systems having to operate beyond the 24 hours typically assumed in internal events PRAs. The results show a moderate increase in SCDF and SLERF.
		U2 SCDF	6.64E-06	14.8%	
		U1 SLERF	3.11E-06	4.0%	
		U2 SLERF	3.26E-06	4.9%	
6	Credit recovery above 1.5g-mapped S3 HEPs to bin %G08	U1 SCDF	5.02E-06	-5.2%	This sensitivity measured the effects of assuming some sort of recovery for seismic bins %G08 by assigning the HEPs in EPRI HEP seismic bin S3 to those bins. The base case model assumes no credit for human actions for these bins. The results show a moderate decrease in SCDF/SLERF for this case.
		U2 SCDF	5.50E-06	-4.9%	
		U1 SLERF	2.69E-06	-10.1%	
		U2 SLERF	2.80E-06	-9.9%	

Table 5.7-1 Sensitivity Study Results Summary

Case #	Description	Risk Measure	Sensitivity SCDF/SLERF	Percent Change from Base Mode	Discussion
7	Fail ex-control room operator actions for S3	U1 SCDF	1.54E-05	190.7%	This sensitivity assumed that all human actions that occur in whole or in part outside of the control room are assumed failed for seismic bins %G03 through %G07. All human actions for seismic bin %G08 and %G09 were also assumed failed, as in the base case. The model shows a large increase in both SCDF and SLERF if this assumption is made.
		U2 SCDF	1.27E-05	119.5%	
		U1 SLERF	7.39E-06	147.0%	
		U2 SLERF	6.44E-06	107.2%	
8	Uncorrelated MCCs-uncorrelated 1A fragility groups	U1 SCDF	5.30E-06	0.1%	This sensitivity was performed to show the effects of uncorrelated MCCs in the SEIS_1A* fragility groups. The expected result was that CDF and LERF would decrease; however, the actual result was an increase. This is believed to be a result of the way that FRANX/ACUBE quantifies the cutsets. The result of uncorrelating these fragility groups is to create many cutsets in place of each single cutset that would result if the fragility groups were assumed to be correlated, leading to an overall higher result.
		U2 SCDF	5.64E-06	-2.5%	
		U1 SLERF	4.21E-06	40.7%	
		U2 SLERF	4.20E-06	35.1%	
9	Double the HINST failure probability	U1 SCDF	5.27E-06	-0.5%	This sensitivity assumed that the probability calculated for HINST (the failure of control room instrumentation) was doubled for each fragility group. The results show a small increase in SCDF/SLERF. The U1_CDF and U1_LERF show a slight decrease because some seismic bins could not be quantified at the base case truncation.
		U2 SCDF	5.79E-06	0.1%	
		U1 SLERF	2.95E-06	-1.4%	
		U2 SLERF	3.18E-06	2.3%	
10	Halve the HINST failure probability	U1 SCDF	5.28E-06	-0.3%	This sensitivity assumed that the probability calculated for HINST (the failure of control room instrumentation) was halved for each fragility group. The results show a small decrease in SCDF/SLERF.
		U2 SCDF	5.78E-06	-0.1%	
		U1 SLERF	2.92E-06	-2.4%	
		U2 SLERF	3.04E-06	-2.2%	
11	Large late release does not go to large early release above 0.5g	U1 SCDF	5.30E-06	0.0%	The base case assumes that in all scenarios at or above 0.5g, the Level 2 sequences usually assumed to be late are assumed to be early, in order to account for changes in evacuating times caused by a seismic event. This sensitivity assumes that this is not the case (i.e., the IEPRA late sequences are late in the SPRA also). The results show a moderate decrease in SLERF.
		U2 SCDF	5.79E-06	0.0%	
		U1 SLERF	2.71E-06	-9.4%	
		U2 SLERF	2.81E-06	-9.6%	
12	Do not credit recovery from relay chatter events-OP-LOCKOUT*	U1 SCDF	1.15E-05	117.1%	The seismic model includes a recovery event for certain relay chatter events affecting the EDG lockout relays, the 6.9kV shutdown board lockout relays, and the 480V shutdown board lockout relays. This sensitivity shows the effect of not crediting this recovery action for any seismic bin. The results show a significant increase in SCDF/SLERF.
		U2 SCDF	1.00E-05	72.0%	
		U1 SLERF	4.73E-06	58.1%	
		U2 SLERF	4.45E-06	42.9%	
13	Modify seismic bin intervals	U1 SCDF	5.35E-06	1.0%	This sensitivity shows the effects of modifying the break points of the mid-range seismic intervals. Seismic bins %G01, %G02 and the break point between bin %G07 and %G08 were not changed so that the HEPs of the various bins did not need to be changed. The bin intervals used were: %G01: 0.09 to 0.18g
		U2 SCDF	5.96E-06	2.5%	
		U1 SLERF	3.02E-06	0.9%	

Table 5.7-1 Sensitivity Study Results Summary

Case #	Description	Risk Measure	Sensitivity SCDF/SLERF	Percent Change from Base Mode	Discussion
		U2 SLERF	3.06E-06	-1.7%	%G02: 0.18 to 0.3g %G03: 0.3 to 0.4g %G04: 0.4 to 0.5g %G05 0.5 to 0.6g %G06: 0.6 to 1.05g %G07: 1.05 to 1.5g %G08: 1.5 to 3g %G09: >3g The results show a small increase in SCDF/SLERF, although not all seismic bins could be quantified at their original base case truncations.
14	Make block wall fragilities rugged	U1 SCDF	4.79E-06	-9.6%	This sensitivity assumed that the fragilities of block walls that fail equipment and impede or prevent operator access were assumed to be rugged, or essentially not to fail in the model. This resulted in a moderate decrease in SCDF/SLERF.
		U2 SCDF	5.35E-06	-8.0%	
		U1 SLERF	2.64E-06	-11.8%	
		U2 SLERF	2.78E-06	-10.7%	
15	No credit for manual control of TDAFW pump	U1 SCDF	5.70E-06	7.6%	This sensitivity shows the effect of assuming that there is not manual control of the turbine-driven auxiliary feedwater pump following battery depletion. The results show a moderate increase in SCDF/SLERF.
		U2 SCDF	6.26E-06	7.6%	
		U1 SLERF	3.14E-06	5.0%	
		U2 SLERF	3.26E-06	4.7%	
16	Increase piping fragility by 30%	U1 SCDF	5.30E-06	0.1%	This sensitivity assumed that the fragility of buried ERCW piping (SEIS_0-1) was increased by 30%. The expected result would be a slight decrease in SCDF and SLERF, and given that the results are so close to the base case, the small increase is believed to be due to rounding error in FRANX, since using the delete term function in the CAFTA cutset editor shows no difference in the cutsets from the base case.
		U2 SCDF	5.82E-06	0.0%	
		U1 SLERF	2.99E-06	0.0%	
		U2 SLERF	3.11E-06	-0.1%	
17	Decrease piping fragility by 30%	U1 SCDF	5.30E-06	0.1%	This sensitivity assumed that the fragility of buried ERCW piping (SEIS_0-1) was decreased by 30%. The expected result would be a slight increase in SCDF and SLERF, and given that the results are so close to the base case, the small increase for U1_LERF is believed to be due to rounding error in FRANX, since using the delete term function in the CAFTA cutset editor shows no difference in the cutsets from the base case.
		U2 SCDF	5.82E-06	0.1%	
		U1 SLERF	2.99E-06	0.0%	
		U2 SLERF	3.11E-06	-0.1%	
18	Use 84% upper bound for hazard curve	U1 SCDF	9.44E-06	78.2%	This sensitivity used the 84% upper uncertainty bound of the seismic hazard curve in FRANX to determine the AFE for each seismic bin. The results show a relatively large increase in SCDF and SLERF.
		U2 SCDF	1.05E-05	80.6%	
		U1 SLERF	5.99E-06	100.2%	
		U2 SLERF	5.90E-06	89.4%	

Table 5.7-1 Sensitivity Study Results Summary

<table>
<tr><th>Case #</th><th>Description</th><th>Risk Measure</th><th>Sensitivity SCDF/SLERF</th><th>Percent Change from Base Mode</th><th>Discussion</th></tr>
<tr><td rowspan="4">19</td><td rowspan="4">Use 16% lower bound for hazard curve</td><td>U1 SCDF</td><td>9.66E-07</td><td>-81.8%</td><td rowspan="4">This sensitivity used the 16% lower uncertainty bound of the seismic hazard curve in FRANX to determine the AFE for each seismic bin. The results show a relatively large decrease in SCDF and SLERF.</td></tr>
<tr><td>U2 SCDF</td><td>1.05E-06</td><td>-81.9%</td></tr>
<tr><td>U1 SLERF</td><td>4.09E-07</td><td>-86.3%</td></tr>
<tr><td>U2 SLERF</td><td>4.04E-07</td><td>-87.0%</td></tr>
<tr><td rowspan="4">20</td><td rowspan="4">Increase 480V FLEX diesel fragility by 25%</td><td>U1 SCDF</td><td>5.30E-06</td><td>0.1%</td><td rowspan="4">This sensitivity shows the effect of increasing the fragility of the FLEX diesel components by 25%. The results show a negligible decrease in SCDF and SLERF. The small increase in some of the results is believed to be due to rounding error in FRANX, since using the delete term function in the CAFTA cutset editor shows no difference in the cutsets from the base case.</td></tr>
<tr><td>U2 SCDF</td><td>5.79E-06</td><td>-0.4%</td></tr>
<tr><td>U1 SLERF</td><td>2.99E-06</td><td>-0.1%</td></tr>
<tr><td>U2 SLERF</td><td>3.11E-06</td><td>-0.1%</td></tr>
<tr><td rowspan="4">21</td><td rowspan="4">Eliminate sequences with FLG_ATWS that are not ATWS sequences</td><td>U1 SCDF</td><td>5.26E-06</td><td>-0.7%</td><td rowspan="4">This sensitivity eliminates sequences that are not considered anticipated transient without scram (ATWS) sequences and yet have a flag in in the cutsets indicating an ATWS (FLG_ATWS). These cutsets were noticed during final quantification, and this sensitivity was performed to determine the effect of removing these erroneous sequences. This was due to certain portions of the mutually exclusive logic being modified for the seismic analysis to eliminate sequences that would be eliminated incorrectly by the original internal events mutually exclusive logic. The results show a slight decrease in SCDF and SLERF.</td></tr>
<tr><td>U2 SCDF</td><td>5.76E-06</td><td>-1.0%</td></tr>
<tr><td>U1 SLERF</td><td>3.01E-06</td><td>0.6%</td></tr>
<tr><td>U2 SLERF</td><td>3.10E-06</td><td>-0.5%</td></tr>
<tr><td rowspan="4">22</td><td rowspan="4">No guaranteed failure components</td><td>U1 SCDF</td><td>5.56E-06</td><td>5.0%</td><td rowspan="4">This sensitivity was performed to determine the effect of assuming the components that were assumed guaranteed to fail in the seismic model (such as the TB components) were instead allowed to fail at their normal failure rates or by seismically induced failure fragilities. The expected result of this sensitivity would be an overall decrease in SCDF and SLERF; however, the opposite effect was observed. This is believed to be a result of the way that FRANX/ACUBE quantifies the cutsets. The result of adding many more cutsets that would normally be absent if the component was assumed to be failed and set to false is to create many cutsets in place of each single cutset that would result if the fragility groups were assumed to be correlated, leading to an overall higher result.</td></tr>
<tr><td>U2 SCDF</td><td>5.93E-06</td><td>2.0%</td></tr>
<tr><td>U1 SLERF</td><td>3.77E-06</td><td>26.0%</td></tr>
<tr><td>U2 SLERF</td><td>3.99E-06</td><td>28.1%</td></tr>
<tr><td rowspan="4">23</td><td rowspan="4">Increase selected components with an A_m of 1.3 or more by a factor of 1.75</td><td>U1 SCDF</td><td>4.97E-06</td><td>-6.2%</td><td rowspan="4">This sensitivity involved increasing the fragilities of the following fragility groups each by a factor of 1.75 simultaneously:
<table>
<tr><td>SEIS_0-30-1</td><td>Relay/breaker Cab. & Model = SSPS Cabinets (R-048, R-051), Westinghouse/AR440AR</td></tr>
<tr><td>SEIS_16-1</td><td>AUX Control Air Compressor</td></tr>
<tr><td>SEIS_1A-5</td><td>480V RX MOV BD</td></tr>
<tr><td>SEIS_1A-7</td><td>125V DC VITAL BAT BD</td></tr>
<tr><td>SEIS_1C-1-2</td><td>U2 6.9KV SD BD</td></tr>
<tr><td>SEIS_3-2-1</td><td>Aux. Battery Charger Sub 1</td></tr>
<tr><td>SEIS_4-2</td><td>6.9KV/480V Shutdown Transformers</td></tr>
<tr><td>SEIS_FLEX_BUS480</td><td>FLEX 480V DG Bus Panel</td></tr>
</table></td></tr>
<tr><td>U2 SCDF</td><td>5.57E-06</td><td>-4.2%</td></tr>
<tr><td>U1 SLERF</td><td>2.64E-06</td><td>-11.8%</td></tr>
<tr><td>U2 SLERF</td><td>2.75E-06</td><td>-11.5%</td></tr>
</table>

Table 5.7-1 Sensitivity Study Results Summary

<table>
<tr><th>Case #</th><th>Description</th><th>Risk Measure</th><th>Sensitivity SCDF/SLERF</th><th>Percent Change from Base Mode</th><th>Discussion</th></tr>
<tr><td></td><td></td><td></td><td></td><td></td><td><table><tr><td>SEIS_23-4</td><td>NSSS PZR</td></tr><tr><td>SEIS_23-6</td><td>NSSS SG</td></tr></table>
This sensitivity case was performed because there is indication from the final quantification results that the seismic level that has the greatest impact on plant risk could be in the seismic bin higher than the GMRS (this is discussed in the Fragility Notebook). Therefore, a sensitivity study was performed to determine the impact on the results of the SPRA if the risk were to be dominated by this higher seismic level. As discussed in detail in the Fragility Notebook, a scale factor was determined to apply to top risk contributors for the sensitivity study. The scale factor to apply is 1.75 to all the top risk contributors in the ACB.

The results indicate a small to moderate decrease in SCDF and SLERF.</td></tr>
<tr><td rowspan="4">24</td><td rowspan="4">Set fragility of SEIS_1A-7 to A_m=1.74 to correspond to assumed Q1 fragility</td><td>U1 SCDF</td><td>5.29E-06</td><td>-0.1%</td><td rowspan="4">This sensitivity involved setting the fragility of group SEIS_1A-7 (125V DC VITAL BAT BD) to the same value assumed in the first official quantification. The results indicate virtually no change in SCDF or SLERF.</td></tr>
<tr><td>U2 SCDF</td><td>5.81E-06</td><td>0.4%</td></tr>
<tr><td>U1 SLERF</td><td>2.99E-06</td><td>-0.1%</td></tr>
<tr><td>U2 SLERF</td><td>3.11E-06</td><td>0.0%</td></tr>
<tr><td rowspan="4">25</td><td rowspan="4">Increase fragilities of NSSS components by 20%, except for steam generators</td><td>U1 SCDF</td><td>5.08E-06</td><td>-4.1%</td><td rowspan="4">This sensitivity involved increasing the fragilities of groups SEIS_23-1a, SEIS_23-1b, SEIS_23-2, SEIS_23-3, SEIS_23-4, SEIS_23-5). This resulted in a small to moderate reduction in SCDF and SLERF.</td></tr>
<tr><td>U2 SCDF</td><td>5.63E-06</td><td>-2.7%</td></tr>
<tr><td>U1 SLERF</td><td>2.74E-06</td><td>-8.4%</td></tr>
<tr><td>U2 SLERF</td><td>2.86E-06</td><td>-8.0%</td></tr>
<tr><td rowspan="4">26C</td><td rowspan="4">Add post-freeze date changes</td><td>U1 SCDF</td><td>5.22E-06</td><td>-1.4%</td><td rowspan="4">This sensitivity incorporates the changes known to be required to the model that were discovered subsequent to the freeze date for model changes following the final official quantification. The changes involve:
Performing a detailed analysis on HEP HAOB3 since it appeared as risk-significant on the final quantification
Updating the fragility of group SEIS_0-30-2 (Relay Chatter - Test Panels (R-052, R-053))
Updating the fragility of group SEIS_0-30-7 (Relay Chatter - 6.9kV SD BD Logic Relay Panel)
Adding a new fragility group SEIS_0-30-15 that includes updated fragilities for EDG AGASTAT time-delay relays, and linking this fragility group to diesel failure
Mapping fragility group SEIS_0-30-14 (Relay Chatter - Trip & Throttle Valve TDAFW Pump) to the TDAFW pumps
Assuming the CSTs are not guaranteed failure
Adding the modifications done for sensitivity cases #21 and #24
The greatest change resulted from the revised value of HAOB3 and the associated dependency analysis, with the updated fragility group mapping and fragility group values having a negligible impact.</td></tr>
<tr><td>U2 SCDF</td><td>5.75E-06</td><td>-1.1%</td></tr>
<tr><td>U1 SLERF</td><td>3.02E-06</td><td>0.8%</td></tr>
<tr><td>U2 SLERF</td><td>3.11E-06</td><td>-0.1%</td></tr>
<tr><td rowspan="2">27</td><td rowspan="2">RB scale factors increased (*VSLOCA scale factor not increased)</td><td>U1 SCDF</td><td>3.26E-06</td><td>-20.8%</td><td rowspan="2">This sensitivity increased the scale factors of the following fragility groups:<table><tr><th>Item</th><th>Scale Factor</th><th>Building</th><th>New Fragility</th></tr></table></td></tr>
<tr><td>U2 SCDF</td><td>4.08E-06</td><td>-16.1%</td></tr>
</table>

Table 5.7-1 Sensitivity Study Results Summary

Case #	Description	Risk Measure	Sensitivity SCDF/SLERF	Percent Change from Base Mode	Discussion			
		U1 SLERF	1.84E-06	-27.9%	SEIS_0-16-2	1.5	RB	3.06
		U2 SLERF	1.86E-06	-23.7%	SEIS_23-4	1.5	RB	3.81
					SEIS_23-5	1.5	RB	2.685
					SEIS_23-6	1.5	RB	3.36
					SEIS_23-6A	1.5	RB	3.36
					SEIS_23-7	1.25	RB	3.6875
					SEIS_VSLOCA*	1.5	RB	3.06
28	ACB Scale Factors Increased	U1 SCDF	4.06E-06	-1.3%	This sensitivity increased the scale factors of the following fragility groups:			
		U2 SCDF	4.79E-06	-1.5%	Item	Scale Factor	Building	New Fragility
		U1 SLERF	2.21E-06	-13.4%	SEIS_0-30-1	1.75	ACB	2.275
		U2 SLERF	2.21E-06	-9.4%	SEIS_0-30-10	1	ACB	0.84
					SEIS_0-30-11	1	ACB	0.71
					SEIS_0-30-5	1	ACB	0.84
					SEIS_0-30-6	1	ACB	0.63
					SEIS_14-1-2	3	ACB	3.21
					SEIS_16-1	1.75	ACB	2.9925
					SEIS_19-9	1	ACB	1.04
					SEIS_1A-5w	1.75	ACB	2.45
					SEIS_1C-1-2w	1.75	ACB	2.625
					SEIS_3-4-1	1	ACB	0.8
					SEIS_3-4-2a	1	ACB	0.8
					SEIS_4-3	1	ACB	1
					SEIS_5-4	1.75	ACB	2.31
					SEIS_FLEX_BUS480	1.75	ACB	2.5375
29	DGB Scale Factors Increased	U1 SCDF	4.02E-06	-2.3%	This sensitivity increased the scale factors of the following fragility groups:			
		U2 SCDF	4.77E-06	-2.0%	Item	Scale Factor	Building	New Fragility
		U1 SLERF	2.43E-06	-4.8%	SEIS_0-12	2	DGB	2.14

Table 5.7-1 Sensitivity Study Results Summary

<table>
<tr><th>Case #</th><th>Description</th><th>Risk Measure</th><th>Sensitivity SCDF/SLERF</th><th>Percent Change from Base Mode</th><th>Discussion</th></tr>
<tr><td></td><td></td><td>U2 SLERF</td><td>2.35E-06</td><td>-3.6%</td><td><table>
<tr><td>SEIS_19-8</td><td>2</td><td>DGB</td><td>2.48</td></tr>
<tr><td>SEIS_1A-4</td><td>2</td><td>DGB</td><td>2.04</td></tr>
</table></td></tr>
<tr><td rowspan="4">30</td><td rowspan="4">Combines cases 27, 28, 29 and adds scale factors for VSLOCA and the Yard</td><td>U1 SCDF</td><td>2.91E-06</td><td>-29.3%</td><td rowspan="4">This sensitivity combined increasing the scale factors of cases 27-29, and in addition increased the following scale factors:
<table>
<tr><th>Item</th><th>Scale Factor</th><th>Building</th><th>New Fragility</th></tr>
<tr><td>SEIS_0-1</td><td>1</td><td>YARD</td><td>1.73</td></tr>
<tr><td>SEIS_0-12</td><td>2</td><td>DGB</td><td>2.14</td></tr>
<tr><td>SEIS_0-16-2</td><td>1.5</td><td>RB</td><td>3.06</td></tr>
<tr><td>SEIS_0-30-1</td><td>1.75</td><td>ACB</td><td>2.275</td></tr>
<tr><td>SEIS_0-30-10</td><td>1</td><td>ACB</td><td>0.84</td></tr>
<tr><td>SEIS_0-30-11</td><td>1</td><td>ACB</td><td>0.71</td></tr>
<tr><td>SEIS_0-30-5</td><td>1</td><td>ACB</td><td>0.84</td></tr>
<tr><td>SEIS_0-30-6</td><td>1</td><td>ACB</td><td>0.63</td></tr>
<tr><td>SEIS_14-1-2</td><td>3</td><td>ACB</td><td>3.21</td></tr>
<tr><td>SEIS_16-1</td><td>1.75</td><td>ACB</td><td>2.9925</td></tr>
<tr><td>SEIS_19-1</td><td>1.25</td><td>YARD</td><td>0.9875</td></tr>
<tr><td>SEIS_19-5</td><td>1.25</td><td>YARD</td><td>1.3875</td></tr>
<tr><td>SEIS_19-8</td><td>2</td><td>DGB</td><td>2.48</td></tr>
<tr><td>SEIS_19-9</td><td>1</td><td>ACB</td><td>1.04</td></tr>
<tr><td>SEIS_1A-4</td><td>2</td><td>DGB</td><td>2.04</td></tr>
<tr><td>SEIS_1A-5w</td><td>1.75</td><td>ACB</td><td>2.45</td></tr>
<tr><td>SEIS_1C-1-2w</td><td>1.75</td><td>ACB</td><td>2.625</td></tr>
<tr><td>SEIS_23-4</td><td>1.5</td><td>RB</td><td>3.81</td></tr>
<tr><td>SEIS_23-5</td><td>1.5</td><td>RB</td><td>2.685</td></tr>
<tr><td>SEIS_23-6</td><td>1.5</td><td>RB</td><td>3.36</td></tr>
<tr><td>SEIS_23-6A</td><td>1.5</td><td>RB</td><td>3.36</td></tr>
<tr><td>SEIS_23-7</td><td>1.25</td><td>RB</td><td>3.6875</td></tr>
<tr><td>SEIS_3-4-1</td><td>1</td><td>ACB</td><td>0.8</td></tr>
<tr><td>SEIS_3-4-2a</td><td>1</td><td>ACB</td><td>0.8</td></tr>
</table></td></tr>
<tr><td>U2 SCDF</td><td>3.73E-06</td><td>-23.3%</td></tr>
<tr><td>U1 SLERF</td><td>1.79E-06</td><td>-29.9%</td></tr>
<tr><td>U2 SLERF</td><td>1.77E-06</td><td>-27.4%</td></tr>
</table>

Table 5.7-1 Sensitivity Study Results Summary

<table>
<tr><th>Case #</th><th>Description</th><th>Risk Measure</th><th>Sensitivity SCDF/SLERF</th><th>Percent Change from Base Mode</th><th>Discussion</th></tr>
<tr><td></td><td></td><td></td><td></td><td></td><td>
<table>
<tr><td>SEIS_4-3</td><td>1</td><td>ACB</td><td>1</td></tr>
<tr><td>SEIS_5-4</td><td>1.75</td><td>ACB</td><td>2.31</td></tr>
<tr><td>SEIS_FLEX_BUS480</td><td>1.75</td><td>ACB</td><td>2.5375</td></tr>
<tr><td>SEIS_LOSP</td><td>1</td><td>YARD</td><td>0.3</td></tr>
<tr><td>SEIS_VSLOCA</td><td>1.5</td><td>RB</td><td>3.06</td></tr>
</table>
</td></tr>
<tr><td rowspan="4">31</td><td rowspan="4">Evaluation of updated fragilities.</td><td>U1 SCDF</td><td>3.92E-06</td><td>2.5%</td><td rowspan="4">The following fragility groups were changed based on updates self-identified following the final quantification.
<table>
<tr><th>FragGroup ID</th><th>A_m used in Base Case model</th><th>Updated A_m</th></tr>
<tr><td>SEIS_0-30-5</td><td>0.84</td><td>1.18</td></tr>
<tr><td>SEIS_0-30-15</td><td>1.83</td><td>1.88</td></tr>
<tr><td>SEIS_1A-4</td><td>1.02</td><td>1.07</td></tr>
</table>
The following beta values of fragility group SEIS_0-1 was changed based on updates self-identified after quantification during closure review:
<table>
<tr><th>FragGroup ID</th><th>Beta used in Base Case model</th><th>Updated Beta Value</th></tr>
<tr><td>βu</td><td>0.43</td><td>0.51</td></tr>
<tr><td>βr</td><td>0.19</td><td>0.51</td></tr>
</table>
These calculations were performed using the simplified one top model that has slightly different base risk values.</td></tr>
<tr><td>U2 SCDF</td><td>3.88E-06</td><td>2.5%</td></tr>
<tr><td>U1 SLERF</td><td>3.70E-06</td><td>2.5%</td></tr>
<tr><td>U2 SLERF</td><td>3.65E-06</td><td>3.0%</td></tr>
</table>

5.8 SPRA Logic Model and Quantification Technical Adequacy

The SQN SPRA risk quantification and results interpretation methodology [1] were subjected to an independent peer review against the pertinent requirements in the ASME/ANS PRA Standard [8]. The risk quantification and results interpretation methodology were peer reviewed relative to Capability Category II for the full set of supporting requirements in the Standard. After completion of the subsequent independent assessment, the full set of supporting requirements was met, and the seismic hazard analysis was determined to be acceptable for use in the SPRA.

The peer review assessment, and subsequent disposition of peer review findings through an independent assessment, is further described in Appendix A, and references [7] and [20].

6.0 Conclusions

A SPRA has been performed for SQN in accordance with the guidance in the ASME/ANS PRA Standard [8] and the SPID [3]. The SPRA shows that the point estimate SCDF is 4.1E-06 per reactor year (ry) for Unit 1 and is 4.9E-06 per ry for Unit 2 [1]. The SLERF is 2.6E-06 per ry for Unit 1 and is 2.4E-06 per ry for Unit 2 [1].

Appendix A includes as assessment of plant changes not included in the model and how the changes impact the model results.

No seismic hazard vulnerabilities were identified, and no plant actions have been taken or are planned given the insights from this study.

7.0 References

1) TVA Calculation MDN0009992017000045, Revision 1, "SQN Seismic PRA Quantification, Sensitivity and Uncertainty Notebook," June 11, 2019.

2) NRC 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 (ML12056A046).

3) 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.

4) TVA Letter to NRC, "Tennessee Valley Authority's Seismic Hazard and Screening Report (CEUS Sites), Response to NRC Request for Information Pursuant to 10CFR50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," March 31, 2014 (ML14098A478).

5) TVA Letter to NRC, "Spent Fuel Pool Evaluation Supplemental Report for Sequoyah Nuclear Plant, Units 1 and 2 - Response to NRC Request for Information Pursuant to Title 10 of the Code of Federal regulations 50.54(f) Regarding Recommendation 2.1 of the Near-Term, Task Force Review of Insights from the Fukushima Dai-ichi Accident," December 23, 2016 (ML16361A204).

6) NRC Letter to TVA, "Sequoyah Nuclear Plant, Units 1 and 2 - Staff Review of Spent Fuel Pool Evaluation Associated with Reevaluated Seismic Hazard Implementing Near-Term Task Force Recommendation 2.1 (CAC Nos. MF3767 and MF3768)," February 16, 2017 (ML17041A387).

7) PWROG-18002-P, Revision 0, "Peer Review of the Sequoyah Seismic Probabilistic Risk Assessment," Westinghouse PWR Owners Group, Risk Management Committee, July 2018.

8) ASME/ANS RA-S-2008, "Standard for Level 1 / Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications", American Society of Mechanical Engineers, September 30, 2013.

9) NEI 12-13, "External Hazards PRA Peer Review Process Guidelines", Revision 0, Nuclear Energy Institute, August 2012.

10) NRC Letter to NEI, "U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 07-12, and 12-13, Close-Out of Facts and Observations (F&Os)," May 3, 2017 (ML17079A427).

11) NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," March 2017.

12) NUREG-2115, "Central and Eastern United States Seismic Source Characterization for Nuclear Facilities", January 2012.

13) Regulatory Guide 1.208, Revision 0, “A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion,” March 2007.

14) EPRI 3002005684, “Central and Eastern United States Seismic Source Characterization for Nuclear Facilities: Maximum Magnitude Distribution Evaluation,” Electric Power Research Institute, June 29, 2015.

15) EPRI 3002000717, “EPRI (2004, 2006) Ground-Motion Model (GMM) Review Project,” Electric Power Research Institute, June 13, 2013.

16) EPRI Letter, “Errata Sheet for EPRI Technical Report EPRI (2004, 2006) Ground-Motion Model (GMM) Review Project, Product 3002000717,” Electric Power Research Institute, July 23, 2013.

17) NUREG/CR-6372, “Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts,” April 1997.

18) NRC Letter to TVA, “Sequoyah Nuclear Plant, Units 1 and 2 - Staff Assessment of Information Provided Pursuant to Title 10 of The Code of Federal Regulations, Part 50.56(f), Seismic Hazard Reevaluations Relating to Recommendation 2.1 of The Near-Term Task Force Review of Insights from The Fukushima Dai-ichi Accident (TAC Nos. MF3767 and MF3768),” April 27, 2015 (ML15098A641).

19) NUREG-2117, Revision 1, “Practical Implementation Guidelines for SSHC Level 3 and 4 Hazard Studies,” April 2012.

20) Jensen-Hughes Report 006063-RPT-01, Revision 0, “Sequoyah SPRA F&O Independent Assessment & Focused Scope-Peer Review,” April 2019.

21) Fugro PR No. 150017-PR-01, Revision 3, “Probabilistic Seismic Hazards Analysis for TVA Sequoyah Nuclear Plant PSHA Results Report,” February 2019.

22) EPRI 3002000709, “Seismic Probablistic Risk Assessment Implementation Guide,” Electric Power Research Institute, December 18, 2013.

23) TVA Calculation MDN0009992017000047, Revision 1, “SQN Probabilistic Risk Assessment - Seismic Equipment List (SEL) Report,” June 11, 2019.

24) ENERCON Report TVAESQN010-REPT-002, Revision 2, “Sequoyah Units 1 & 2 Seismic PRA Chatter Analysis Report,” July 2018.

25) Seismic Qualification Utility Group (SQUG), Revision 3A, “Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment,” December 2001.

26) EPRI NP-6041-SL, Revision 1, “A Methodology for Assessment of Nuclear Power Plant Seismic Margin,” Electric Power Research Institute, August 1991.

27) Regulatory Guide 1.200, Revision 2, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” March 2009.

28) ENERCON Report TVAESQN010-REPT-003, Revision 2, “Seismic Walkdown Report of Units 1 & 2 Sequoyah Nuclear Power Plant for Seismic Probabilistic Risk Assessment,” April 2019.

29) ASCE/SEI 4-98, “Seismic Analysis of Safety-Related Nuclear Structures,” American Society of Civil Engineers, 1998.

30) ASCE/SEI 4-16, “Seismic Analysis of Safety-Related Nuclear Structures,” American Society of Civil Engineers, 2017.

31) SC Solutions Report SQN-17-001, Revision 2, “Sequoyah SPRA - Structural Response Analysis Report,” November 2018.

32) TVA Sequoyah Design Criteria SQN-DC-V-46.0, Revision 2, “Seismic/Structural Qualification of Seismic Category I and I(L) In-Line Valves and Other In-Line Fluid System Components,” March 4, 2015.

33) EPRI 1019200, “Seismic Fragility Application Guide Update,” Electric Power Research Institute, December 2009.

34) EPRI TR-103959, “Methodology for Developing Seismic Fragilities,” Electric Power Research Institute, June 1994.

35) EPRI Report 3002004396, “High Frequency Program: Application Guidance for Functional Confirmation and Fragility Evaluation,” Electric Power Research Institute, July 2015.

36) ENERCON Report TVAESQN010-REPT-001, Revision 1, “Units 1 & 2 Sequoyah Nuclear Power Plant Seismic Probabilistic Risk Assessment – Fragility Analysis Notebook,” February 2019.

37) NUREG/CR-6544, “A Methodology for Analyzing Precursors to Earthquake-Initiated and Fire-Initiated Accident Sequences”, April 1998.

38) TVA Calculation MDN000NA2017000042, Revision 1, “SQN Seismic PRA Seismic-Fire Interaction,” March 22, 2019.

39) TVA Calculation MDN00000020100203, Revision 3, “SQN Internal Flooding,” August 18, 2014.

40) TVA PRA Evaluation SQN-0-17-095, Revision 0, “The Selection of Seismic Flooding Scenarios,” September 7, 2017.

41) EPRI 3002008093, “An Approach to Human Reliability Analysis for External Events with a Focus on Seismic,” Electric Power Research Institute, December 2016.

42) TVA Calculation MDN0009992017000043, Revision 1, “SQN Probabilistic Risk Assessment - Seismic PRA Human Reliability Analysis,” January 23, 2019.

43) EPRI 1016737, “Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments,” Electric Power Research Institute, December 2008.

44) NUREG/CR-4840, “Procedures for the External Event Core Damage Frequency Analyses for NUREG-1150,” November 1990.

45) NUREG/CR-6728, “Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-Consistent Ground Motion Spectra Guidelines,” October 2001.

46) Report CJC-SQN-C-001, Revision 1, "Updated Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant (SQN)," April 2019.

47) TVA Calculation MDN00000020100200, Revision 3, "SQN Probabilistic Risk Assessment – Summary Notebook," August 4, 2014.

8.0 Acronyms and Abbreviations

ACAS	Auxiliary Control Air Subsystem
ACB	Auxiliary-Control Building
ADGB	Additional Diesel Generator Building
AEB	Additional Equipment Building
AFE	Annual Frequency of Exceedance
AFW	Auxiliary Feedwater
Am	Median Acceleration Capacities
ANS	American Nuclear Society
AOV	Air-Operated Valve
ARV	Atmospheric Relief Valve
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
CARF	Containment Air Return Fans
CCDP	Conditional Core Damage Probability
CCP	Centrifugal Charging Pump
CCS	Component Cooling System
CDF	Core Damage Frequency
CDFM	Conservative Deterministic Failure Margin
CET	Containment Event Tree
CEUS	Central and Eastern United States
CEUS-SSC	Central and Eastern United States Seismic Source Characterization
CG	Center of Gravity
CHR	Containment Heat Removal
CLERP	Conditional Large Early Release Probability
CR	Center of Rigidity
CS	Containment Spray
CSB	Concrete Shield Building
CST	Condensate Storage Tank
CVCS	Chemical Volume Control System
DG	Diesel Generator

DGB	Diesel Generator Building
DM	Direct Method
DOE	Department of Energy
Dp	Compression-Wave Damping
Ds	Shear-Wave Damping
EDG	Emergency Diesel Generator
EF	Error Factor
EPRI	Electric Power Research Institute
ERCW	Essential Raw Cooling Water
ESEP	Expedited Seismic Evaluation Process
ESVR	East Steam Valve Room
ET	Event Tree
FEM	Finite Element Model
F&O	Facts and Observations
FIRS	Foundation Input Response Spectra
FLEX	Diverse and Flexible Coping Strategies
FSAR	Final Safety Analysis Report
F-V	Fussell-Vesely
GIP	Generic Information Procedure
GMC	Ground Motion Characterization
GMRS	Ground Motion Response Spectra
GTRAN	General Transient
HEP	Human Error Probability
HCLPF	High Confidence of Low Probability of Failure
HF	High Frequency
HFE	Human Failure Event
HLR	High-Level Requirements
HRA	Human Reliability Analysis
HVAC	Heating, Ventilating, and Air Conditioning
IAEA	International Atomic Energy Agency
ICS	Internal Concrete Structure
IE	Initiating Event
IEPRA	Internal Events Probabilistic Risk Assessment

IIP	Integrated Interaction Program
IPEEE	Individual Plant Examination for External Events
IPS	Intake Pumping Station
ISLOCA	Interfacing System Loss of Coolant Accident
ISRS	In-Structure Response Spectra
JCNRM	Joint Committee on Nuclear Risk Management
LAR	License Amendment Request
LB	Lower Bound
LERF	Large Early Release Frequency
LERP	Large Early Release Probability
LF	Low Frequency
LHS	Latin Hybercube Simulation
LLOCA	Large Loss of Coolant Accident
LOCA	Loss of Coolant Accident
LOSP	Loss of Offsite Power
LMSM	Lumped Mass Stick Model
LVS	Low Voltage Switchgear
MAFE	Mean Annual Frequency of Exceedance
MCC	Motor Control Center
MDAFW	Motor-Driven Auxiliary Feedwater
MLOCA	Medium Loss of Coolant Accident
MOV	Motor-Operated Valve
MSL	Mean Sea Level
MVS	Medium Voltage Switchgear
MW	Mega-watt
NEI	Nuclear Energy Institute
NRC	United States Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NSVR	North Steam Valve Room
NTTF	Near Term Task Force
OBE	Operating Basis Earthquake
PAF	Plant Availability Factor
PDS	Plant Damage States

PGA	Peak Ground Acceleration
PORV	Power-Operated Relief Valve
PRA	Probabilistic Risk Assessment
PSHA	Probabilistic Seismic Hazard Analysis
PWR	Pressurized Water Reactor
PWROG	PWR Owners Group
RAW	Risk Achievement Worth
RB	Reactor Building
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RG	Regulatory Guide
RHR	Residual Heat Removal
RLE	Review Level Earthquake
RLGM	Review Level Ground Motion
RLME	Repeated Large Magnitude Earthquakes
RMOV	Reactor Motor-Operated Valve
RPS	Reactor Protection System
RWST	Refueling Water Storage Tank
SASSI	System for Analysis for Soil-Structure Interaction
SBO	Station Blackout
SC-SASSI	SC Solutions version of SASSI computer model
SCV	Steel Containment Vessel
SEL	Seismic Equipment List
SFR	Seismic Fragility Element Within ASME/ANS PRA Standard
SGTR	Steam Generator Tube Rupture
SHA	Seismic Hazard Analysis Element Within ASME/ANS PRA Standard
SI	Safety Injection
SIET	Seismic Initiating Event Tree
SMA	Seismic Margin Assessment
SMM	Spectrally Matched Motions
SLOCA	Small Loss of Coolant Accident
SLOCAV	Very Small Loss of Coolant Accident (also VSLOCA)
SoV	Separation of Variables

SOV	Solenoid-Operated Valve
SPID	Screening, Prioritization and Implementation Details
SPR	Seismic PRA Modeling Element Within ASME/ANS PRA Standard
SPRA	Seismic PRA
SQN	Sequoyah Nuclear Plant
SQUG	Seismic Qualification Utility Group
SR	Supporting Requirement
SRSS	Square-Root-Sum-of-Squares
SRT	Seismic Review Team
SSC	Seismic Source Characterization
SSC	Structures, Systems and Components
SSE	Safe Shutdown Earthquake
SSEL	Safe Shutdown Equipment List
SSHAC	Senior Seismic Hazard Analysis Committee
SSPS	Solid State Protection System
SSI	Soil-Structure Interaction
TB	Turbine Building
TDAFW	Turbine-Driven Auxiliary Feedwater
TH	Time History
TI-SGTR	Thermally-Induced Steam Generator Tube Rupture
TVA	Tennessee Valley Authority
UB	Upper Bound
UHRS	Uniform Hazard Response Spectrum
V	Volt
V/H	Vertical/Horizontal
Vp	Compression-wave Velocity
Vs	Shear-wave Velocity
VSLOCA	Very Small Loss of Cooling Accident (also SLOCAV)
WBN	Watts Bar Nuclear Plant

Appendix A

Summary of SPRA Peer Review and Assessment of PRA Technical Adequacy for Response to NTTF 2.1 Seismic 50.54(f) Letter

A.1 Introduction

This Appendix provides a summary of the SPRA peer review and Facts & Observations (F&O) closure reviews and provides the bases for why the SPRA is technically adequate for the 50.54(f) response.

A.2 Peer Review of SQN SPRA

The Sequoyah Nuclear Plant (SQN) PRA was subjected to an independent peer review against the pertinent requirements in Part 5 of ASME/ANS PRA RA-Sb-2013 [8]. The peer review assessment [7], and subsequent disposition of peer review findings, are summarized here. The scope of the review encompassed the set of technical elements and supporting requirements (SRs) for the seismic hazard analysis (SHA), seismic fragilities analysis (SFR), and SPRA modeling (SPR) technical elements for seismic SCDF and SLERF. The peer review, therefore, addressed the set of SRs identified in Tables 6-4 through 6-6 of the SPID [3].

The information presented here establishes that the SPRA has been peer reviewed by a team with adequate credentials to perform the assessment, establishes that the peer review process followed meets the intent of the peer review characteristics and attributes in Table 16 of RG 1.200 R2 [27] and the requirements in Section 1-6 of the ASME/ANS PRA Standard [8], and presents the significant results of the peer review.

The SQN SPRA peer review was conducted during the week of April 23, 2018, at the TVA offices in Chattanooga, Tennessee. As part of the peer review, a walkdown of portions of SQN Units 1 and 2 was performed on April 24, 2018, by several members of the peer review team who have the appropriate Seismic Qualification Users Group (SQUG) training.

A.2.1 Summary of the SQN SPRA Peer Review Process

The peer review was performed against the requirements in Part 5 (Seismic) of Addenda B of the PRA Standard [8], using the peer review process defined in NEI 12-13 [9]. The review was conducted over a four-day period, with a summary and exit meeting on the morning of the fifth day.

The SPRA peer review process defined in [9] involves an examination by each reviewer of their assigned PRA technical elements against the requirements in the Standard to ensure the robustness of the model relative to all the requirements.

Implementing the review involves a combination of a broad scope examination of the PRA elements within the scope of the review and a deeper examination of portions of the PRA elements based on what is found during the initial review. The SRs, in combination with the peer reviewers' PRA experience, provide the structure and basis for examining the various PRA technical elements. If a reviewer identifies a question or discrepancy, then that issue is further investigated until it is resolved or an F&O is written describing the issue and its potential impacts with suggestions for possible resolutions.

For each technical element, i.e., SHA, SFR, SPR, at least two peer reviewers were assigned, with one having lead responsibility for a given area. For each SR reviewed, the responsible reviewers reached consensus regarding which of the Capability Categories defined in the Standard the PRA meets for that SR, and the assignment of the Capability Category for each SR was ultimately based on the consensus of the full review team. The Standard also specifies high-level requirements (HLR). Consistent with the guidance in the Standard, Capability Categories were not assigned to the HLRs, but a qualitative assessment of the applicable HLRs in the context of the PRA technical element summary was made based on the associated SR Capability Categories.

As part of the review team's assessment of Capability Categories, F&Os were prepared. There are three types of F&Os defined in [9]: Findings, which identify issues that must be addressed for an SR (or multiple SRs) to meet Capability Category II; Suggestions, which identify issues that the reviewers have noted as potentially important but not requiring resolution to meet the SRs; and Best Practices, which reflect the reviewers' opinion that a particular aspect of the review exceeds normal industry practice. The focus in this Appendix is on Findings and their disposition relative to this submittal.

A.2.2 Peer Review Team Qualifications

The review was conducted by Dr. Andrea Maioli of Westinghouse, Dr. Annie Kammerer of Annie Kammerer Consulting, Dr. Gabriel Toro of Lettis Consultants International, Mr. Greg Hardy of Simpson Gumpertz & Heger, Mr. William Horstman and Mr. Nathan Barber of Pacific Gas & Electric, Mr. Adam Helffrich of RIZZO, and Mr. Edmond Wiegert of Duke Energy.

Dr. Andrea Maioli, the team lead, has over 11 years of experience at Westinghouse in nuclear safety and PRA for both existing and new nuclear power plants. He is the Westinghouse technical lead for all SPRA activities. He has supported and led peer reviews for internal events, internal flooding, fire PRAs, high winds and other external hazards as well as SPRAs. He is a member of the ASME/ANS Joint Committee on Nuclear Risk Management (JCNRM) and the JCNRM Subcommittee on Standard Maintenance, which is maintaining the ASME/ANS PRA Standard.

Dr. Annie Kammerer was the lead for the review of the SHA technical element. Dr. Kammerer is an expert in seismic hazard and risk and integrated performance-based, risk-informed engineering, particularly as applied to nuclear facilities. She has over 15 years of experience. She is an independent consultant, as well as a visiting scholar at the Pacific Earthquake Engineering Research Center at the University of California, Berkeley. She spent 7 years at the NRC, where she developed and coordinated the NRC Seismic Research Program. Dr. Kammerer has served on several national- and international-level committees and working groups. She leads the seismic hazard working group for updates to the ASME/ANS JCNRM SPRA standard. Dr. Kammerer has served as peer reviewer for multiple SPRAs.

Dr. Kammerer was assisted in the SHA review by Dr. Gabriel Toro. Dr. Toro has more than 38 years of experience in the development of probabilistic models for natural hazards, particularly earthquakes, working in close cooperation with earth scientists who are experts in the natural phenomenon being evaluated. Dr. Toro's experience includes leading the Technical Integration Team for ground motion characterization in the Senior

Seismic Hazard Analysis Committee (SSHAC) Level 3 seismic hazard study for a proposed new nuclear plant in the inter-mountain western United States. Dr. Toro has supported F&Os closure activities for different plants through the PWR Owners Group (PWROG).

Mr. Greg Hardy was the lead for the review of the SFR technical element. Mr. Hardy has more than 40 years of experience in structural mechanics engineering. His responsibilities have included natural hazards PRAs, earthquake experience data-based studies, aircraft impact analyses, stress analysis, finite element analysis, seismic margin studies, and shock and vibration environmental testing for hardware qualification. He has been a principal consultant in the area of structural mechanics to highly protected industries such as Nuclear, Defense, and the Department of Energy (DOE). He has also consulted with EPRI, NEI, and the International Atomic Energy Agency (IAEA) and has participated in development and peer review of multiple SPRAs.

Mr. Hardy was assisted in the fragility review by Mr. William Horstman and Mr. Adam Helffrich. Mr. William Horstman is a Geosciences Consultant Expert in the Geosciences Department at Pacific Gas and Electric (PG&E). He has more than 30 years of experience in the fields of structural engineering and structural mechanics, primarily at the Diablo Canyon Power Plant, where he was in charge of the fragility aspect of the Diablo Canyon SPRA. Mr. Horstman served as reviewer for the Vogtle SPRA peer review through the PWROG. Mr. Helffrich has 7 years of structural engineering experience. Mr. Helffrich has supported the Fermi and Watts Bar SPRAs and defended the respective peer reviews.

Mr. Nathan Barber was the lead for the review of the SPR technical element. Mr. Barber is the Senior Advising Engineer for probabilistic Risk Assessment at the PG&E Diablo Canyon Power Plant, and the model owner for the Diablo Canyon SPRA. He has more than 15 years of experience in PRA, defended peer reviews of the Diablo Canyon PRA models and supported PRA peer reviews through the PWROG (e.g., SPRA peer review for the Beaver Valley plant). He was assisted in the SPR review by Mr. Edmond Wiegert. Mr. Wiegert has 25 years of experience in the nuclear industry and 18 years of experience in multiple areas of PRA. Mr. Wiegert is the lead engineer in the PRA applications and models group at Duke Energy. Mr. Wiegert has been supporting the SPRA modeling task for the Duke plants and supported numerous peer reviews (e.g., SPRA peer review for the North Anna plant).

Two working observers (Dr. Yigit Isbiliroglu from RIZZO and Ms. Rachel Christian from Westinghouse Electric Company) supported the review of the SFR and SPR technical elements, respectively. Ms. Christian also acted as the review lead in training. Any observations and findings these working observers generated were given to the peer review team for their review and "ownership." As such, Dr. Isbiliroglu and Ms. Christian assisted with the review but were not formal members of the peer review team.

The peer review team members met the peer reviewer independence criteria in NEI 12-13 [9].

A.2.3 Summary of the Peer Review

The review team's assessment of the SPRA elements is excerpted from the peer review report as follows. Where the review team identified issues, these are captured in peer review findings, the dispositions for which are summarized in the next section of this appendix.

A.2.3.1 Seismic Hazard Analysis (SHA)

As required by the Standard, the frequency of occurrence of earthquake ground motions at the site was based on a probabilistic seismic hazard analysis (PSHA). The seismic source characterization (SSC) inputs to the PSHA are based on the Central and Eastern U.S. (CEUS) regional SSC model published in U.S. Nuclear Regulatory Commission Regulation (NRC Document Series) NUREG-2115 [12] (i.e., the "CEUS-SSC" model), with updates described in EPRI, 2015 [14]. The ground motion characterization (GMC) inputs to the PSHA are based on an updated model published in 2013 by EPRI's CEUS ground motion update project (EPRI, 2013a and 2013b, [15, 16]). The seismic hazard analysis for the SQN site also accounts for the effects of local site response.

The SSHAC methodology defines a process of structured expert interaction (elicitation) that is considered a minimum technical requirement for conduct of a PSHA. The SSHAC process (NUREG/CR-6372 [17], and NUREG-2117 [19] of conducting a PSHA was used to develop both the SSC and GMC models used as inputs to the analysis. Use of the SSHAC methodology ensures that data, methods and models supporting the PSHA are fully incorporated and that uncertainties are fully considered in the process at sufficient depth and detail necessary to satisfy scientific and regulatory needs. The SSHAC-related guidance documents define and describe four "levels." The level of study is not mandated in the Standard; however, both the SSC and the GMC parts of the PSHA were developed using a SSHAC Level 3 analysis. In the case of the GMC, a SSHAC Level 2 analysis was carried out to update a prior Level 3 study. These Level 3 studies satisfy the requirements of the Standard related to the general method of conduct of the PSHA, as well as address several individual requirements related to data collection, data evaluation and model development, and quantification of uncertainties supporting HLR-SHA-A to HLR-SHA-D.

As a first step to performing a PSHA, the Standard requires an up-to-date database, including regional geological, seismological, geophysical data, and local site topography, and a compilation of information on surficial geologic and geotechnical site properties. These data include a catalog of relevant historical, instrumental, and paleo-seismic information within 320 km of the site. The CEUS-SSC study involved an extensive data collection effort that satisfies the requirements of the Standard as it relates to developing a regional-scale seismic source model.

In the implementation of the CEUS-SSC model for the SQN site, all distributed seismic sources in the CEUS-SSC model with the appropriate distance were included in the PSHA calculations. By including these seismic sources in the analysis, the contribution of "near-" and "far-field" earthquake sources to ground motions at the SQN site were considered. An effort was made to identify local seismic sources that may not have been included in the regional model. Additional information pertinent to the site response analyses was collected and assessed.

The CEUS-SSC and EPRI regional models discussed above were used for the SQN site PSHA. Even though the PSHA conducted was performed specifically for the SQN site, the underlying models were existing models and the seismicity database that underpins significant aspects of the CEUS-SSC only includes earthquakes through 2008. According to SHA-H1, if an existing model is used, a data collection and evaluation effort should be conducted to determine: (1) whether new information has become available since the data was compiled for the existing model and, if so, (2) whether any new information challenges the validity of the technical basis of the existing study. It is not the case that identification of new data automatically requires an update to the PSHA existing model. Rather, an evaluation of the new data determines whether the existing model is appropriate for its continued use in the intended application. In the case of the PSHA for the SQN site, this data collection and review was performed for earthquakes through January 2015 to support the continued use of the CEUS-SSC as a basis for the PSHA, given that the CEUS-SSC catalog was finalized in 2008. Supporting requirement SHA-H1 was, therefore, found to be met for the SQN site. In the case of SQN, an effort was made to collect and assess information that would provide insight into the possible vibratory ground motion hazard from induced or triggered earthquakes (e.g., the presence or absence of injection wells in the area). However, the study was not fully documented, leading to a documentation finding.

The PSHA results are provided over an appropriately wide range of spectral frequencies and annual frequencies of exceedances (AFEs). Uncertainties on the rock hazard are quantified, analyzed and reported as required in the Standard. The lower-bound magnitude chosen for the analysis is consistent with standard practice. The results include fractile, median and mean hazard curves, and uniform hazard response spectra (UHRS). As a result of the above, SHA-F1 and SHA-F3 are met.

The SHA for the SQN site included a site response analysis. As part of the characterization of the site, historical, site-specific shear-wave velocity measurements and information were used to inform the site response analysis. The analysis includes the effects of site topography, surficial geologic deposits, and site geotechnical properties on ground motions at the site. The Standard requires that spectral shapes be based on a site-specific evaluation considering the contributions of de-aggregated magnitude-distance results of the PSHA The PSHA fully accounted for the “near-” and “far-field” source spectral shapes. The horizontal UHRS used in the SPRA is based on site-specific results and incorporates analysis results for all spectral frequencies. To ensure that the spectral shape captures potential site response effects, the results for the seven (7) spectral frequencies used to calculate the hard-rock hazard were supplemented with additional spectral frequencies via interpolation and extrapolation. The vertical to horizontal (V/H) ratios used to calculate vertical response spectra were consistent with current practice. Both the aleatory and epistemic uncertainties have been addressed in characterizing the seismic sources, ground motion models, and site response analyses. These uncertainties were carried through the PSHA quantification. As a result, SHA-E1, SHA-E2, SHA-G1 are met.

The seismic hazard analysis for the SQN site included several sensitivity analyses, namely rock hazard by source, rock hazard by ground motion prediction equation, sensitivity to the recurrence parameters and maximum magnitude of the dominant seismic source, and additional rock sensitivities. In addition, the rock hazard calculations are based on the CEUS-SSC and EPRI GMC models. During the development of these models, uncertainties in the seismic sources and ground motion prediction equations were included and appropriate sensitivity analyses were performed for a hypothetical site in Chattanooga, Tennessee, to demonstrate the sensitivity of the results to uncertainties in key model parameters. These sensitivity analyses, which may be considered applicable to SQN, are documented in [12] and [14]. In addition, the sensitivity of the calculated surface hazard curves to alternative soil profiles and degradation properties was calculated and displayed for the Ground Motion Response Spectra (GMRS) and all Foundation Input Response Spectra (FIRS). As a result of the above, SHA-F2 is met.

Supporting requirement SHA-I1 addresses the bases and methodology used for any screening of the seismic hazards other than vibratory ground motion. A screening-level analysis of other soil-related hazards, including the potential for triggering liquefaction, settlement, lateral spread, bearing capacity and structural sliding, slope stability, sand upstream dam failure, was provided, and these hazards screened into further evaluation. The potential for direct fault rupture, tsunami, seiche, low water levels, and landslide was also assessed and screened out.

There were several technical issues with the evaluation of slope stability and the evaluation of liquefaction and its associated phenomena identified, such that revised evaluations are appropriate. In the case of liquefaction, the use of alternative methods for evaluation of triggering should be considered. The technical basis for several assumptions related to the secondary phenomena stated in the report should be provided. Additionally, based on SPRA quantification results, evaluations of secondary phenomena provided did not extend to large enough ground motion levels to ensure that the risk associated with site phenomena was appropriately captured in the SPRA. The range of amplitudes of the parameters needed for fragility calculations was not provided. As a result of the above, SHA-I2 was not met and findings were issued related to the technical issues.

The Standard requires that documentation of the PSHA that supports the PRA applications, peer review and potential future upgrades of the seismic hazard analysis be provided. This requirement establishes a high standard for documentation of the PSHA that allows for examination of the PSHA methodology, its implementation, and the PSHA results to evaluate whether the approach is appropriate, the analyses were performed correctly, and the results are reasonable. The documentation provided was generally of sufficient quality, clarity, and completeness to allow for a peer review to be conducted. However, several additions and corrections that must be made to the documentation are listed in a finding under SHA-J2. Suggestions of improvement of the documentation are also provided under SHA-J2.

A.2.3.2 Seismic Fragility Analysis (SFR)

The SQN Structural Response Analysis Report (SC Solutions Report No. SQN-17-001, Revision 1) defines the seismic response analysis methods and assumptions used for the generation of seismic responses of the plant buildings for use as input to the SQN SPRA fragility evaluations. New structural models were generated for the SQN buildings that contain seismic equipment list (SEL) components. Note that some of the structural models were adapted from those developed for the Watts Bar Nuclear Station (WBN) SPRA, which is very similar to SQN, while the remaining structural models were developed specifically for SQN. New median centered and probabilistic SSI analyses were performed for the GMRS/FIRS input. Sensitivity studies were performed at twice the

GMRS level to assess the impact of earthquakes beyond the GMRS that impact the Seismic Core Damage Frequency (SCDF) and Seismic Large Early Release Frequency (SLERF) risks. These levels of review include some of the earthquake return periods that dominate the SQN SPRA risk. However, return periods beyond those reviewed as part of the SPRA also contributed significantly to final seismic risks. As such, findings are written to establish the reference earthquake level based on the final SPRA risk results and assess the impact (if any) on the seismic response and seismic fragility results.

The SQN Fragility Analysis Notebook (ENERCON Report No. TVAESQN010-REPT-001, Revision 0) outlines the development of the seismic fragilities for the SQN SPRA. This document is subdivided into four (4) parts (i.e., summary, representative fragilities for first quantification, detailed fragilities for second quantification, and detailed fragilities for third quantification) and documents the final list of fragilities for all structures, systems, and components (SSCs) listed on the SEL, which are used as input to the SPRA model. Seismic fragilities were developed based on plant-specific data and/or generic data as available and appropriate. Realistic fragilities were developed for the majority of risk-significant SSCs, as part of an iterative process based on the successive SPRA quantifications.

Discussion of potential failure modes for each of the SSCs is provided throughout the Fragility Notebook. If multiple failure modes are identified, fragility calculations are developed for each failure mode. Generally, for the SSCs addressed in the Fragility Notebook, the relevant failure modes are stated and the basis for selecting the critical failure mode(s) is documented. A small number of instances were noted where the failure modes needed a reassessment. Plant-specific data was used in the calculation of the fragility parameters for structures and components. Fragility parameters for the major structures at SQN utilized information from WBN, with the appropriate SQN plant-specific adjustments. Fragility parameters for components utilized plant-specific design basis qualification data, along with generic equipment ruggedness spectra, where necessary. Problems with the organization of the Fragility Analysis Notebook were noted by the peer review team.

The SQN Walkdown Report (ENERCON Report No. TVAESQN010-REPT-003 [28]) outlines the methodology used for the performance of the walkdowns of SSCs, and the results of the walkdowns of these SSCs for use in the development of the seismic fragilities for the SQN SPRA. The intent of the seismic walkdowns for the SPRA is to provide an assessment of the seismic ruggedness and perform an assessment for any potential seismic interactions. The seismic walkdowns were performed and the results of the walkdowns were utilized in the fragility evaluations. The scope of the walkdowns includes all applicable SSCs on the SEL. Walkdown characteristics included anchorage, functional considerations, and seismic interactions (II/I, fire, and flood). The results of the walkdowns were documented on Screening Evaluation Worksheets and included in appendices to the Walkdown Report. Screening of high-capacity components is described in the SEL Development Calculation (TVA Nuclear Calculation No. MDN0009992017000047, Revision 0), as well as Section 3.3.4 of the Walkdown Report. The screening of high-capacity components conforms to that described in EPRI-NP-6041 [26] and meets standard industry practice. Problems with the organization of the

Walkdown Notebook and issues with certain aspects of the screening were noted by the peer review team.

The fragility-related documentation is generally presented in a manner that facilitates PRA applications, upgrades, and peer review. The processes implemented to fully develop the information needed to justify the fragility analysis is presented. This information includes detailed reports on building response, walkdowns, and fragility calculations. Instances where documentation was not complete were identified by the peer review team. To address deficiencies in the otherwise generally good documentation, multiple findings are developed to address missing documentation in the existing reports.

A.2.3.3 Seismic Plant Response Analysis (SPR)

The SQN SPRA systems logic model was developed through systematic modification of the existing at-power, internal events PRA model. Seismic initiating event frequencies, based on the results of the PSHA, were divided into nine discrete bins and incorporated into the SPRA model. Seismic fragilities for components and structures were identified during SEL development and included in the SPRA by mapping to the appropriate model impacts. A seismic initiating event tree was developed to group accident scenarios with similar characteristics and provided a link to the appropriate event trees.

The SEL was developed by including all systems from the SQN internal events PRA, the Individual Plant Examination for External Events (IPEEE) and seismic walkdown report. In addition, all other plant systems were reviewed to find previously unidentified seismic impacts. SEL development included consideration of seismically induced fire and flooding sources. For flooding, the internal events flooding PRA was used to identify sources of flooding. A separate screening process was used to identify potential seismically induced ignition sources. This process was comprehensive, but an issue was identified regarding some of the criteria used in the seismically induced fire screening.

The SQN SPRA model includes seismic initiating events, seismically induced SSC failures, seismically induced contact chatter, non-seismic random failures as well as human action failures that result in significant accident sequences. The seismic initiating event acceleration range was appropriately established and uses a lower bound equal to the SQN operating basis earthquake of 0.09g peak ground acceleration (PGA). No seismic capacity-based screening was used in the SQN SPRA resulting in a highly detailed set of seismic SSCs impacts that are included in the model.

Except for main control room abandonment human actions, all modeled operator recoveries were developed using a detailed methodology that appropriately accounts for post-earthquake stresses and cue availability. An extensive review of operator action accessibility was performed and identified the need to model seismically induced block wall failures impacting local actions. Issues associated with assumed turbine building failure and system modeling of FLEX diesels were identified.

The SQN SPRA includes modeling of very small LOCAs (VSLOCA). VSLOCAs originate from failure of RCS instrumentation tubing with less than 3/8" diameter for which a specific fragility was developed. Modeling of VSLOCAs was accomplished using a separate VSLOCA event tree and associated success criteria.

The SQN SPRA appropriately integrates the seismic hazard, fragilities and systems logic model to quantify SCDF and SLERF. When correlating seismic failures, either full correlation was assumed for components that shared similar location and failure mode attributes or zero correlation was assumed when SSCs were dissimilar. No plant-specific, partial-correlation values were used. A finding on the model quantification convergence criteria was written.

Although some minor issues were identified, the SQN SPRA documentation was assessed to be adequate to facilitate application and update of the PRA as well as provide the means for peer review.

The review team concluded that, in general, the data, methodologies and seismic risk models used for the SQN Unit 1 and 2 were appropriate and sufficient to meet most of the Standard requirements. As noted in the peer review report, the ASME/ANS PRA Standard was met for all but 10 supporting requirements. In the judgment of the peer review team, the SQN SPRA meets the remaining supporting requirements based on the SPRA methodology used, the SPRA models and results, and the detailed documentation.

The peer review team identified specific areas for improving the technical adequacy of the SPRA. These areas are documented as F&Os.

A.2.3.4 Peer Review Findings

Based on the peer review, the SQN SPRA is judged to be consistent with the ASME/ANS PRA Standard and can be used for risk-informed applications. If the areas identified for enhancements in the SPRA impact a specific risk-informed application, then additional bounding analyses may be required to support that application.

In summary, the peer review team concludes that the technical adequacy of the SQN SPRA is very good and meets most of the requirements of the ASME/ANS PRA Standard.

However, the peer review team identified specific areas for improving the technical adequacy of the SPRA. These areas are documented as F&Os. At the conclusion of the peer review, there were 54 open Finding-Level F&Os as shown in Table A-1.

Table A-1 Summary of Facts & Observations for the Sequoyah Unit 1 and 2 SPRA Peer Review

Element	F&Os			
	Findings	Suggestions	Best Practice	Total by Element
SHA[(1)]	7	2	0	9
SFR[(1)]	28	6	0	34
SPR[(1)]	25	1	0	26
TOTAL[(2)]	54	9	0	63

Notes:
(1) F&Os by element refer to linked F&Os (i.e., a single F&O can be linked to more than one SR)
(2) Total refers to unique F&Os (i.e., not linked)

A.3 Revision of Model and Documentation

Following the peer review, the SQN SPRA model and documentation were updated to address each of the 54 F&Os. In addition, TVA generated closure documentation for each of the F&Os from the peer review against the ASME/ANS PRA Standard of the SQN SPRA.

Subsequently, the updated SQN SPRA model and documentation were subjected to an independent closure review. This review is described in Section A.4.

A.4 Finding-Level F&O Independent Closure Review

The SQN Seismic Probabilistic Risk Assessment (SPRA) Finding-Level F&O Independent Assessment & Focused-Scope Peer Review was performed at the TVA Corporate offices in Chattanooga, Tennessee, from February 4 - 8, 2019. The purpose was to perform an independent assessment in accordance with Appendix X of NEI 05-04/12-13 to review TVA's proposed close out of Finding-Level F&Os of record from prior PRA peer reviews against the ASME/ANS RA-Sb-2013 PRA Standard [8] and to perform a Focused-Scope Peer Review for an Upgrade to the SPRA.

The process used for the independent technical review is outlined in the Appendix X of NEI 12-13, which has been accepted by NRC. The review focused on the closure of the 54 open F&Os.

The review was based on results of a completed PWROG review of the SQN SPRA (final report issued July 2018). The result of this independent assessment is intended to be used to support future License Amendment Request (LAR) submittals. Finding-Level F&O dispositions reviewed and determined to have been adequately addressed through this independent assessment are considered “closed” and no longer relevant to the current PRA model. Therefore, these F&Os do not need to be carried forward or discussed in future LAR submittals.

The Independent Assessment Team consisted of 6 team members with extensive qualifications and extensive experience in all areas of SPRA. All reviewers met the criteria specified in NEI 05-04 and NEI 12-13 and the ASME/ANS RA-Sa-2013 PRA Standard Section 1-6.2, and in NRC's memoranda outlining expectations for a finding closure independent assessment. Detailed resumes for each of the team members are provided in Appendix D of the peer review report.

A.4.1 Summary of the Finding Level F&O Independent Technical Review Process

Review team criteria (NEI 12-13 and Section 2.2) and Review Schedule (NEI 12-13 Section 2-3) were addressed in recruiting and approving the closure review team members and defining the schedule for the review. Reviewer independence was established, approved, and documented in the closure review report. Reviewer experience meets the criteria specified in the NEI guidance documents and ASME/ANS RA-Sa-2009 PRA Standard Section 1-6.2. Overall review team experience is such that there were two qualified reviewers for each F&O.

TVA provided the PRA model files and PRA notebooks sufficiently in advance of the start of the onsite review to allow the reviewers to prepare and conduct a more efficient technical review. As input to the review, TVA provided a copy of the SQN peer review report, the list of peer review findings to be considered, and their suggested resolution of each finding.

In accordance with the guidance in NEI 12-13, Appendix X, a lead reviewer and supporting reviewer was assigned for each Technical Element. The reviewers reviewed the associated finding-level F&Os and made the initial determination regarding adequacy of resolution of each finding within their scope. A consensus process was followed during which the full team present on the day of the associated consensus session considered and reached consensus on the adequacy of resolution of each F&O using the appropriate SRs of the ASME/ANS PRA Standard for the review criteria. The team performed additional consensus sessions via teleconference to disposition F&Os not fully resolved at the conclusion of the onsite review.

A.4.2 Independent Technical Review Team Qualifications

The members of the Independent Technical Review were Mr. Richard Anoba of Jensen Hughes, Mr. Walter Djordjevic of Stevenson & Associates, Mr. Jeffrey Kimball of RIZZO, Todd Radford of Jensen Hughes, Dr. Glen Rix of Geosyntec Consultants, and Mr. Barry Sloane of Jensen Hughes.

Mr. Anoba is a Senior Consultant with over 40 years of experience in areas of engineering analysis, system reliability analysis, safety analysis, PRA, project management, design engineering, and power plant operation. He is a registered professional engineer in the states of California and North Carolina, and a member of the American Society of Mechanical Engineers (ASME). He has been employed by The Atomic Energy Commission, the Energy Research and Development Administration, the DOE, Westinghouse Hanford Company, Pacific Gas and Electric Company, Martin Marietta Corporation, Brown and Root Incorporated, Bovay Engineers, Carolina Power & Light Company, Science Applications International Corporation, and Data Systems and Solutions. He was the president of Anoba Consulting Services that provided PRA support services including model updates, model reviews against the ASME PRA Standard, and external event model development. At the time of this review Mr. Anoba was the Director of Risk-Informed Engineering in the Power Services Group of Jensen Hughes. He is currently the Chief PRA Engineer for ENERCON.

Mr. Djordjevic is a Senior Consultant with 43 years of experience. Mr. Djordjevic founded the Jensen Hughes (Stevenson & Associates) Boston area office in 1983 and serves as President and General Manager. Mr. Djordjevic is expert in the field of structural engineering, specifically in the areas of structural vulnerabilities to the effects of seismic and other extreme loading phenomena. Mr. Djordjevic has been involved in numerous seismic analysis and design projects for Jensen Hughes, including finite element evaluations, building structural analyses and component tests and analyses. Mr. Djordjevic is heavily involved in post-Fukushima engineering support for nuclear clients throughout the US and internationally. Mr. Djordjevic is an expert in developing seismic fragilities for power plant structures, systems and components.

Mr. Jeffrey Kimball is a Chief Seismologist with RIZZO. Mr. Kimball has 38 years of experience with the evaluation and characterization of natural phenomena hazards and the design of critical facilities to resist these hazards. He led the preparation of DOE standards and guides to define requirements and procedures to complete assessment of natural phenomena hazards. Mr. Kimball has extensive knowledge of a wide range of nuclear facility regulations, regulatory guides, standards, manuals, and review plans associated with nuclear facility design and evaluation. He is also a recognized expert in site characterization; ground motion modeling including site response and PSHA, including guidance for completing PSHA.

Mr. Todd Radford is a Senior Engineer with 12 years of experience including 8 years at Jensen Hughes. In addition to acting as Structural Engineering Manager in the Wakefield, MA office of Jensen Hughes, Dr. Radford has been involved in numerous seismic evaluation projects for Jensen Hughes as both project manager and lead analyst. He is an expert in building modeling and soil-structure interaction and has led response analysis efforts for multiple SPRA projects for nuclear power plants. Dr. Radford has been responsible for post-Fukushima engineering support, including R2.3, ESEP, FLEX, SFP evaluations, and R2.1 HF confirmations. Dr. Radford has also led development efforts for Jensen Hughes internal engineering analysis software including Spectra, SULTAN, and ANCHOR.

Dr. Glenn Rix is a Senior Principal in Kennesaw, Georgia, with expertise in seismic hazard evaluation, geotechnical earthquake engineering, and performance-based and risk-based analyses. Dr. Rix joined Geosyntec in 2013 after a distinguished 24-year career as a faculty member in the School of Civil and Environmental Engineering at the Georgia Institute of Technology specializing in geotechnical and earthquake engineering.

Mr. Barry Sloane is a Senior Consultant with over 37 years of experience serving the commercial nuclear power industry, 33 years of which have been focused in risk assessment, risk management, reliability, and related areas. Mr. Sloane is a manager responsible for the Jensen Hughes Power Services Group risk management co-sourcing services, and for leading various PRA modeling and risk application development and implementation programs. He has been involved in developing and updating PRA standards and self-assessment guidance since 2000 and has experience in internal events and SPRA. Mr. Sloane is currently one of the Power Services Group's leads for development and implementation of 10 CFR 50.69 (Risk-Informed Engineering Programs) and Risk-Managed Technical Specifications.

A.4.3 Independent Technical Review Team Conclusions

Four of the seven SHA findings were assessed to be upgrades during the closure review and were resolved as part of a focused scope review of HLRs SHA-I and SHA-J. This focused scope review was performed during the week of the closure review. Three of the SHA findings, all twenty-eight SFR findings, and twenty-three of the SPR findings were resolved during the onsite review session. The remaining two SPR finding were resolved after the onsite review session as discussed below.

Finding SPR F&O 8-3 was initially partially resolved because of documentation issues (i.e., all technical aspects were resolved). It was later fully resolved based on updated documentation provided on March 22, 2019.

Finding SPR F&O 8-11 was initially partially resolved (i.e., there were still some remaining technical issues). It was later fully resolved based on updated documentation provided on April 1, 2019.

A.5 Summary of SPRA Capability Relative to SPID Tables 6-4 through 6-5

The PWROG performed a peer review of the SPRA in 2018. The SPRA was peer reviewed relative to Capability II for the full set of requirements in the Standard. After completion of the subsequent independent assessment in 2019 which utilized the process given in Appendix X of NEI 12-13, the full set of supporting requirements were met.

The final F&O dispositions are provided in the following pages in this Appendix. There are two tables: Table A-1 provides the dispositions for the original peer review findings within the scope of the F&O independent assessment. Table A-2 provides the dispositions for the new F&Os originated from the focused scope peer review defined in Section 1.2.

Each table is sorted by Review Unit in the first column. The columns in the table provide the following information (numbers denote column number):

1. Review Unit.
2. The SR number against which the peer review Finding was referenced.
3. The original peer review team's assessment of Capability Category for the referenced SR.
4. The Finding Number from the peer review report.
5. The Finding Description from the peer review report.
6. A summary of the Basis and Suggested Resolution for the Finding from the peer review report.
7. TVA's description of the resolution of the Finding.
8. References to appropriate portions of the SQN SPRA Model and documentation to support TVA's resolution.

9. The Independent Assessment Team's assessment of whether TVA's resolution of the Finding represents PRA Maintenance or Upgrade.
10. The Independent Assessment Team's basis for Maintenance or Upgrade determination.
11. The Independent Assessment Team's assessment of adequacy of the Finding resolution.
12. The Independent Assessment Team's assessment of the new Capability Category of the referenced SR given the Finding resolution.

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
	SHA										
1	SHA-J2	Met	3-1	SHA-J2 requires that sufficient documentation be provided for the scientific interpretations that are the basis for the inputs and results. The clarity or completeness of the documentation must be improved in selected areas. Refer to 3-12 and 3-13 for a continuation of 3-1. Originated from SR SHA-J2; Met; Finding.	The following issues in the documentation must be addressed: 1. Bob Hatcher (U. of Tennessee, Knoxville) and co-workers (Hatcher et al., 2013; Warrell, 2013; Cox et al., 2014) document potential paleoseismic and paleoliquefaction features in the Douglas Reservoir area of Tennessee. They interpret these as indicative of large magnitude earthquakes in the ETSZ (part of the PEZ-N and PEZ-W zones of the CEUS-SSC). The Fugro PSHA report cites this post-CEUS-SSC work but does not evaluate this new information and whether it is consistent with the CEUS-SSC model. In discussions with PSHA analysts, an evaluation was performed but was not included in the report. A robust written description of the evaluation of the Hatcher work was provided in response to a peer review team question. 2. The de-clustering approach used in the Fugro PSHA report for the new earthquake catalog (1/1/2009-1/31/2015) consists of removing events within 24 hours and +0.1 degrees (retaining only the largest of these events). In discussions with PSHA analyses, a basis was provided for determining that Is this approach reasonable in comparison to more common approaches (e.g., EPRI-SOG, Reasenberg, Kagan-Knopoff, Grunthal). 3. The PSHA report contains a treatment of induced seismicity (both from surface water reservoirs and from deep injection) and demonstrates that there is no need to modify the CEUS-SSC model or to introduce additional seismic sources to consider induced seismicity. A documentation deficiency was identified because the report does not indicate that government databases were consulted to determine that there are no deep-injection wells in the region around SQN. Based on discussions with PSHA analysts, an evaluation was performed, and appropriate sources	1. The written evaluation of the work was included in the updated PSHA report. 2. The discussion and justification provided during the peer review were incorporated into the updated PSHA report. 3. Text was added to the updated PSHA report documenting the results of the well data search. 4. Report was updated with the correct reference.	SHA F&O 3-1 3-12 3-13 Resolution.pdf Fugro Consultants, Inc Project Report PR No. 150017-PR-01 Revision 2, 11/30/2018, "Probabilistic Seismic Hazard Analysis for TVA Sequoyah Nuclear Plant PSHA Results Report"	Maintenance	Resolution of documentation finding, no new methodology.	RESOLVED: Fugro Consultants Inc. Project Report No. 150017-PR-01, Revision 2 (11/30/2018), Probabilistic Seismic Hazard Analysis for TVA Sequoyah Nuclear Plant, PSHA Results Report was reviewed to confirm that the documentation was modified to address the technical issues raised as part of SHA Finding 3-1. The following items are addressed: 1. Section 3.3 of the PSHA Results Report was modified to provide a written evaluation of the work performed by Hatcher and others related to potential paleoliquefaction. 2. Section 3.2.2.2 of the PSHA Results Report was modified to provide a discussion and justification of the declustering approach used. 3. Section 3.4.5 of the PSHA Results Report was modified to provide a summary of the well data search performed as part of assessing the potential for induced seismicity. 4. Sections 1.2, 2.2, 3.2, 3.2.2.2, 4.1, 5.0, 5.2, 8.2, and Appendix B of the PSHA Results Report were modified to provide the correct reference for the CEUS-SSC Report (EPRI, 2015a).	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
					searched. 4. The report incorrectly references EPRI (2015, aka Richards 2015) as USNRC (2015).						
1	SHA-I2	Not Met	3-6	SHA-I2 requires that the frequency of hazard occurrence and the magnitude of hazard parameters of interest to fragility are assessed for hazards that are not screened out in SHA-I2. Liquefaction and its associated phenomena screens into further evaluation for the site. Report 1401450.401, "Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant," describes three methods of evaluation for potentially liquefiable soils. Two of the three methods are not appropriate in this case. Other methods that are more appropriate and up-to-date were not used. Originated SR SHA-I2; Not Met Finding.	Report 1401450.401, "Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant," describes three methods of evaluation for potentially liquefiable soils. The first of these methods uses Idriss and Boulanger (2008) with shear stresses developed from the site response analysis. The Idriss and Boulanger method was empirically developed using a generalized shear stress reduction coefficient (written as rd) to determine shear stress. It is important that the method used to develop shear stress is consistent with the approach used in the empirical development of the liquefaction method. The Idriss and Boulanger (2008) method was developed using a generic rd (not using results from site response analysis). Thus, method 1 is inappropriate. The need to use the appropriate rd based on that used in relationship development was first raised in EERC report 2003-06 (Seed et al.2003). Page 84 of the 2016 National Academy report on liquefaction summarized the technical issue and continued understanding of the liquefaction technical community when it stated "If the empirical rd relationship associated with the particular liquefaction method is not an unbiased estimator of rd (e.g., for the triggering relationships developed by Idriss and Boulanger [2008] and Boulanger and Idriss [2014]), the empirical rd relationship	The evaluation methodologies used to assess liquefaction effects and related settlement and lateral spreading are described in Section 5.1 of CJC-SQN-C-001. The evaluations were updated to use methodologies consistent with recommendations included in the National Academy study, including the use of consistent rd relationships. These recommendations include recommended probabilistic approaches.	SHA 3-6 3-7 3-8 3-11 FORESPONSES_FINAL2.pdf. CJC-SQN-C-001 Rev 0 "Updated Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant (SQN)"	Upgrade	New methods and approach were used to address Finding.	RESOLVED: Carl J. Costantino and Associates Report CJC-SQN-C-001, Revision 0 (January 16, 2019), Updated Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant (SQN) provide a new approach and method for evaluating soil failure modes caused by liquefaction and cyclic softening. The new analysis includes selecting up-to-date methods for assessing liquefaction associated vertical settlement, lateral spreading, and slope stability, and a probabilistic assessment for a range of input variables including soil properties, depth to bedrock, site response amplification factors, and earthquake deaggregation. The revised analysis was performed for five loading levels at annual frequencies of exceedance from 1x10-4 to 1x10-7. The CJC and Associates report included an assessment to evaluate whether the use of equivalent linear methods was justified. A focused technical review was performed and is reported as part of evaluation of the CJC and Associates report relative to determining whether Supporting Requirement SHA-I2 had been adequately addressed.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
					associated with the particular liquefaction method should be used to compute induced shear stresses at depth, and the results of site response analyses should be used only to refine the estimate of the PGA at the ground surface.” The third of the methods used is described as Youd and Idriss (2001, typically known as Youd et al 2001). Generally, the Youd et al. (2001) method is out of date, particularly for depths beyond approximately 15m. A range of other methods more consistent with current practice in the liquefaction engineering community are available. The National Academy study was initiated to evaluate the range of modern methods, including probabilistic methods that allow for more direct assessment of parameters needed for SPRA. Thus, two of the three methods used are not appropriate for this study, particularly given the availability of a number of more modern approaches. We note also that the three liquefaction models give very different results for some locations (e.g., DGB; see Figures G16-G20 of the Soil Failure report.)						

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
1	SHA-I2	Not Met	3-7	The liquefaction evaluation does not extend the evaluation to the full extent of the potentially liquefiable layers under the ADGB and DGB, instead artificially limiting the evaluation to 60 feet. The evaluation does not provide an assessment of the magnitude of the phenomena (i.e., the value of the parameter of interest for input to fragility calculation) for ground motion levels of interest to the SPRA. Originated from SR SHAI2; Not Met; Finding.	SHA-I2 requires that the frequency of hazard occurrence and the magnitude of hazard parameters of interest to fragility are assessed for hazards that are not screened out in SHA-I2. Liquefaction and its associated phenomena screens into further evaluation for the site. Report 1401450.401, “Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant,” provides evaluations of liquefaction and its consequences. The report appears to state that evaluations of liquefaction and its consequences are limited to evaluation of the first 60 feet as a result of stated limitations in the stress reduction factor. However, this may not cover the entire depth of soil that may contribute to seismic risk to the plant. Additionally, the report is not clear on how the modifications to the SPT blowcounts were applied. It is important that the blowcounts are corrected using the soil overburden in place at the time of the SPT testing. In several places in the report assumptions are implicit or stated that are not fully justified. Examples are the assumption that free-field evaluations of settlement are directly applicable to the settlement of the associated SSCs.	The effect of the use of equivalent linear methods in lieu of non-linear methods to compute site response is evaluated as described in the response to F&O 3-11. As discussed in the response to F&O 3-6, more modern liquefaction evaluation methodologies that use probabilistic approaches are incorporated into the CJC-SQN-C-001 report. These updated methodologies accommodate profiles that extend to depths larger than 60 ft. Section 3.4 of the report describe the depths to rock for the areas evaluated. As indicated in this section, depth to rock in the DGB/ADGB area is 65 ft (β=0.31), 21 ft (β=0.234) in the ERCW area, and 22 ft (β=0.35) in the yard areas. The approach used to modify the boring data to reflect current in-situ conditions is described in Section 3.3 of the CJC-SQN-C-001 report.	SHA 3-6 3-7 3-8 3-11 FORESPONSES_FINAL2.pdf. CJC-SQN-C-001 Rev 0 "Updated Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant (SQN)".	Upgrade	New methods and approach were used to address Finding.	RESOLVED: Carl J. Costantino and Associates Report CJC-SQN-C-001, Revision 0 (January 16, 2019), Updated Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant (SQN) provide a new approach and method for evaluating soil failure modes caused by liquefaction and cyclic softening. The new analysis includes selecting up-to-date methods for assessing liquefaction associated vertical settlement, lateral spreading, and slope stability, and a probabilistic assessment for a range of input variables including soil properties, depth to bedrock, site response amplification factors, and earthquake deaggregation. The revised analysis was performed for five loading levels at annual frequencies of exceedance from 1x10-4 to 1x10-7. The CJC and Associates report included an assessment to evaluate whether the use of equivalent linear methods was justified. A focused technical review was performed and is reported as part of evaluation of the CJC and Associates report relative to determining whether Supporting Requirement SHA-I2 had been adequately addressed.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
1	SHA-I2	Not Met	3-8	SHA-I2 requires that the frequency of hazard occurrence and the magnitude of hazard parameters of interest to fragility are assessed for hazards that are not screened out in SHA-I2. Slope stability screens into further evaluations for the site. Report No. 1401450.401 uses the sliding-block method of Bray and Travasarou (2007) to estimate liquefaction-induced lateral spreading. There are several details on how the procedure is applied in this document that required correction. Originated from SR SHAI2; Not Met; Finding.	The issues in the procedure applied include the following: a. The ground-motion input in the Bray and Travasarou (2007) paper is applied at the bottom of the sliding block. When applying the method to lateral spreading, this corresponds to the ground motion at the elevation of the liquefiable layer. It appears that, in this report, the ground-motion at the ground surface is used as input. b. In this report, the yield coefficient ky is calculated as the PGA associated with the onset of liquefaction. In other publications that document the state of practice (e.g., NAS 2016 State of Practice Review; FHWA-NHI-11-032 GEC No. 3), ky is calculated as the acceleration required to overcome the residual undrained strength of the liquefied layer. c. Figure G-7 in the report is used to make the argument that the rigid-block assumption is more conservative than the flexible- block assumption for the range of sliding-block periods of interest. This argument is not straightforward due to the complexity of equations G.2 and G.3 (page G3). In particular, the argument does not consider that Sa(1.5Ts) may be greater than PGA, depending on the spectral shape and on the period of the sliding mass. To complicate matters, there are also Sa(1.5Ts)**2 and PGA**2 terms, as well as ln(ky)* Sa(1.5Ts) and ln(ky)*PGA terms, that make the comparison even more difficult. Thus, this argument should not be made or should be strengthened. Another way to look at this concern is to subtract Eq. G.3 from Eq. G.2. The result will consist of the quantities shown in Figure G7, plus terms that depend linearly and quadratically on the quantity Sa(1.5Ts)/PGA (with additional dependence on ky). In other words, the conservatism (if any) of the rigid-block approximation depends on the ratio Sa(1.5Ts)/PGA and on the value of ky. Also, too much conservatism in the rigid-block	Data related to deaggregation of the seismic hazard provided by Fugro, provided the data needed to take advantage of the methodology developed by Rathje and colleagues (Rathje et al. (2004)) which uses Magnitude plus two ground motion parameters (PGA, PGV) to better constrain the data and provide more accurate predictions of lateral deformations. The Rathje model also is provided in terms of a probabilistic approach, which permits the implementation of the peer review teams recommendation in F&O 3-6 to use probabilistic approaches. Thus, the approach described by Bray and Travasarou (2007) is not used in the updated evaluations. Report CJC-SQN-C-001, Appendix F describes the implementation of the probabilistic methodology developed by Rathje and colleagues (Rathje et al. (2004)). This methodology incorporates; defining the input ground-motion at the liquefiable layer elevation, specifying the yield coefficient in terms of residual undrained strength, and incorporating the period of the sliding soil mass. It should be noted that the corrections to the Bray and Travasarou model identified in the F&O are also applicable to the Rathje model and its implementation incorporates the corrections identified in the F&O.	SHA 3-6 3-7 3-8 3-11 FORESPONSES_FINAL2.pdf. CJC-SQN-C-001 Rev 0 "Updated Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant (SQN)".	Upgrade	New methods and approach were used to address Finding.	RESOLVED: Carl J. Costantino and Associates Report CJC-SQN-C-001, Revision 0 (January 16, 2019), Updated Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant (SQN) provide a new approach and method for evaluating soil failure modes caused by liquefaction and cyclic softening. The new analysis includes selecting up-to-date methods for assessing liquefaction associated vertical settlement, lateral spreading, and slope stability, and a probabilistic assessment for a range of input variables including soil properties, depth to bedrock, site response amplification factors, and earthquake deaggregation. The revised analysis was performed for five loading levels at annual frequencies of exceedance from 1x10-4 to 1x10-7. The CJC and Associates report included an assessment to evaluate whether the use of equivalent linear methods was justified. A focused technical review was performed and is reported as part of evaluation of the CJC and Associates report relative to determining whether Supporting Requirement SHA-I2 had been adequately addressed.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
					assumption would not be appropriate, as lateral spreading does not screen out for locations such as the DGB. The liquefaction-induced soil deformations affect SEL items such as the ERCW piping connected to the DGB, which has been identified as an item requiring an appropriate fragility evaluation because it impacts the operability of the diesel generators.						

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
1	SHA-I2	Not Met	3-11	A screening level analysis of other soil-related hazards including the potential for triggering liquefaction, settlement, lateral spread, bearing capacity and structural sliding, slope stability, and upstream dam failure was provided and these hazards screened into further evaluation. Generally, evaluations of secondary phenomena provided did not extend to large enough ground motion levels to ensure that the risk associated with site phenomena was appropriately captured in the SPRA. The range of amplitudes of the parameters needed for fragility calculations was not provided. Originated from SR SHA-I2: Not Met; Finding.	Report 1401450.401, "Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant," provides evaluations of site response using the LHS method for input to the fragility evaluations. This report described ground motion levels of GMRS to 2xGMRS. It was based on information provided from Fugro (2016). However, evaluations of site response should be performed for a range of ground motion of interest to SPRA. In this case, the evaluations only to ground motions levels associated with 2xGMRS may be insufficient for fully characterizing the potential impact of secondary hazards in relation to the overall risk of the plant. Based on the current SPRA risk results reported, the 0.7 g PGA level at 2XGMRS captures only 26% of the SCDF risk and captures only 13% of the SLERF risk. Based on these reported risk results, the relevant ground motions for assessing risk in this SPRA appear to exceed 0.9g. It is further noted, that the results available for the 2xGMRS levels indicate that linear-equivalent methods are not appropriate for larger ground motions.	Revised liquefaction evaluation methodologies were incorporated in response to F&O 3-6. These methodologies are stress based and use ground motion levels defined in terms of surface PGA. This permits the direct use of Site Amplification Functions (SAFs) from the PSHA site response analyses, which incorporate strain compatible ground motion levels, directly into the liquefaction prediction models for triggering, settlement, and lateral spreading. The updated Soil Failure and Fragility Analysis report (CJC-SQN-C-001) considers settlement and lateral spreading from liquefaction caused by ground motion levels up to those associated with AFE 1x10-07 (2.68g). (Note that these ground motion levels and associated SAFs are taken directly from the PSHA and are not the result of site response analyses performed independent of the PSHA process). The ERCW Access Dike and Pumping Station Access Cells were evaluated using ground motion levels up to 3xGMRS (1.253g). It is noted that SFR-B1 requires only that the analysis must extend up to ground motion levels corresponding to the failure level of the components of interest. The minimum fragility (Am) that fails the SQN ERCW is related to the relays in group SEIS_0-30-6. The Am for this group is 1.14. Thus, the evaluations for Soil Failure and Fragility are sufficient. A sensitivity study was performed and documented in Section 5.6 and Appendix N of CJC-SQN-C-001. This study evaluated the difference in predicted surface responses for ground motion levels up to 3xGMRS and concludes that the use of equivalent-linear models leads to a conservative bias in	SHA 3-6 3-7 3-8 3-11 FORESPONSES_FINAL2.pdf. CJC-SQN-C-001 Rev 0 "Updated Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant (SQN)".	Upgrade	New methods and approach were used to address Finding.	RESOLVED: Carl J. Costantino and Associates Report CJC-SQN-C-001, Revision 0 (January 16, 2019), Updated Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant (SQN) provide a new approach and method for evaluating soil failure modes caused by liquefaction and cyclic softening. The new analysis includes selecting up-to-date methods for assessing liquefaction associated vertical settlement, lateral spreading, and slope stability, and a probabilistic assessment for a range of input variables including soil properties, depth to bedrock, site response amplification factors, and earthquake deaggregation. The revised analysis was performed for five loading levels at annual frequencies of exceedance from 1x10-4 to 1x10-7. The CJC and Associates report included an assessment to evaluate whether the use of equivalent linear methods was justified. A focused technical review was performed and is reported as part of evaluation of the CJC and Associates report relative to determining whether Supporting Requirement SHA-I2 had been adequately addressed.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
						the demands used in liquefaction evaluations (PGA), even at the higher strain levels associated with the 3xGMRS (1.253g) input motions.					
1	SHA-J2	Met	3-12	Continuation of 3-1 Originated from SR SHA-J2; Met; Finding.	5. The profiles in Chapter 6 of the Fugro PSHA report have a high-velocity layer at a depth of 1,500 ft, which corresponds to an edge of the Pond Springs formation. There is uncertainty as to the depth and thickness of this high-velocity layer, but this epistemic uncertainty is not considered (i.e., nearly all simulated profiles have this layer at the same depth and with the same thickness). Additional documentation is required that justifies why it is not necessary to vary these properties (i.e., arguments related to the small relative changes in the thickness of the Conesagua formation and arguments related to the frequencies that would be affected). The rock strata beneath the site have dip angles of approximately 30 degrees and there are some strong velocity contrasts between some of	5. Report was revised to include documentation on why randomization of the depth and thickness of the 1500-ft deep high velocity layer and why the horizontal layer approximation is appropriate in this case. 6. The basis provided is included in the updated report.	SHA F&O 3-1 3-12 3-13 Resolution.pdf Fugro Consultants, Inc Project Report PR No. 150017-PR-01 Revision 2, 11/30/2018, "Probabilistic Seismic Hazard Analysis for TVA Sequoyah Nuclear Plant PSHA Results Report".	Maintenance	Resolution of documentation finding, no new methodology.	RESOLVED: Fugro Consultants Inc. Project Report No. 150017-PR-01, Revision 2 (11/30/2018), Probabilistic Seismic Hazard Analysis for TVA Sequoyah Nuclear Plant, PSHA Results Report was reviewed to confirm that the documentation was modified to address the technical issues raised as part of SHA Finding 3-1. The following items are addressed: 5. Section 6.5 of the PSHA Results Report was modified to include a discussion of the approach to soil profile randomization for the high shear-wave velocity layer and why the horizontal layer approximation is appropriate for the Sequoyah NPP site. 6. Section 2.3 of the PSHA Results Report was revised to include a discussion of the approach used for defining ground motion model aleatory variability (the text included in the report had been reviewed by the original peer reviewers).	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
					these dipping layers. The common rule of thumb is that the horizontal-layer approximation is appropriate when dip angles are less than 20 degrees. Additional documentation is required, in the form of arguments to justify why the horizontal-layer approximation is appropriate in this case, given the depth to the strong velocity interface. 6. Analysis of aleatory residuals from large datasets indicates that the distribution of these residuals has broader upper tails than the commonly used lognormal shape (e.g., Coppersmith et al., 2014; Geopentech, 2015). The effect of these broader tails may be particularly important at the low exceedance probabilities of interest in SPRA. An evaluation of the new models was performed but not included in the report.						
1	SHA-J2	Met	3-13	Continuation of 3-1 Originated from SR SHA -J2; Met; Finding.	7. EPRI (2013, SPID) recommends that the epistemic uncertainty in the degradation properties of soil layers be characterized by using the EPRI and Peninsular Range degradation curves, with equal weights. Examination of Tables 6-5 through 6-8 (last two columns) and Figures 6-8 and 6-9 of the Fugro PSHA report suggest that only the EPRI curves are being used. Upon review of the calculations, it was determined that the Peninsular Range degradation curves were used but were not appropriately included in the report. 8. Section 8.4 of the Fugro PSHA report documents the development of strain-compatible profiles. The figures in this section show the median profile and the logarithmic sigma, but they do not show the actual mean±sigma profiles. The mean±sigma profiles were provided to the peer reviewers in order to allow verification that they are reasonable.	7. The tables in Section 6 related to the FIRS (FIRS3, FIRS5, and FIRS6) that included the Peninsular Range as appropriate per EPRI SPID (2013) and Table 7-1 have been revised accordingly in the updated report. 8. The existing figures in Section 8 are for the purposes of summarizing the results. Plots incorporating mean±sigma strain-compatible profiles would serve no purpose in the current report and would involve 100s of pages without any added value. Those were only needed as part of the calculation packages that are then used to generate the summaries in the Figures in Section 8 and form the basis of the values in the electronic Excel sheets included with the report.	SHA F&O 3-1 3-12 3-13 Resolution.pdf Fugro Consultants, Inc Project Report PR No. 150017-PR-01 Revision 2, 11/30/2018, "Probabilistic Seismic Hazard Analysis for TVA Sequoyah Nuclear Plant PSHA Results Report".	Maintenance	Resolution of documentation finding, no new methodology.	RESOLVED: Fugro Consultants Inc. Project Report No. 150017-PR-01, Revision 2 (11/30/2018), Probabilistic Seismic Hazard Analysis for TVA Sequoyah Nuclear Plant, PSHA Results Report was reviewed to confirm that the documentation was modified to address the technical issues raised as part of SHA Finding 3-1. The following items are addressed: 7. Sections 6.4.1 and Tables 6.5, 6.7, 6.8, and 7.1 of the PSHA Results Report have been revised to explicitly identify the use of the Peninsular Range soil dynamic properties as part of the site response analysis. 8. This item was related to the type and format for a suite of figures included in the PSHA Results Report reporting on strain compatible properties (SCPs). The Finding was requesting that additional figures be added to the report showing the actual mean and plus or minus one standard deviation SCPs; the set of figures in the report show the mean value and associated sigma value (and not the absolute plus or minus on standard deviation value. The electronic appendix to the PSHA Results Report does provide the numerical values for the SCPs. Fugro did not add additional figures to the PSHA Results Report (several dozen figures would need to be added). The response provided by Fugro is reasonable given the data listed in the electronic appendix.	Met
	SFR										

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
2	SFR-C6	CC I/II	3-4	SFR-C6 requires that soil strain levels corresponding to the input ground motions that contribute most to the seismically induced core damage frequency must be used when soil-structure interaction analysis is conducted. The DGB is founded on soil and requires that strain-compatible properties for the appropriate ground motion level are used. Originated from SR SFR-C6; Met: Finding.	Report 1401450.401, "Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant," provides evaluations of site response using the LHS method for input to the fragility evaluations. This report described ground motion levels of GMRS to 2xGMRS. It was based on information provided from Fugro (2016). However, evaluations of site response should be performed for a range of ground motion of interest to SPRA. In this case, the evaluations only to ground motions levels associated with 2xGMRS may be insufficient for fully characterizing the potential impact of site response in relation to the overall risk of the plant. Based on the current SPRA risk results reported, the 0.7 g PGA level at 2XGMRS captures only 26% of the SCDF risk and captures only 13% of the SLERF risk. Based on these reported risk results, the relevant ground motions for assessing risk in this SPRA appear to exceed 0.9g.	A supplemental evaluation of the effects on seismic response of the DGB to degrading soil properties at ground motions above the GMRS is performed to characterize the effects of higher hazards on the DGB response. Because F&O 3-4 challenges the appropriateness of equivalent-linear methods at these higher hazard levels, the supplemental evaluation considers nonlinear hysteretic soil behavior. The approach used in this supplemental evaluation and the results and conclusions obtained therefrom are described in "Utility Responses to SFR F&Os". The supplemental evaluation shows that the use of ISRS generated based on SSI with equivalent-linear soil properties compatible with GMRS ground motion is conservative for the DGB equipment frequency range of interest with respect to soil response at larger ground motions, including nonlinear hysteretic soil behavior.	SC Solutions, Inc., "Sequoyah Nuclear Plant Seismic Probabilistic Risk assessment: Structural Response Analysis", SQN-17-001 Rev 2.	Maintenance	Study with minimal effect on risk results.	RESOLVED: The nonlinear effect is significant (factor of 2 for 3x motion above 10 Hz due to strain softening of soil. This effect is addressed in the resolution to F&O's 5-1, as well as parts of some other F&Os. SUGGESTION: The response states "If any SSCs in the DGB are both (a) significant risk contributors, and (b) are not expected to exhibit failure until ground motions significantly larger than the GMRS, then plant seismic risk insights could be affected by the current choice of hazard range of interest for SSI analysis and corresponding strain-compatible soil properties." This is not addressed in this F&O response but is addressed in response to F&O 5-1. Suggest adding a cross-reference in the documentation.	CC I/II
2	SFR-G2	Met	3-9	Report SQN-17-001 Rev. 1 documents the structural-response calculations, including the LHS randomization of the strain-compatible velocity profiles. There is a documentation deficiency in this report, in that this report does not indicate that damping ratio is taken as having a perfect negative correlation with Vs (as suggested by the physics of damping in soil and rock). Originated from SR SFR-G2; Met Finding.	Sufficient documentation be provided for the scientific interpretations that are the basis for the inputs and results. The missing documentation on the profile randomization for SSI is important for the documentation of this basis.	The second paragraph of Section 7.5.1 of the Front Body of the Report SQN-17-001 is revised to clarify the pairing correlations, including perfect negative correlation between the soil damping and shear-wave velocity.	Final signed Response to FO3-9 (Rev0).pdf SC Solutions, Inc., "Sequoyah Nuclear Plant Seismic Probabilistic Risk assessment: Structural Response Analysis", SQN-17-001 Rev 2.	Maintenance	Documentation update	RESOLVED: Verified that damping and shear-wave velocities are considered as being treated as negatively correlated as documented in Revision 2 of Section 7.5.3 of Report SQN-17-001.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
2	SFR-F1	CC I/II	3-10	Appendix AF of report TVAESQN010-REPT-001 – PART B uses the vertical liquefaction-induced settlements from report No. 1401450.401 Revision 1 to calculate the fragility for the ERCW piping. The vertical settlements for the GMRS and for 2*GMRS motions have betas of 1.28 and 1.01, respectively. These high betas reflect the high uncertainty in nonlinear soil behavior. The calculated fragility for the ERCW piping in the former report has a composite beta of 0.35, which is obtained by using generic betas obtained from the SPID. Considering that pipe stresses are approximately proportional to settlement, the peer review team consensus was that we would expect composite betas of the order of 1 for this component. The applicability of this lower generic beta in this situation must be justified. Otherwise, the beta should be revised to use a more appropriate value consistent with the high uncertainty of the underlying soil processes. Appendix AF also ignores the failure associated with horizontal displacement (although the horizontal liquefaction-induced displacements are larger than the vertical displacements), using the argument that both the building and the pipe move horizontally with the soil mass. Differential horizontal displacements of the ERCW pipe in the vicinity of the DGB and ADGB must be considered, or a stronger argument for ignoring them must be provided. This calculation can consider that report No. 1401450.401 Revision 1 specifies that these differential displacements should be considered to occur over a distance of 28 feet.	SFR-F1 requires that the median capacity and beta be calculated using plant-specific data or, if necessary, on earthquake experience data, fragility-test data, and generic qualification test data. In this particular case, the generic data are not necessary and do not appear to be appropriate. The SPID generic Beta values were never intended to be used for failure modes associated with high variability phenomena such as soil failure modes.	An updated fragility analysis for the buried ERCW piping is described in Sections 5.5, 6.10, and Appendix H of the updated Soil Failure and Fragility Analysis for SQN [CJC-SQN-C-001]. Both vertical and horizontal soil movements are investigated, and site-specific uncertainty parameters are rigorously computed and accounted for in the updated fragility analysis. Median strain levels in the 14-inch and 36-inch pipe remain below 5% strain for both the corroded and uncorroded pipe conditions, at ground motion levels up to 2.6g corresponding to AFE of 1x10−7. The results in the fragility analysis conclude that for the localized condition the highest strain levels occur in the 6-inch diameter ERCW piping located in the DGB/ADGB Area. Median strain levels exceed 5% at FIRS1 PGA levels greater than 2.5g for the uncorroded condition and at ~ 1.6g for the corroded pipe condition. Using log-log interpolation between the values that were computed, with 5% strain as the limit for median, the resulting AM is 1.73g with a composite beta of 1.27. This fragility value was used in the final quantification.	FO 3-10 2019 final signed out.pdf CJC-SQN-C-001 Rev 0 "Updated Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant (SQN)".	Maintenance	Source is changed to document created as part of SHA upgrade, but fragility use and application is unchanged	RESOLVED: Review of Appendix H.4 of Carl J. Costantino and Associates, Report No. CJC-SQN-C-001, "Updated Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant (SQN)," Revision 0, 2019 confirms assessment for corroded pipe (ERCW - 6") having a 1.73g median capacity associated with a (median) 5% pipe material strain due to localized deformation. In lieu of generic variability a plant-specific composite Beta of 1.27 is calculated. Other soil failure modes due to lateral spreading, combined vertical and horizontal displacements and distributed settlement did not reach 5% piping strain levels thus did not govern. SUGGESTION: Not completely clear from the write-up which part of the evaluation is vertical and which part is horizontal response, that points back to the F&O. Suggest a brief discussion to better tie-in the evaluation in Appendix H with the requirements of the F&O.	CC I/II

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
				Originated from SR SFR-F1; Met; Finding.							
2	SFR-C1	CC I/II	5-1	Seismic response analysis was performed using the GMRS as the reference spectrum input for structures housing the SSCs. A sensitivity study was documented in the Fragility Notebook addressing 'high capacity risk significant components' affecting by the concrete cracking in structures which were not cracked at the GMRS level. A finding has been written on the selection of the reference earthquake that was selected for the fragility calculation of SSCs. The SCDF associated with the GMRS 0.35 g level is relatively low. As such, the peer review team concludes that a realistic assessment of what reference earthquake is driving the risks should be conducted by the fragility team. In the assessment of the impact of a potentially higher reference earthquake, the sensitivity study should consider the effects of all non-linearities that could be affected by larger reference earthquakes resulting from the risk information from the final quantification for the SPRA. Originated from SR SFR-C1: Met; Finding.	The Sequoyah fragility notebook contains the following discussion of reference earthquake on page 47: 'Since the selection of the reference earthquake can affect the realism in the seismic response due to non-linearities in the structures and soil/rock properties. An appropriate ground motion level for reference earthquake is considered as earthquake level where most of the risk is originating. The approach to determine the high capacity top-risk contributors and the dominant seismic bin contributing for majority of risk was discussed with industry experts like Dr. Robert P. Kennedy and others. Two criteria that are currently adopted in the industry were discussed. In the first approach the maximum change in slope of Conditional Core Damage Probability (CCDP) and Conditional Large Early Release Probability (CLERP) across the seismic bins is used to identify the dominant seismic bin. In the second approach, the dominant seismic bin is considered as the seismic bin where a cumulative CCDP or CLERP reaches a relative fifty percent to overall CCDP or CLERP. For Sequoyah SPRA the former approach was recommended by Dr. Kennedy.' The peer review team agrees with the reference earthquake discussion in this part of the Fragility Notebook; however, the Sequoyah SPRA team did not follow these stated criteria for the appropriate reference earthquake. The approach	Resolution of F&O 5-1 is accomplished by identifying all potential sources of non-linearities in the response of SQN SSCs for earthquake levels higher than the "Reference Earthquake" GMRS, followed by evaluation of the potential significance of each of these non-linearities on the SQN SPRA risk insights. An exhaustive search for potential sources of beyond-GMRS non-linearities in the SQN SPRA resulted in the following list of items: • Building response (this was identified as the predominate source of non-linearity and evaluated prior to the peer review) • Site Response Analysis • Soil failure / displacements • Building impact • Component impact with other plant features • Shake-space crossings • NSSS support conditions • Spectral shape • V/H Ratio • HRA The potential significance of each of these non-linearities are evaluated below. In all cases, the non-linearities are resolved by one of the following methods: • The fragility is revised to include the effect of the non-linearity • Demonstration that the potential non-linearity was already captured in the fragility		Maintenance	Source is changed to document created as part of SHA upgrade, but fragility use and application is unchanged	RESOLVED: This assessment is wide-ranging and thorough. All reference documents were reviewed including the soil failure analyses and the Fragility and Quantification notebooks. The peer review responses address various soil failure modes, shake-space maintenance and closure, relay fragility including effects of building pounding, SSC capacity increases due to soil and/or structural nonlinearities (softening), turbine building collapse, and sensitivity assessments due to different combinations of SSC capacity increase by building (and system). These assessments are accomplished by explicit and thoughtful analysis to avoid ambiguity and debate. The resolution effort for this work is commendable.	CC I/II

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
					recommended by Dr. Kennedy (above) for assessing the reference earthquake is one several approaches that are currently considered as technically appropriate and the peer review team agrees that this would be an acceptable approach for Sequoyah. Other approaches such as the point where 50% of the SCDF and SLERF risk are accounted for in the quantification could also be considered as a sensitivity study on the reference spectrum level itself.	analysis • Demonstration that the potential non-linearity has no significant effect on seismic response and fragility of the SSC of interest • Demonstration that the non-linearity results in a conservative bias, which is shown to be acceptable.					
2	SFR-C1	CC I/II	5-2	Buried piping was reviewed as part of the SPRA. The only buried piping that affects the SPRA was reported to be the ERCW piping. The ERCW buried piping fragility was calculated for the portion of the line where it enters the Diesel Generator building. The project team argued that this section represented a bounding case. Adequate justification was not provided to the peer review team to verify that this location at the interface to the Diesel Generator Building was the bounding case In addition the ground motion levels addressed in the assessment of the ERCW fragility are not adequately justified to address all of the acceleration bins that form the basis for the seismic risk. Originated from SR SFR-C1: Met; Finding.	The ERCW fragility was developed based on a review of ground motions at the GMRS level and up to twice the GMRS level. In order to define a realistic fragility for the ERCW piping, the seismic response should reflect consideration of response at the reference earthquake level. The fragility for this piping is highly dependent on the soil response, which is nonlinear at higher ground motions.	As described in the resolution of F&O 5-1, non-linearities for both site-response analysis and soil failure analysis were investigated in detail in the updated Soil Failure and Fragility Analysis for SQN [Report No. CJC-SQN-C-001]. Also, as described in Report No. CJC-SQN-C-001, all potential site vulnerabilities to soil failure were identified, described, and assessed in detail. Furthermore, as described in the resolution of F&O 3-10, the Report No. CJC-SQN-C-001 soil failure and fragility analysis determined that the governing condition was for the 6-inch diameter ERCW piping located in the DGB/ADGB Area. The resulting AM of 1.73g with a composite beta of 1.27 was assigned to the buried ERCW line and these fragility values were used in the final quantification.	FO 5-2 2019 signed.pdf. CJC-SQN-C-001 Rev 0 ""Updated Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant (SQN)".	Maintenance	Study with no effect on results.	RESOLVED: Review of Appendix H.4 of Carl J. Costantino and Associates, Report No. CJC-SQN-C-001, "Updated Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant (SQN)," Revision 0, 2019 confirms assessment for corroded pipe (ERCW - 6") having a 1.73g median capacity associated with a (median) 5% pipe material strain due to localized deformation. Other soil failure modes due to lateral spreading, combined vertical and horizontal displacements and distributed settlement did not reach 5% piping strain levels thus did not govern. A comprehensive review of soil failure modes is included in Section 6 of CJC-SQN-C-001. This identifies the ECRW buried piping as the governing case for fragility determination. Localized stresses are developed for multiple sizes of pipe in Appendix H, and the 6" pipe is shown to govern. 6" pipe is representative of the pipe in the DGB/ADGB area which originally was identified as controlling.	CC I/II

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
2	SFR-C4	CC I/II	5-3	Several NSSS components are dominant risk SSCs for the Sequoyah SPRA. The fragilities for these NSSS components are directly affected by the seismic response developed for the SPRA. Discussions with the seismic response team members and review of the seismic response documentation resulted in the peer review team conclusion that the potential effects of non-linear dynamic response were not adequately considered for the higher ground motions that contribute to the seismic risk. The peer review team recommends that these non-linear effects be justified not to affect the seismic response resulting from the earthquake levels influencing the seismic risk. Originated from SR SFR-C4: Met; Finding.	NSSS systems typically have gaps at various support locations, contact and sliding interfaces, one-way tension elements like tie rods, etc., that result in nonlinear response. Some of those non-linearities are addressed as part of the four NSSS model variations that were used for the seismic response analyses. However, it was not clear to the peer review team that these four models represented the appropriate structural dynamics that would exist at higher ground motions than the ground motions used when these models were developed. The peer review team is aware of information from NSSS vendors that would suggest that some of the structural dynamics associated with higher input motions can appreciably affect the effective linear simplified models used in the design analysis.	F&O 5-4 investigates NSSS component support condition non-linearities within the Reactor Coolant Loop (RCL). It was found that the individual and combinations of support conditions are not dependent on the magnitude of the ground motion above design basis. Therefore, the NSSS model that was used in the SQN fragility evaluation is appropriate for the range of response applicable to the SQN SPRA. Additionally, the effect of different support nonlinearities on seismic demands used for NSSS fragility analysis was found to be negligible.	Final Signed Response to FO5-3.pdf.	Maintenance	Study with no effect on results.	RESOLVED: The attachment to F&O 5-3 clearly demonstrates that the geometric nonlinearities considered for design are unaffected by seismic magnitude thus being appropriate and linearly scalable for use in determining seismic fragilities.	CC I/II
2	SFR-D2	Met	5-4	Masonry block walls are typically a key element of the seismic interaction reviews conducted in SPRAs. Block wall assessments were conducted for the Sequoyah SPRA. Appendix AJ in Part B and Appendix AP in Part C of the Fragility Report (TVAESQN010-REPT-001) present Block Wall calculations. Many of the block walls are top supported by cinch anchors and angles as a means of providing lateral support. The fragility evaluation of Block Walls focuses on the flexural capacity and inelastic drift limit but does not include the consideration of the failure of these cinch anchors. This failure mode should be specifically assessed for top supported block walls relying on the lead anchors for anchorage. Originated from SR SFR-D2: Met; Finding.	Lead cinch anchors are known to have lower capacity than steel expansion anchors and, as such, they represent failure modes that are typically evaluated to assess whether they are controlling. The block wall top support is provided by a steel L3x3 angle on each side of the wall, anchored to the ceiling by 3/8" lead cinch anchors spaced at 24 in on center for many of the block walls. Discussions with the TVA team during the peer review week on the reason for not addressing this lead anchor failure mode in their fragility evaluations resulted in the response that they had checked this with back of the envelope calculations and the fragility was much higher than the governing failure modes defined for the block walls. A cursory review of the back of the envelope calculation did not convince the peer review team that the case that the lead anchor bolts would be much higher than the currently stipulated fragility levels documented for the block walls. Verification of this lead anchor failure mode is recommended to be conducted.	To resolve this F&O a systematic and extended condition review of block wall fragility evaluation and components fragility evaluation that are affected by block wall interaction is performed. Based on review of the fragility documentation in Appendix AJ, the following issues are identified: 1. Top restraint failure. 2. Cinch anchor capacity. 3. Block Wall Grouping. 4. Boundary Conditions. The revised block wall fragility evaluation is documented in Appendix BE of TVASQN010-REPT-001-PART C. Component Fragilities Affected by Revised Block Wall Fragilities An extended condition review is performed to identify the component fragilities that are affected due to revised block wall fragilities. The components that were identified to have block wall interaction concern are assigned a governing fragility value lesser of anchorage,	FO 5-4 with attachment (2).pdf. Document on TVA Secure Workspace. Appendix BE of TVASQN010-REPT-001-Part C. Document on TVA Secure Workspace.	Maintenance	Update to established methodologies with minimal impact on results	RESOLVED: General response is comprehensive and evaluates block wall fragilities beyond the scope of this F&O. The revised block wall fragility evaluation is documented in Appendix BE of TVAESQN010REPT-001. Checked calculations and "design" capacities used for Cinch anchors in Appendix BE which are based on WSRC research (testing) of lead Cinch anchors. The load allowables used follow the test data and the approach to develop mean anchor capacities and factors of safety is detailed and deemed reasonable.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
					In addition, the back of the envelope calculation incorporated cinch anchor capacity values based on a TVA Design Standard – General Anchorage to Concrete [DS-C1.7.1] and the Allowable Loads for Evaluation of Lead Caulking Anchors [RIMS B41 88 0107 004]). The basis for the TVA shear capacities were the Star Anchor catalog data that suggests the nominal values from tests were 2 Kips, and a factor of four should be used for design or 0.5 kips. It is not clear why the 'back of the envelope' review used 3.3 kips (and then bumped up to 4 Kips) as the best estimate capacity for these lead anchors. Justification for this assumption should be conducted. As a check of the capacities in the Star Manual and whether the 2 Kip average test values reported could actually be conservative, the peer review team reviewed some Savanah River lead anchor criteria (based on tests performed at Savanah River). The allowable load values were actually lower than those reported in the Star catalog (shear capacity for 3/8inch anchors were actually 400 pounds as opposed to the 500 pounds for TVA.	functional and block wall failure fragilities.					

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
2	SFR-A2	CC II	5-5	Several NSSS components are among the dominant risk contributor list both for SCDF and SLERF. The fuel assemblies have been one of the NSSS components that are affected by seismic loading. A review of the fragility showed that this fuel fragility was based on a representative fragility. The basis of the fragility was not justified. Originated from SR SFR-A2: Met; Finding.	The Fuel Assembly Fragility basis could not be determined from a review of the Part A Representative Fragility Report. The final fragility table states that the basis for the Fuel Fragility is this Q1 assessment. The fragility of the RPV is exactly the same as the fragility of the fuel assemblies but there is no evidence of how the fragility was calculated and whether this was somehow assumed to be the same fragility for the fuel. The SPRA team members could not provide the documented basis for this fragility. It is recommended that a fragility be conducted for the fuel, as has been conducted for the other NSSS components.	Based on review of the Fuel Assembly configuration documented in FSAR (Ref. Figure 4.2.1-1, FSAR), there are 193 fuel assemblies arranged on a circular fuel assembly lower grid plate. The unrestrained length of each fuel assembly is about 142” (Ref. L86180503001). The maximum number of fuel assemblies in a row is fifteen. Each fuel assembly measures 8.426” x 8.426”. The largest interior dimension across the core barrel is 127.137” (Drawing 108D480, Sheet 8 of14). Under this constrained configuration, the maximum possible lateral displacement that the fuel assemblies can undergo is 0.747” (=127.137” – 15x8.426”). Also, there are intermediate grid straps placed at regular intervals that are constrained by baffle plates. This reduces the effective flexural length of the fuel rods to roughly 24”, which is essentially rigid. The loss of function design allowable permanent deflection limit of a single fuel assembly is 1.42” (Ref. Westinghouse Letter N4995), which is greater than 0.747”. Therefore, it is judged that the fuel assemblies are rugged and thus have higher seismic capacity than the Reactor Pressure Vessel (RPV). The following fragility parameters, which is same as the other rugged components, in the PRA model are provided to the PRA systems analyst for incorporation into post-peer review quantification. A_m = 5.0g, β_R = 0.24, β_U = 0.32, β_C = 0.40 Section 6.8 is added in Appendix AE, TVAESQN010-REPT-001 – PART B, that documents the fuel assembly fragility.	FO 5-5 Rev 1.pdf . Appendix AE TVAESQN010-REPT-001 - Part B.	Maintenance	Simple change in fragility determination and mapping	RESOLVED: Updated F&O response provides reasonable case that fuel rods are rugged and assigns a rugged fragility. Section 6.8 is added in Appendix AE, TVAESQN010-REPT-001 – PART B, that documents the fuel assembly fragility.	CC II

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
2	SFR-C5	CC I/II	5-6	The Peer Review team spot checked several of the clipped ISRS. Two issues were noted as a result of those spot checks: 1. Clipping conducted on an ISRS with double peaks that are significantly separated were shown to result in a clipped level below the valley between the peaks. This is not appropriate and a misapplication of the clipping methodology. 2. Clipping was not applied to a peak where clipping could have been applied In the face to face meetings with the fragility team, arguments were made as to why these spot checked issues would not result in changes to the fragilities, but since these were just a very small sample of the clipping performed, the peer review team consensus was that a reassessment of the clipping related to these issues was warranted to address the wider use of clipped spectra. Originated from SR SFR-C5: Met; Finding.	Clipping is not always performed accurately for ISRS with double peaks (e.g., Figure 13 – page 1728 of Fragility Report Part B APPENDIX AK, Figure 24- page 1745 of Fragility Report Part B APPENDIX AK). Both ISRSs appear to the peer review team to be sufficiently narrow banded for some clipping. Figure 17 – page 524 of Fragility Report Part C Appendix AU shows the clipped ISRS accelerations are below the valley between the peaks.	To address this F&O, the ISRS clipping criteria used in the project is re-evaluated. The clipping is performed following guidance from EPRI NP-6041, EPRI TR-103959, EPRI 2013 SPRA training course and EPRI 2015 SPRA workshop. The clipping for narrow-banded single peaks is relatively straight forward. Based on the fragility analyst judgement, the ISRS with coupled modes are characterized into one of the three following categories: a. ISRS with well separated peaks, for which the clipping procedure is applied to each peak. b. ISRS with closely coupled peaks, for which the bandwidth ratio of both peaks governs. c. ISRS with intermediate coupling, for which the maximum clipped spectral value determined for the set of peaks governs. The fragility analyst carefully looked at the results to determine the reasonableness of clipped spectra. In cases where the broadband clipped spectra fall below the valleys between coupled peaks, the ISRS was raised to spectral level corresponding to the valley. The clipping performed for the ISRS used in the first, second and third quantifications is reviewed and checked to ensure that clipping procedure described above is implemented consistently. Subsequently, the associated functional capacity of the SSCs that are affected are updated. Any change in the fragility value is notified to the PRA team for use in the final risk quantification. To resolve this F&O all the appendices in Fragility Notebook TVAESQN010-REPT-001 are	FO 5-6.pdf . TVAESQN010-REPT-001 - Part B and Part C.	Maintenance	Update to established methodologies with minimal impact on results	RESOLVED: Spot check of affected Appendices of Fragility Notebook TVAESQN010-REPT-001 verified clipping procedure veracity related to the F&O comment(s). The peak clipping procedures are accurately (re-)performed in accordance with EPRI 103959, EPRI 6041, and EPRI Fragility Training.	CC I/II

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
						reviewed and revised, as required.					
2	SFR-E3	Not Met	6-2	DOCUMENTATION for the screening out of components during the walkdowns, including anchorage calculations that justify the screening is insufficient to determine the basis for screening out of certain components. Originated from SR SFR-E3: Not Met; Finding.	The use of Note 1 in Appendix A of the Walkdown Notebook (Report No. TVAESQN10-REPT-003) provides multiple options for screening out, some of which are based on the walkdown while others are not. Therefore, it is not clear which components were screened out solely based on the walkdowns. In addition, anchorage calculations for components screened out based on the walkdown were not provided in the Walkdown Notebook.	1. The process associated with the 'screening' of the components listed in Appendix A of Walkdown Notebook (TVAESQN010-REPT-003) is described in Section 3.3.4, Screening SEL. Initial screening of SEL prior to walkdowns was performed iteratively by PRA systems analysts and Fragility analysts. Components that were identified to be inherently rugged, rule-of-the-box, balance-of-plant, non-consequential and non-safety related were screened out from the walkdowns. During the walkdowns, the SRT performed area walk-bys and looked for the screened-out components to validate the screening judgement. 2. Appendix A of Walkdown Notebook (TVAESQN010-REPT-003, Rev. 1) is updated using the same screening codes as used in SEL Notebook (MDN0009992017000047, Rev 1). Where deemed necessary, multiple screening codes were assigned for SEL items that were screened based on walkdown observations. 3. During the walkdowns, the SRT inspected the equipment base stiffness and anchorage condition. In some cases, the SRT judged that the anchorage is rugged and has sufficient strength to achieve the	F&O 6-2.pdf ENERCON, "Seismic Walkdown Report …." Section 3.3.4 "Screening SEL"; TVAESQN010-REPT-003 Rev 1. ENERCON, "Seismic Walkdown of Units 1 & 2 ….", TVAESQN010-REPT-003 Rev 1; "Appendix A: Seismic Equipment List" and "Appendix G: FLEX Items SEWS". TVA, "SQN Seismic PRA Seismic Equipment List", MDN0009992017000047 Rev 1.	Maintenance	Primarily documentation update	RESOLVED: Verified consistent screening codes are identical between TVAESQN10-REPT-003 Appendix A (Walkdown Report) and MDN0009992017000047, Rev 1. (SEL Notebook) in Appendix G (pg. 45). Also verified use of new codes in associated tables for each report. Basis for screening has been refined. Verified anchorage calculation in Appendix G.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
						functional capacity. The anchorage for these components was screened out by inspection and the fragility was assigned as lesser of functional and seismic interaction fragilities. Where necessary to support the screening judgement a calculation is performed. For example, see Appendix G FLEX items SEWS.					
2	SFR-E4	Met	6-3	The screening of certain non-safety related electrical equipment identified in the Fire PRA for consideration during the Seismically Induced Fire and Flood walkdowns, resulting in failure to EVALUATE these components as potential seismically induced fire sources during the walkdowns. Originated from SR SFR-E4: Met; Finding.	Table 4 of the Walkdown Notebook (Report TVAESQN10-REPT-003) lists the potential seismically induced fire sources that were identified in the Fire PRA. Note 2 indicates that several electrical cabinets were screened-out because they de-energized due to LOSP (Offsite Power has very low capacity). However, there is no guarantee that the LOSP will always occur during the seismic event, so that there is some probability that the equipment will still be energized and could result a fire due to high energy arcing - beneficial failure. Note 1 indicates that several oil-filled electrical transformers were screened out as credible sources of seismically induced fires, but the basis for this screening out is not documented. The response provided by TVA during the peer review indicated that these oil-filled transformers are also considered to be de-energized due to LOSP (similar to Note 2).	The seismic-induced fire walkdown documentation was updated to reflect the actual detailed walkdown that was performed. This walkdown addressed SQN 480V and higher non-safety-related (NSR) electrical components including MCCs, switchgear, and oil-filled transformers as identified by the SPRA quantification team based on fire PRA studies. For F&O resolution bounding configurations were identified for fragility analysis. The 480V NSR MCCs were judged to be the bounding configurations based on anchorage, aspect ratio, and elevation in the plant. Anchorage capacity did not control over the functional capacity of the MCCs, so the fragility analysis was conservatively performed based on seismic experience data functional capacity. To perform the seismic-induced fire screening analysis, a point estimate SCDF was calculated by convolving the governing fragility of a single MCC with the site-specific hazard curve consistent with the SPID and including a 2% probability of ignition for the MCC in accordance with EPRI guidance. The resulting point estimate SCDF = 1.181 x10-7 is less than 5.0 x 10-7, thus seismic-induced fire is screened out from the SQN SPRA.	FO 6-3 ENERCON, "Seismic Walkdown of Units 1 & 2 ….", TVAESQN010-REPT-003 Rev 1; "Appendix D5: July 2017 Units 1 & 2 Operator Path, Seismic Induced Flood and Fire Walkdowns SEWS (Third Phase SPRA Walkdowns". ENERCON, "Detailed Component Fragilities for use in Third Risk Quantification…." Report No. TVAESQN010-REPT-001-Part C, Rev 1; Appendix AY: "Seismic Fragility Evaluation of the Non-Safety related 480V MCC.	Maintenance	Documented justification for existing methodology	RESOLVED: Anchorage of the electrical components were assessed and found not governing. Functional capacity governs and results in an annualized frequency of exceedance (AFE) of less than 5E-7 which forms the basis to screen out SQN 480V and higher non-safety-related (NSR) electrical components including MCCs, switchgear, and oil-filled transformers. SUGGESTION: Correct F&O response to point to REPT-001 Appendix AY as opposed to Appendix AX.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
2	SFR-G2	Met	6-5	The DOCUMENTATION of the justification for not considering Structure-Soil-Structure Interaction Effects (SSSI) between the Reactor Building (RB) and the Refueling Water Storage Tank (RWST) is not adequate. Originated from SR SFR-G2: Met; Finding.	FRC Report 'Structure-Soil-Structure Interaction Effects between Reactor Building and Refueling Water Storage Tank' provides justification for not considering SSSI between the concludes that the SSSI effects of the RB on the input to the RWST is not significant, based on a comparison of the three-component in-structure response spectra (ISRS) at the RB basemat, generated from the seismic SSI response analysis of the RB alone, with the corresponding three-component free-field foundation input motion response spectra (FIRS) at the bedrock outcrop. The reduction in the ISRS at high frequencies relative to the FIRS is anticipated at the RB foundation, due to the kinematic and inertial SSI effects. Nevertheless, the RB seismic vibration can still modify the ground motion on the free surface nearby its foundation, especially around the RB and surrounding soil column frequencies (although the resulting displacements at the grade level are small). Therefore, the potential SSSI effects on the RWST resulting from RB vibration (through the backfill/soil deposit) are not addressed. NOTE: The RWST is founded on a soil deposit from the ground surface at El. 705 ft. to the bedrock nominally at El. 679 ft, whereas the RB is embedded in Class "A" backfill over a portion of its perimeter from El. 679 ft to El. 703 ft.	To respond to the Peer Review F&O 6-5, a supplemental evaluation of the potential SSSI effects due to dynamic response of the RB embedment soil layer on the SSI response of the soil-surface-supported RWST is performed. As the current seismic SSI response analysis of the RB does not include soil embedment of the RB above the bedrock, the seismic response of the embedment soil layer above bedrock cannot be obtained directly from the results of the current SSI analysis of the RB. Thus, to evaluate the potential SSSI effects from the RB to the RWST using the currently available SSI analysis results, an approximate evaluation approach is taken in this supplemental evaluation by, first, identifying the source and amplitude of the seismic excitation motion (i.e., seismic driving motion) that will cause dynamic response of the soil layer that embeds the RB and, then, determining the seismic response motion of the embedment soil layer at the soil surface at the RWST site location by using an approximate surface-wave-propagation attenuation relation. The approximate soil surface motion at the RWST site location caused by the seismic driving motion of the RB so obtained is then combined with the free-field soil surface seismic input motion as characterized by the foundation input motion response spectra (FIRS) for the RWST to assess to what extent the response spectra of the combined motion that includes the SSSI effects on the soil response from the RB exceed the free-field foundation input response spectra (FIRS) of the RWST. Based on the results of approximate evaluation described in this supplemental	Final signed Response to Peer Review FO 6-5(1-12-19) signed.pdf.	Maintenance	Study with no effect on results.	RESOLVED: The assessment is reasonable and even somewhat conservative. The driving motion and the FIRS derive from the same earthquake, so SRSS is justified but will also always increase above the FIRS. Reviewers concur that the rock founded RB has no appreciable effect (SSSI) on the soil-founded RWST.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
						evaluation, it can be concluded that the SSSI effects of the bedrock-supported RB embedded in the 44-ft thick soil layer on the soil-surface-supported RWST, including the effects caused by SSSI-induced soil response of the embedment soil layer, are insignificant for the SQN RWST.					
2	SFR-G2	Met	6-6	The basis for the assignment of the Steel Containment Vessel (SCV) fragility parameters to Fragility Group 'SEIS_0-2' (Electrical Penetrations and Bellows Penetrations) is not DOCUMENTED in the Fragility Notebook. Originated from SR SFR-G2: Met; Finding.	The entries in Table 7 (SSC Fragility Summary) of the Fragility Notebook (Report No. TVAESQN10-REPT-001) for Fragility Group 'SEIS_0-2' (Electrical Penetrations and Bellows Penetrations) indicates that Failure Mode is 'Structure Fragility/Same as SCV' and the Status is 'Q1-Structure'. However, Table 1 (SQN Equipment Class 0 - Miscellaneous Items) of Appendix F (Seismic Capacity Review of SQN Equipment) of Part A (Representative Fragilities) of the Fragility Notebook indicates that these components were Screened Out and Bounded by LOCA Analysis. Appendix B (Development of Structural Fragility for SQN SPRA) of Part A of the Fragility Notebook includes the fragility evaluation of the SCV associated with the 'Q1-Structure' assessment does not document a comparison of the capacity of the penetrations with the capacity of the SCV in order to provide justification for the applicability of the SCV fragilities to the penetrations.	As recommended, a discussion of the basis for the assignment of the SCV fragilities to the penetrations in Fragility Group SEIS_0-2 was added to the Fragility Notebook.	F&O 6-6.pdf. ENERCON, "Representative Fragilities ….", Report No. TVAESQN010-REPT-001-Part-A; Appendix B: "Development of Structural Fragility for SQN SPRA" and Appendix G: "Representative Fragilities of SQN SSCs".	Maintenance	Simple change in fragility determination and mapping	RESOLVED: Verified SCV screening value updates to Appendix G, TVAESQN010-REPT-001- PART A that document the summary of fragility parameters used in Q1 and subsequent quantification; see Page 14 of 85 in SEL and Walkdown Notebooks. SUGGESTION: Update description in Closure Review Spreadsheet to be consistent with F&O response regarding bellows and penetration fragilities actually used.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
2	SFR-D2	Met	7-1	Pdf page 953 of Part A of the Fragility Notebook states that 125V DC Vital battery board Group 1A-7 is governed by the 120v AC Vital instrument Power Boards (which is group 1A-8). This basis is taken from the IPEEE report. It then develops a Q1 fragility for 1A-7 based on the failure mode of a half inch bolt on the 120v board but uses the frequency of the 125v board. After discussion with the SPRA team and then seeing the components in the peer review walkdown, it is noted that these two components are entirely dissimilar in both equipment and anchorage configuration. The basis for using the capacity of one panel as bounding for another may be appropriate for IPEEE since it was a margin assessment, this is not the case for SPRA. Further, the IPEEE analysis simply states the weakest failure mode between the two panels and assigns the HCLPF to both accordingly. The reported SPRA analysis improves the fragility Group 1A-7 by using the failure mode of Group 1A-8, but the frequency data of 1A-7. This amounts to an apples-to-oranges comparison. It is not appropriate to use the failure mode of one panel and the frequency data of another to develop a capacity. It is also not appropriate to use a generic scaled fragility from IPEEE for a component that is a top contributor to the model. Originated from SR SFR-D2: Met; Finding.	Realistic failure modes must be IDENTIFIED. By referencing the failure mode of the 120V Panel, a more accurate failure mode for the 125V Panel has not been considered. The 125V Panel is a top contributor to the model according to Table 6 of the Fragility Notebook. It must be ENSURED that the fragility for this top contributor is reasonably non-conservative. The use of a generic scaled fragility is not appropriate in this case.	A detailed fragility evaluation is performed using the hybrid approach.	FO 7-1Rev 1.pdf ENERCON, "Detailed Component Fragilities for use in Third Risk Quantification...." Report No. TVAESQN010-REPT-001-Part C, Appendix BF.	Maintenance	Primarily fragility value documentation update	RESOLVED: Fragility evaluation is performed in a complete manner for the 125V DC Vital Battery Boards. SUGGESTION: Correct F&O response to point to the Appendix BF for fragility evaluation and BE for block wall evaluation. Also correct black wall fragility parameters in F&O response to be consistent with Appendix BF.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
2	SFR-G2	Met	7-2	There are several disconnects between the Information in the SEL Document MDN0009992017000047 and Table 7 of the Fragility Notebook. There is multiple equipment that may have been included in one, but not in the other. Additionally, there are several columns of information that may be included in one document, but not the other. It appears that the SEL had changed after the SEL document had been finalized (e.g., Screened items and FLEX equipment). Originated from SR SFR-G2: Met; Finding.	The processes from the SEL Development to the Fragility Analysis must be DOCUMENTED. Information between fragility analysts and PRA analysts must be the same in order to reduce modeling errors.	Comprehensive SEL developed by PRA systems analysts (MDN0009992017000047, Rev 1), which includes updates after peer review, is reviewed and used by fragility analysts and walkdown teams for verification of data tabulated fragility and walkdown notebooks. Previously, the components screened out from the walkdown were not documented exclusively in fragility notebook. For consistency between fragility and SEL notebook, the following information is included in fragility notebook: UNID, Unit, Description, Component type, Building / Floor, Room, Elevation, Source, Screening Notes, Component Screened prior to walkdown?, Component screened after walkdown?, WD/WB, EPRI class, Fragility Group, Anchorage HCLPF, Functional HCLPF, Block Wall HCLPF, Failure mode, Overall HCLPF,βR, βU,βC, Am, Fragility notes. For consistency between walkdown and SEL notebook, the following information is included in walkdown notebook: UNID, Unit, Description, Component type, Building / Floor, Room, Elevation, Source, Screening Notes, Component Screened prior to walkdown?, Component screened after walkdown?, WD/WB, EPRI class, Fragility Group, Documentation reference. Screened out components are included for completeness. The changes are incorporated in the fragility and walkdown notebook.	F&O 7-2.pdf. TVA, "SQN Seismic PRA Seismic Equipment List", MDN0009992017000047 Rev 1. ENERCON, "…Fragility Analysis Notebook"; TVAESQN010-REPT-001 Rev 1.	Maintenance	Primarily documentation update	RESOLVED: Reviewed revisions to SEL (MDN0009992017000047, Rev 1), Walkdown and Fragility Notebooks to verify consistency of metadata (e.g., UNID) and screening codes. Information in Walkdown and Fragility Notebooks has been updated as described, which is sufficient to resolve the F&O.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
2	SFR-G2	Met	7-3	Page 28 on SQN-17-001 Rev. 1 (SQN Structural Response Analysis) states that hard rock coherency functions are conservatively used for structures founded on soft rock/soil sites as well. However, documentation is lacking on SQN-17-001 Rev. 1 to justify using conservative hard rock coherency functions for structures on soil sites, which house certain SEL equipment whose fragility is governed by high frequency response. Also, documentation regarding to comparison of coherent and incoherent response is lacking in the same report which is important to understand whether the incoherency effects are incorporated appropriately. Originated from SR SFR-G2: Met; Finding.	The difference between Abrahamson soft rock/soil and hard rock coherency functions in EPRI TR-1015110 seems noteworthy (Figs. 5-7 to 5-9 in EPRI TR-1015110). If the incoherency effects are important for structures sitting on soil sites (e.g., DGB and ADGB), this might lead to overly conservative estimates in the high frequency range. However, justification for not using best estimate soft soil coherency functions for DGB and ADGB is not DOCUMENTED.	Comparisons of coherent versus incoherent response, both in terms of transfer functions and in-structure response spectra (ISRS), are added to SQN-17-001 Rev. 2 for the ACB (Appendix F), DGB (Appendix L), and ADGB (Appendix O). For the ACB, which is founded on a limestone interbedded with shale rock (best estimate shear-wave velocity of 4800 fps), consideration of incoherency causes reduction of high-frequency response consistent with that expected for the Abrahamson hard rock coherency function. Given the foundation conditions of the ACB, the Abrahamson hard rock coherency function is more appropriate than the Abrahamson soft rock/soil coherency function. For the DGB and ADGB, which are founded on soil, consideration of incoherency does not significantly affect structural response. This is noted in the report revision. Therefore, potential conservatism in the use of the Abrahamson hard rock coherency function (versus the Abrahamson soft rock/soil coherency function) is not significant for the seismic demands of SEL equipment housed inside the DGB and ADGB.	Final signed Response to FO7-3 (Rev 0).pdf. SC Solutions, Inc., "Sequoyah Nuclear Plant Seismic Probabilistic Risk assessment: Structural Response Analysis", SQN-17-001 Rev 2. ACB (Appendix F), DGB (Appendix L) ADGB (Appendix O).	Maintenance	Primarily documentation update	RESOLVED: Checked Appendices F, L & O of report SQN-17-001, R2 to check differences in ARS. The ACB is a mostly rock foundation so incoherent response effectively filters higher frequency (>10Hz) ground motion as expected. The DGB and ADGB are soil situated and consideration of incoherency effects are not significant.	Met
2	SFR-B1	CC I/II	7-5	During the peer review walkdown temperature sensor 2-TS-300-450B upstairs in the DGB was found to be mounted on a block wall. There are three other TS's on this same wall, and it is expected that there are four on the wall for each DG compartment. This amounts to 16 TS's in total. The TS is in MDN0009992017000047 the SEL document but is not walked down. It says location information is unknown and that it is screened because it is an electrical switch. It is not appropriate to screen this element as rugged since it is mounted on a non-rugged	The temperature elements were screened from the fragility analysis as rugged, but this was not DOCUMENTED. As a result, a component that should have had a fragility value was screened.	The temperature sensors identified here were re-screened. The re-screening identified that the temperature sensors are mounted on a block wall. The SEL documents this info. The fragility team reviewed and concurred with the screening. PRA Evaluation SQN-0-18-110 documents a review of screened out components to ensure that this specific instance is not systemic.	PRA Evaluation SQN-0-18-110 Rev 1. .	Maintenance	Incorporated established methods for block walls to evaluate relevant components	RESOLVED: Review is comprehensive and ensures all devices mounted on block walls are identified. This identification is carried over into Appendix G of the walkdown report. Spot check shows these are also carried over the fragility notebook and assigned block wall fragilities as described.	CC I/II

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
				block wall. It appears that the element was screened from the SEL by PRA team before being sent to the fragility team for fragility and walkdown. These were seen on an area walkdown, but no notes were seen in the Walkdown Notebook. Either this is not noticed, or they inadvertently disregarded the element as not relevant. There is a disconnect between screening done by the PRA team and whether the walkdown should have screened something. Originated from SR SFR-B1: CC I/II; Finding.							
2	SFR-E1	Met	7-6	The FLEX System installed in the ADGB was not walked down fully. The most recent walkdown of the system was for the ESEP in 2015. At that point, not everything in the building had been installed yet for FLEX. Originated from SR SFR-E1: Met; Finding.	A walkdown of all systems needed for the SPRA has not been CONDUCTED. While on the peer review walkdown, multiple items were noticed. 1. When entering the ADGB, a potential II/I interaction affecting operator pathways was noted in two large blue cabinets, in a written response to a question the SPRA team notes that multiple pathways ensure that the FLEX can be accessed if needed. However, this is not documented anywhere, and the SPRA team had stated that no operator pathway walkdown was done for FLEX equipment. 2. Multiple components were seen that are not on the SEL but are needed to bring the FLEX System into operation in the ADGB. It is expected through other F&O's; these items will be brought into the SEL. 3. Components 0-RES-360B and 0-BD-360-3B are expected to be needed for the FLEX system but are not on the SEL currently. Both components are in the Walkdown Notebook as having their ESEP walkdowns credited. This is on page D1-779 to D1-784. There are no walkdown notes for these, the anchorage pad appears to show no signs of degradation in the older ESEP photos. However, the peer review walkdown noted several	FLEX components were initially walked down as part of the ESEP in 2015 and several equipment that were supposed to be permanently installed for FLEX in Additional Diesel Generator Building (ADGB) were not yet installed. A list of components containing FLEX systems was prepared and a walkdown was performed post peer-review on 08/07/2018 to walkdown the additional FLEX components that were installed later. Accessibility to various rooms of ADGB for any required operator actions were also assessed. The walkdown observations are documented in Appendix G, TVAESQN010-REPT-003, Rev 1 (Walkdown Notebook).	FO 7-6 with attachment-signed (2).pdf TVAESQN010-REPT-003 Appendix G Rev 1.pdf.	Maintenance	Primarily documentation update	RESOLVED: For concern 1, it is clearly presented that an alternative pathway exists and was considered to accommodate the unanchored cabinet. For concern 2, approach is comprehensive and well-documented, including references to calculations as needed. For concern 3, this is covered under the concern 2 approach which includes all previously walked down items. SUGGESTION: Based on discussion, it's clear that the only FLEX equipment operator pathways are in the ADGB and are therefore covered. Suggest clarifying this in response.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
					large cracks in the anchorage pad these components share, these seem to be new cracks since the ESEP walkdown. This is different from what is presented in the walkdown notebook. There are multiple cracks going under the switchgear (the anchorage itself cannot be seen), and there is a large crack going through an anchor bolt on the Grounding Resistor. In response to a written question, the SPRA team stated that TVA is aware of the cracking and the plant has accepted the as-is condition for design basis. However, there is no documentation for this in the SPRA notebooks, and a fragility calculation for these has yet to be developed.						
2	SFR-G2	Met	7-9	1. Section 3.2 of the Structural Response Analysis Report states the WBN SPRA models are heavily credited for the SQN SPRA. However, this section references a version of the WBN SPRA model report that does not include the resolution of F&O's from that Peer Review. 2. The Fragility Notebook uses scaled building fragilities from WBN in the Q1 analysis. These fragilities are taken from a version of the WBN fragility that had not incorporated F&O's from their peer review. Given that the SQN analysis began with models from WBN it is likely that SQN and WBN would share many of the SAME F&O's unless the SQN models referenced WBN models that have closed all F&O's from their model. Originated from SR SFR-G2: Met; Finding.	The process for how the WBN models is credited and how their F&Os are addressed shall be DOCUMENTED,	PART 1: A review was performed of the Watts Bar SPRA Finding Level F&O Independent Technical Review Report (Report No. 06044-RPT-01, Revision R1) which contains the list of all WBN SPRA F&Os and their final resolutions as accepted by closure review. The purpose of this review was to document how the F&O's related to WBN structural modeling, and their resolution, are applicable to the structural models that are credited in SQN SPRA. In addition to the assessment of the WBN F&Os presented herein, the reference to WBN SPRA structural response analysis is updated in the SQN building response analysis report to reflect the latest revision (i.e. Revision no. 2) of the WBN SPRA building response analysis report (TVA Doc. CDN0000002015000711), which includes the resolution of the WBN F&Os. PART 2: Watts Bar Nuclear Plant (WBN) is a sister plant to Sequoyah Nuclear Plant (SQN). The general outline and equipment layout at SQN are essentially same as that of WBN. During SQN PRA first quantification, some of the SQN component and building	FO 7-9.pdf. Part 1: CDN0000002015000711 revision 2.pdf. Part 2: TVAESQN010-REPT-001 Part A Rev 2.pdf TVAESQN010-REPT-001 Part C Rev 1.pdf CDN0000002017000710 rev 2 June 2017.pdf.	Maintenance	Generally, no methodology change, but update of fragility values based on changes to source values.	RESOLVED: A thorough review of the WBN F&Os was performed and potential effects on the SQN SPRA are documented. Structural model issues are deemed resolved based on WBN peer review which resulted in no changes. Changed (updated) fragility capacities were verified (spot checked) in SQN Fragilities Notebook Part A appendices AX & B.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
						fragilities were scaled from WBN seismic fragility analysis documented in TVA Report CDN0000002015000710, Rev. 0, September 2015 (Ref. 1). Ref. 1 was updated to address WBN peer-review F&O and Rev. 1 was issued in February 2017 and Rev. 2 was issued in June 2017. Based on a through documentation review, SQN fragility groups which could have been potentially affected due to use of revised WBN fragility and response analysis documentation are identified. Seventeen fragility groups in TVAESQN010-REPT-001 – PART A (Q1 fragility report) and three fragility groups in TVAESQN010-REPT-001 – PART C (Q3 fragility report) are identified where WBN information was used. The detailed response, "F&O 7-2.pdf", lists the fragility groups that are affected and provides a resolution.					

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
2	SFR-F2	CC I/II	7-10	Group SEIS_0-30-5 is analyzed in Appendix AK of the Fragility Notebook. This top contributor is a relay model GEHEA99BT according to the PRA Chatter Analysis Report. For this, 10g GERS are used based on meeting the caveats of EPRI NP-7147. However, the vendor manual for this relay on GE's website simply adds a cover sheet to the HEA 61 model relay with a statement that the two relays are identical aside from a single contact. It notes that Contact 11 is actually similar to contact 5 in the same model of relay. If this similarity is confirmed, then a more robust 14g GERS level can be used based on the caveats in EPRI 105988. Originated from SR SFR-F2: Met; Finding.	Relays needed for the SPRA model shall be CALCULATED. Additionally, since this is a top contributor, it must be ENSURED that fragilities are realistic for significant accident sequences in the SPRA model.	To address this F&O an extended condition review was performed to identify if there are any relay types that are documented incorrectly in Chatter Analysis Notebook, TVAESQN010- REPT-002 (Ref. 1). From documentation review, it was confirmed that the relay manufacturer and relay type of lockout relays in Unit 1,2 SQN 6.9kV Shutdown Boards is GE HEA. However, the relay model for 86S lockout relays is identified as GE 12HEA61, whereas the remaining lockout relays (86-718, 86-716; 86-728, 86-726; 86-818, 86-816; 86-828, 86-826;) are identified as GE 12HEA99BT. Appendix AK of Fragility Notebook (Ref. 2) used the EPRI NP-7147-SL GERS capacity of 10g corresponding to GE 12HEA61A, B and C lockout relays. An initial assessment was performed to determine the lockout relay type by reviewing the bill of materials. Subsequently, a plant walkdown was performed by TVA Plant Ops (Jay Whitworth) to confirm the as-installed and as-operated relay type of all lockout relays in 6.9kV Shutdown Boards. The walkdown findings were provided to the fragility team in an email transmittal, which is provided as an attachment. Chatter Analysis Notebook (Ref. 3) is updated and Rev. 2 is issued that incorporates the appropriate relay type of lockout relays. As pointed out by PRT, the seismic capacity of GE HEA relay in EPRI TR-105988 is 14g. However, this robust capacity is based on the testing of GE 12HEA61A225-X2. TVA contacted EPRI to confirm if the seismic capacity of GE 12HEA61A225-X2 is applicable to other GE 12HEA61B and C lockout relays. EPRI affirmatively confirmed that the seismic capacity reported for GE HEA relays in EPRI TR-105988	FO 7-10 with Attachments.pdf. ENERCON, "….Chatter Analysis Report", Report No. TVAESQN010-REPT-002, Rev 2 Chatter Analysis Notebook.	Maintenance	Update to several fragility values without methodology change	RESOLVED: Based on J. Richards explanation that TR-105988 relay testing results for GE HEA61X225 applies to all models of GE HEA61 relay models (A, B & C at a minimum) the use of relay capacity of 14g is justifiable. Applicable relay capacities are updated consistently throughout the documentation	CCII/II

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
						supersedes the seismic capacity reported in EPRI NP-7147-SL and is applicable to all GE HEA61 lockout relays (see Attachment 2). The fragility of lockout relays is updated in Appendix AK of Fragility Notebook (Ref. 2). A sensitivity analysis is performed by the PRA analysts by increasing the seismic capacity of the affected relay group by a factor of 1.4. The results of the sensitivity study are documented in SQN-0-19-005.					

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
2	SFR-G2	Met	7-12	Section 2.6 of the Final SEL Document MDN0009992017000047 states that polar cranes are known II/I concerns and that the polar crane for each unit is specifically added to the SEL based on being an interaction concern for the NSSS. Additionally, Table 4 Note 1 of the Fragility Notebook states that the crane is included as part of the Class 0 components. However, the crane is not in Table 7 of the Fragility Report and there is no discussion of the crane in either Walkdown or Fragility Notebooks. There is no documented reconciliation of this conflict. All walkdown SEWS in Appendix D1 state that no II/I interactions were noted in upper containment of the RB. However, this statement is broad and does not cover a specifically screened in SEL component. During a discussion with the SPRA team during the peer review, the SPRA team said the polar crane was screened as a non-realistic II/I concern based on the cranes inability to come off the crane wall and impact the NSSS. However, this is not documented anywhere. Originated from SR SFR-G2: Met; Finding.	Containment cranes are commonly looked at in plant SPRAs. There is a disconnect between the SEL document and the Walkdown/Fragility Notebook on whether the crane is relevant. It should be DOCUMENTED what was actually done.	The report "TVA SQN SPRA Polar Crane Fragility" provides clear documentation that the polar crane can be screened as a non-realistic II/I concern based on the cranes inability to come off the crane wall and impact the NSSS. Two design features prevent the crane from coming off the crane wall and impacting the NSSS; (1) bumpers that preclude any possible movement parallel to the crane bridge and (2) trolley wheels outfitted with rugged uplift restraints.	F&O 7-12 R1 SQN Polar Crane Fragility 2 07 2019.pdf	Maintenance	Primarily documentation update	RESOLVED: TVAESQN010-REPT-003 Appendix G Rev 1 a screening assessment for excluding (screening out) the polar crane was reviewed. The failure (collapse of the polar crane) was deemed not credible due to constraining geometry and seismic displacements necessary to dislodge the crane bridge. It is concluded that the response fully addressed all the comments offered by the Peer Review Team. Justification of screening the polar crane as a plausible II/I is complete and appropriate. SUGGESTION: Consider if the final documentation is sufficient to address the "discrepancy" from the SEL document. Suggest final resolution address whether polar crane is an SEL component.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
2	SFR-G2	Met	7-13	In the Walkdown/Fragility Notebooks, there is no discussion of the pounding failure mode between the RB and AB. This is a common failure mode addressed in SPRAs. During the peer review walkdown, the peer reviewers were told that the gap closure fragility was acceptable based on the high gap closure fragility calculated from WBN. While on the peer review walkdown, the gap between the RB and AB at elevation 759 around the MG Sets and Pressurizer Heater Transformers appeared to vary from about ¾ inch to 1-1/2 inch. There is no discussion provided for this like the supplied memo that resolves the TB to CB gap closure. Additionally, there is no discussion that the calculated building displacements in Appendix S of the Structure Response Notebook were used to develop a gap closure fragility. Originated from SR SFR-G2: Met; Finding.	Gap closure is an important failure mode that is typical for SPRAs. The basis for screening of this failure mode must be DOCUMENTED.	Whitepaper "SQN SPRA Building Impact" documents evaluation of gap closure for the space between the RB and AB and other critical gap locations. The evaluations use the calculated building displacements in Appendix S of the Structure Response Notebook. In all cases, there are no SEL components in the vicinity of the potential building impact zones that could be affected by possible local structural damage such as concrete spalling. The evaluations demonstrate that the gap closures (this includes building pounding and the associated impulsive shocks due to the impact) may govern the fragility for relay groups 0-30-6 and 0-30-9.	FO 7-13-signed.pdf.	Maintenance	Update to several fragility values without methodology change	RESOLVED: Reviewed Facility Risk Consultants, Inc., “SQN SPRA Building Impact Assessment,” Prepared for TVA dated November 15, 2018. Calculation is reasonably conservative and deemed acceptable. Resulting fragilities are updated consistently through documentation and Building Impact Assessment is included as Attachment 4 to REPT-001 Appendix AK.	Met
2	SFR-G2	Met	7-14	The backfill material under the ESVR structure is also modeled as part of the SASSI model in SQN-17- 001 Rev. 1 (SQN Structural Response Analysis). However, justification for not discretizing the backfill mesh for a passing frequency of 50 Hz is not presented. Originated from SR SFR-G2: Met; Finding.	Table 3 on of Page 419 on SQN-17-001 Rev. 1 reports that the average shear-wave velocity of the backfill is about 650 ft/sec. The corresponding element size for a passing frequency of 50 Hz would be hmax=650ft/sec/(5*50Hz)=2.6 ft. The vertical mesh size of the solid elements representing the backfill under ESVR is 5.8 ft. However, potential effects of the coarse meshing are not DOCUMENTED.	The "SQN Building Analysis Report" was revised to document that a mesh size in the SSI analysis that allows for a passing frequency of 50 Hz is not needed because no high frequency components are in this structure which would justify that a passing frequency of 50Hz is needed.	Final Signed Response to FO7-14 (Rev0).pdf SC Solutions, Inc., "Sequoyah Nuclear Plant Seismic Probabilistic Risk assessment: Structural Response Analysis"	Maintenance	Primarily documentation update	RESOLVED: Discussion and documentation is provided justifying not refining the supporting soil elements of the ESRV to pass high-frequency motions. Discussion is credible and addresses the concerns raised in the F&O. In addition, the only components in consideration in the ESRV are some power-actuated valves and they are not high frequency susceptible, nor are they risk-significant, so no concern with high frequency response due to inadequate mesh discretization exists.	Met
2	SFR-G2	Met	7-16	The following issues relate to documentation of sine beat testing: 1) Functional capacity of some SSCs in the Fragility Report TVAESQN010-REPT-001 (e.g., SEIS 16-1 Aux Control air compressors – page 895 of Part B APPENDIX X and SEIS 5-4 Panel and Cabinet – page 1065	1.) As shown in EPRI NP-5223-SL (Sec. A.3), the amplification factor depends on the number of beat cycles (7.55 corresponds to 10 cycles/beat). 2.) The corresponding figures illustrate the transmissibility ratios rather than the response spectra associated with the sine-beat motion.	This F&O relates to documentation of the sine beat test data used to evaluate the functional capacity of SSCs. Even though the F&O describes three instances, an extended condition check is performed for all functional fragility evaluations that are based on either sine beat test or sine dwell test data. For all functional fragility	FO 7-16 Rev 1.pdf. ENERCON, "Detailed Component Fragilities for use in Second Risk Quantification...." Report No. TVAESQN010-REPT-001-Part B, Rev 1;	Maintenance	Primarily documentation update	RESOLVED: Review of referenced Appendices of the Fragility Notebook demonstrates proper (correct) application of sine beat-to-response spectrum conversion following EPRI Report NP-5223-SLR1 methodology, thus addressing the concerns raised in the F&O. Discrepancies noted in Appendix AK Clipped TRS and Clipped TRS Envelopes cited in the F&O have been corrected.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
				of Part B APPENDIX AA) is based on sine beat testing. An amplification factor of 7.55 is used constantly to obtain the spectral capacity without fully DOCUMENTING the testing characteristics to justify the amplification factor. 2) Response spectra generated from sine beat motion does not have the correct shape (e.g., Figure 13 of page 896 in Part B APPENDIX X), although it has the correct peak spectral value. 3) Potter & Brumfield Relay Model No. MDR 134-1 (SEIS 0-30-4) (page 1730 of Fragility Report Part B APPENDIX AK) uses sine beat testing for the capacity. The report states that the excitation input was up to 3.0g horizontally and 2.0g vertically. However, the horizontal and vertical TRS shown on Figure 15 & Figure 16 in Part B APPENDIX AK do not match with the description of the sine beat testing. Originated from SR SFR-G2: Met; Finding.	3.) TRS accelerations corresponding to the 3g sine beat test are expected to be higher than the reported values.	evaluations that are based on either sine beat or sine dwell test data the fragility report is revised to (1) document the beats per cycle used in the sine beat test from the corresponding seismic qualification test report (2) present the correct response spectra shapes generated from sine beat motion and (3) ensure the description of the testing matches the TRS shape. The fragilities affected by this finding are addressed as discussed in the ""Detailed Response to F&O"". The updated fragility results have been provided to the PRA team for use in the final risk quantification.					
	SPR							Maintenance			
3	SPR-A1	Met	8-1	Seismically induced failure of the ERCW connection to the EDGs was modeled as resulting in failure of EDGs, but not ERCW. Originated from SR SPR-A1: Met; Finding.	MDN0009992017000044 describes the consequence of the ERCW connection to the EDGs; "ERCW buried piping is rugged except the connection interface to the EDG. Thus, the generic ruggedness fragility group SEIS_0-20 was assigned to new basic event ERCWPIPING_S and the seismic group SEIS_0-1 'Buried ERCW piping' was assigned to EDG failure of fail to run". A review of the ERCW system drawings from MDN-000-067-2010-0222 Revision 1 showed that an ERCW connection failure may result in a failure of the entire ERCW system due to a significant flow diversion.	Preliminary Response In accordance with the suggestion of F&O 8-1, the seismic fragility of the ERCW buried piping (SEIS_0-1) was mapped to the ERCW pumps in the seismic model. model. See SQN-0-18-121, Section 2.2.1. Updated SPRA Model stored in zip file with Quantification Notebook.	PRA Evaluation SQN-0-18-121, "SQN SPRA F&O Resolutions Summary"; Model Files	Maintenance	The model change to map the ERCW buried piping fragility to the ERCW pump is not a methodology change and is an update to logic that provides improved basis for the expected seismic response.	RESOLVED: Examination of the FRANX Groups to Components mapping for SEIS_0-1 shows that the set of basic events and component failures listed in SQN-0-18-121 are mapped to the ERCW buried piping fragility group. Sampled grouped basic events in the fault tree to see that impact is to fail ERCW.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
3	SPR-A1	Met	8-2	The Unit 2 SQN model does not include a contribution to excessive LOCA from seismically induced pressurizer failure. Originated from SR SPR-A1: Met; Finding.	The SQN_Seismic_Rev0-2-1-18.caf fault tree was reviewed as a basis for this F&O. Within the U2 fault tree, the gate U2_EXLOCA_S_NSSS represents seismic contributions to excessive LOCA. The S2RLCP_S basic event (Seismic - LOSS OF PRIMARY FLOW [RCP TRIP]) is used as one of the contributors, instead of the pressurizer failure basic event (PRZSUPPORTFAIL_S). The fragility values for the RCPs and pressurizer are very similar and so this substitution does not have a significant impact on the overall results. However, the Unit 2 pressurizer fragility importance is underestimated by not including this failure.	See Section 2.1.2 of SQN-0-18-121. As suggested in the "Possible Resolution" for F&O 8-2, the original gate logic shown in Figure 12 was replaced by that shown in Figure 13. The Unit 1 model was correct as-is and was not changed.	PRA Evaluation SQN-0-18-121, "SQN SPRA F&O Resolutions Summary"; Model Files	Maintenance	The model change to revise the pressurizer failure contributor is not a methodology change and is an update to logic that provides improved basis for the expected seismic response and basic event importances.	RESOLVED: Checked fault tree SQN_Seismic_Rev0-Post Peer Review and confirmed logic is as shown in Figure 13 of SQN-0-18-121. Also confirmed that U1 logic is same as corrected Unit 2 logic.	Met
3	SPR-A1	Met	8-3	Screening assessment performed for seismically induced fire eliminates components that may have a contribution to seismic risk. Originated from SR SPR-A1: Met; Finding.	Table J-1 of MDN000NA2017000042 documents the seismically induced fire ignition source screening process. Ignition sources from the FPRA were screened using 1 of 7 criteria, which are identified in sections 5.2.1 through 5.2.5. 1. Criterion 2 screening is applied if the SEL component of interest was screened using risk-based fragility screening. Some of the components screened using Criterion 2 (for example 0-BDB-201-SJ - 480V AUX BLDG COMMON BD BUS A) do not appear to meet any risk-based screening criteria (the fragilities for safety related 480V boards have a median capacity of 1.26g). Screening components that do not meet risk-based screening criteria may result in the exclusion of potentially significant seismically induced ignition sources. 2. The description for Criterion 3 states, 'Safety related equipment and those structures and components that have been screened on the SEL are screened out based on seismic capacity. Some of the components that are screened, using Criterion 3, do not appear to have been screened from the SEL. For example, 1-BDB-201-DK-A, "480V SHUTDOWN BD 1A2-A" is screened from further consideration for seismically induced fire using Criterion 3 but is	See SQN-0-18-121 R0 for the details on this response. For the latest revision of the SQN Fire PRA - Seismic-Fire Interactions Assessment for SQN [MDN000NA2017000042] it was revised as follows: Section 5.2.1.2 states: The screening process developed for the purposes of this report does not rely on the criterion for structures, systems, and components important to safety as stated on Appendix A of 10 CFR Part 50. Safety Related Equipment is set to No Screening with the letter "N" in column F, Screening 3 – Safety Related Equipment , of tab "S-F Screening Summary" of the Microsoft Excel spreadsheet embedded in Appendix J. Specifically component 1-BDB-201-DK-A, "480V SHUTDOWN BD 1A2-A" was screened-in based on seismic capacity. A similar process was performed for all the other similar components.	PRA Evaluation SQN-0-18-121, "SQN SPRA F&O Resolutions Summary"; SQN Fire PRA - Seismic-Fire Interactions Assessment for SQN (MDN000NA2017000042)	Maintenance	The re-assessment of the applicable criteria for certain elements of the screening is within the originally peer reviewed methodology and is an update to the screening results to correct the noted inconsistencies.	Initial Disposition PARTIALLY RESOLVED WITH DOCUMENTATION: SQN-0-18-121, PDF Pages 84 through 86 (Sections 2.6.11) were reviewed to confirm the basis for closure. MDN000NA2017000042, Revision 1, PDF Page (Section 5.2.1.2) was not available for review. However, a draft version of this report (File Name 006042.000-RPT-01 SQN SF 11-16-18) was reviewed to confirm that the screening methodology does not rely on the criterion for structures, systems, and components important to safety as stated on Appendix A of 10 CFR Part 50. Table J-1 was not modified in the draft report to display the changes resulting from the criteria clarification. Final Disposition RESOLVED, On March 22, 2019, TVA provided MDN000NA2017000042, Revision 1. Section 5.2.1.2 was reviewed to confirm that that the screening methodology does not rely on the criterion for structures, systems, and components important to safety as stated on Appendix A of 10 CFR Part 50 (Screening Criteria # 3). The report states that "*The screening process developed for the purposes of this report does not rely on the criterion for structures, systems, and components important to safety as stated on Appendix A of 10 CFR Part 50. Safety Related Equipment is set to 'No Screening' in column N (Screening 3 – Safety Related Equipment), tab "Screening Analysis" of the Microsoft Excel spreadsheet embedded in Appendix J.*" A review of Table J-1 indicates that there is no "Y" in the cells for Screening Criteria #3.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
					not screened on the SEL and has a median capacity of 1.26g.						

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
3	SPR-A5	Met	8-4	Screening reasons are missing from the seismic equipment list and should be added to ensure that the model reflects earthquake caused failures. Originated from SR SPR-A5: Met; Finding.	Approximately 300 components were screened from the SEL without an explicit screening code (from Appendix G of MDN0009992017000047 revision 0). In many cases, other database fields (not displayed in Appendix H of MDN0009992017000047 revision 0) could be used as a basis for screening. For example, SQN-1-FCV-067-0081-A was screened from the SEL without use of a screening code. However, under the miscellaneous notes table of the access SEL access database, the following reason, 'Not required to change position and fragility not needed' was added which implies that the NSC screening code should have been used. In a number of other cases, components were screened without a screening code and without additional screening notes. For example, SQN-0-TNK-032-0062 (AUX CONTROL AIR RECEIVER A-A) was screened and no additional screening notes were found.	This error was corrected after filtering through the SQN SEL post peer review for components missing a screening code and/or screening basis [“SQN Seismic PRA Seismic Equipment List”, MDN0009992017000047, Rev. 1]. A screening code was applied to all components missing input in this field with a basis provided.	“SQN Seismic PRA Seismic Equipment List”, MDN0009992017000047, Revision 1	Maintenance	The completion of the missing screening codes, using the existing Appendix G criteria is within the originally peer reviewed methodology, is an update to the screening results to add the missing documentation of bases for screening.	RESOLVED: Review of Appendix H of revised SEL notebook MDN0009992017000047, Rev. 1 shows that all components with a "Y" in the "Component Screened from Walkdown" field have a screening basis code in the "Screening Notes" column. Checked the entry for SQN-1-FCV-067-0081-A, basis code is now NSC; same for the -B valve. Per discussion with TVA PRA staff, in some cases the missing basis field entries needed to be created to resolve this finding, i.e., some of the bases were missing at the time of the peer review. For this closure review, several basis entries for components with "Y" in the "Screened" field were spot-checked to confirm that the basis entries made sense. No apparent inconsistencies were identified in this sampling. SUGGESTION: A number of components (e.g., related to ice baskets and some other components) are in the table with ROB (rule of the box) screening codes, but no Comment to indicate the relevant major-component (pages 258-262, 427, 437, 438 of MDN0009992017000047); it can be inferred from the component tag numbers what the relevant major component is, but this could be clarified in the notebook. SUGGESTION: Two sets of discrepancies were noted in Appendix H, which appear to be database printout/format issues affecting how several entries appear in the table: on pages 476-477 there are 2 or 3 components with no entries in the Screening Notes or Source; similar on pages 482-483. These were discussed with TVA SPRA staff who indicated the database would be checked and corrected. Both items noted above are editorial in nature, so based on this review, this finding is RESOLVED.	Met
3	SFR-E3	Not Met	8-5	The basis for SEL screening of components, when using the walkdown screening code, is not clear. Originated from SR SFR-E3.	The walkdown screening code is used in the SEL to indicate that certain components can be screened from further analysis based on walkdown insights. A spot check of components that use the walkdown screening code identified two components, SQN-1-HDR-067-0001A and SQN-2-FSV-068-0395, that, in turn, were screened from the fragility walkdown using Note 2 from TVAESQN010-REPT-003 Appendix A. Note 2 says, 'Screened out by PRA analysts'. Because of the circular reference, it is not clear whether the components were screened based on the walkdown or for other reasons.	Several components in the SQN were defined as screened from fragility walkdown analysis by PRA analysts. The walkdown code was used as a screening basis which implies the component was walked down, creating some circular references. These components were re-analyzed and screened using appropriate screening codes. All components were justified as screened requiring no additional re-analysis [“SQN Seismic PRA Seismic Equipment List”, MDN0009992017000047, Revision 1].	“SQN Seismic PRA Seismic Equipment List”, MDN0009992017000047, Revision 1	Maintenance	The re-assessment of the screening bases for the components in question is within the originally peer reviewed methodology and is an update to the screening results to correct the noted inconsistencies.	RESOLVED: Reviewed TVAESQN010-REPT-003 Appendix A Rev 1. Only references to PRA are for components screened post-walkdown in reference to "Further screening may be performed by PRA systems analysts." Spot check of screening codes showed no WD/WB entries for which any other reference is made to screening by PRA analysts.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
3	SPR-B1	Met	8-6	For the SQN SPRA, additional systems analysis was performed to add the FLEX 480V and 6.9KV equipment. Appendix A of MDN0009992017000044, which documents the FLEX systems analysis of, was reviewed. Some of the Part 2 SY requirements are not met. For example, not all required components appear to have been included (SY A-14, e.g., failure of load breaker to close on demand) and assessment of pre-initiators (SY-A16). For FLEX recovery actions, additional recovery actions, following successful EDG start, would be necessary and are not modeled. For example, SI pump start and/or MDP AFW pump start would be needed once power is restored. It is also not clear whether the systems logic model requires successful secondary heat removal prior to crediting FLEX EDG recovery actions. Originated from SR SPR-B1: Met; Finding.	Systems logic modeling does not SATISFY some of the Part 2 requirements for systems analysis (e.g., SY-A16, SY-A14).	The issue with PRA Standard Part 2 SY requirements has been addressed by creating a new PRA notebook for the FLEX diesel system [MDN0003602018000089]. See SQN-0-18-121, Section 2.1.5. for a discussion of the fault tree changes made to the FLEX diesel fault tree modeling to address the remainder of the F&O. The fault tree modeling has been updated to include the assumption that turbine-driven auxiliary feedwater pump success is necessary for the action of aligning the 6.9kV FLEX diesel to be credited in the fault tree. This is because if secondary heat removal by the TDAFW pump fails there will be insufficient time to align the FLEX diesels for them to be successful in their credited function of providing backup power to the 6.9kV shutdown boards. Figure 26 through Figure 27of SQN-0-18-121 shows this modeling for Unit 2. The modeling for Unit 1 is similar other than for minor nomenclature differences. The 6.9kV FLEX diesel logic was modified such that failure of adequate ventilation fails the FLEX diesels. Figure 28 and Figure 29 of SQN-0-18-121 show the new logic. Logic was added to the gates where the 480V FLEX diesel provides backup power to the vital battery boards. If the vital battery charger fails or certain breaker or transfer switch failures occur, the FLEX diesel will be unable supply power to the vital battery boards. The revised logic is shown in Figure 30 through Figure 33 of SQN-0-18-121.	PRA Evaluation SQN-0-18-121, "SQN SPRA F&O Resolutions Summary "MDN0003602018000089 "SQN PRA - FLEX DIESEL GENERATORS"; Model Files.	Maintenance	The wording of this Finding is vague and says only that "some of the Part 2 requirements are not met" and gives as examples only SY-A16 and SY-A14. The peer reviewed model included FLEX DG modeling, and the peer review graded SPR-B1 as Met, so the judgment for the closure review is that the peer review adequately considered the FLEX modeling such that the relatively narrow scope of changes made to address this finding represent PRA maintenance.	RESOLVED: Reviewed SQN-0-18-121 Sec 2.1.5 and associated portions of the FT model. Noted the new FT logic related to failure of FLEX DG and failure of recovery actions if TDAFWP fails or ventilation fails (items mentioned in the Finding). Reviewed portions of the HRA notebook MDN0009992017000043 Rev 1 for basis for operator actions and timing. These reflect operator input to the evaluations for the modeled scenarios, and detailed evaluations of post-initiator HFEs and HEPs. Reviewed MDN0003602018000089 for details of FLEX DG modeling. FLEX DG Load Breakers (mentioned in the Finding) were not found in the FT model but the FLEX notebook states that output breakers are considered to be within the DG component boundary. This is consistent with the DG component boundaries defined in NUREG/CR-6928. The fault tree model correctly includes failure of bus load shed breakers to open as leading to failure of FLEX power to the bus, consistent with the modeling for the Emergency DGs. Discussion of the load shed breaker modeling is not specifically described in the FLEX DG NB or in SQN-0-18-121, but the treatment in the model is appropriate and consistent with the modeling for similar breakers for other DGs in the internal events PRA model. The FLEX DG Notebook describes the use of a remote-control device needed for operation of the FLEX diesels and breakers. This is not included in the model and not further discussed, except that the FLEX DG component boundary is described as including output breakers, lube oil, and controls. The Seismic HRA notebook, MDN0009992017000043 Rev 1, Assumption 7 notes that for local actions in which the location is determined to be feasible for Seismic HRA, pre-staged tools and equipment are assumed available for EPRI Bins 1-3. Per discussion during the review with an SRO-qualified TVA staff member familiar with FLEX implementation, equipment like the remote-control device and other equipment needed to perform FLEX implementation actions are pre-staged, maintained in controlled cabinets, and checked routinely, which is consistent with the modeling treatment per Assumption 7. It was not immediately clear from information in the Data Notebook MDN-000-000-2010-0202 whether the component and data analysis boundaries are consistent for the FLEX DGs, which are noted in the FLEX DG notebook as being modeled, for data analysis purposes, as equivalent to normal EDGs. A check of DG component boundaries used in NUREG/CR-6928 indicates that the assumed FLEX DG boundaries are consistent with the generic data. FLEX DG maintenance unavailability is not in the model although maintenance is mentioned in the FLEX DG NB. Section 5.9 (Maintenance and Testing) of the FLEX DG NB provides the basis for exclusion, indicating that expected maintenance unavailability for the 6900V DGs is on the order of 4E-4/year based on approximately 4 hours per year, and noting that this is lower than the failure rate of the DGs. Table 5.2B of the FLEX DG NB shows the DG failure rate to be 9.27E-4/hr over 24 hr, which would be 2.2E-2. There is also an 8.8E-3/demand failure to start probability for the DGs, so the total start and run failure probability is on the order of 3E-2. The relevant PRA Standard criterion is in SR SY-A15: "(b) One or more failure modes for a component may be excluded from the systems model if the contribution of them to the total failure rate or probability is less than 1% of the total failure rate or probability for that component, when their effects on system operation are the same." The maintenance failure mode contribution is technically just above this criterion, but close enough that the additional justification provided in Section 5.9 of the FLEX DG NB is reasonable, i.e., "FLEX DG is only required in a small	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
										portion of accidents", TVA staff further noted that other contributors to failure of the FLEX diesels include the human error probabilities of failure to align and operate the diesels, which are also in the range of E-2, again roughly two orders of magnitude above that of the maintenance unavailability, so exclusion of the maintenance failure mode will not significantly impact the results or insights, and the criterion in SR SY-A15 is adequately addressed. There is an assumption in the FLEX DG notebook that although refueling is required within 10 hours, the operator action HEP can be ignored. The justification provided in the notebook does not reconcile the FLEX DG mission time with FLEX DG start time and initial fuel capacity other than to claim that it is unlikely that refueling will be overlooked. Per discussion during the review with an SRO-qualified TVA staff member familiar with FLEX implementation, and additional clarification provided by TVA staff, the fuel consumption time for the day tank are 13 hours and 24 hours at rated load and at 50% rated load, respectively. The TVA SRO noted that operators will establish fuel oil makeup to the fuel oil day tank as soon as possible after the 6.9kV DG startup is achieved, per procedure FSI 5.02. Given the initial capacity of the FLEX DG day tank, the operators should have ample time to establish the alignment of the makeup fuel to the day tank. The steps in FSI 5.02 include the alignment of the pump and the proper positioning of the hose. Once the makeup alignment is established and the control is set for auto operation, the day tank level will be automatically controlled between 40% and 90% level. Given the procedural basis and the long available time, exclusion of explicit HEP consideration is reasonable. The F&O also mentions an issue of failing to model starting of pumps after FLEX power is restored. A check of the sequence logic in the FT was made, looking at the SI modeling in several LOCA and Transient sequences. For LOCA sequences, SI pump fail to start (auto and manual) is included in the logic, including following loss of offsite power. This is technically not a restart after restoration of power given that the pump initially started on an SI signal prior to loss of power. However, it is consistent with the LOSP modeling in the SQN internal events model and consistent with the SQN SPRA assumption that offsite power is not restored, so the only significant pump start event would be following restoration of power via FLEX. SUGGESTION: Update the FLEX DG notebook to address the fuel tank alignment procedural guidance and provide additional detail regarding the basis for the incremental actions not being significant. SUGGESTION: Provide additional detail regarding the numerical basis for FLEX DG maintenance exclusion relative to the SR SY-A15 criterion b. SUGGESTION: The peer review concerns stated in the finding are interpreted as being narrowly focused on the FLEX modeling relative to the noted SY-A SRs, because no other issues were cited by the peer review team. Unfortunately, the original peer review team noted that some Part 2 SY requirements were not met and stated "ensure SY requirements from part 2 are met", This could be viewed as implying that a more systematic review of the revised models and documentation be performed relative to the Part 2 SY requirements. It is suggested that, if this has not already been done for the internal events PRA, TVA document a self-assessment of the FLEX DG system modeling against the full set of SY HLR and SR to establish additional confidence that there are no Part 2 issues.	

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
3	SPR-F1	CC I/II	8-7	New HFEs were developed specifically for the SQN SPRA (e.g., 6.9KV lockout reset, FLEX recovery actions). The detailed HFE assessment was not included in the HRA report MDN0009992017000048 but was provided to the review team from the HRA database. Originated from SR SFR-F1: Met; Finding.	Documentation of FLEX recovery actions is missing from MDN0009992017000048.	The relay lockout reset actions were missing from the SQN SHRA calculation (MDN0009992017000043), as well as the local action operator walkdown pathway documentation. This information was added to the applicable appendices in the SQN SHRA calculation. Note the F&O should refer to document MDN0009992017000043 rather than MDN0009992017000048.	“SQN Seismic PRA Human Reliability Analysis”, MDN0009992017000043, Revision 1.	Maintenance	Documentation of the missing HFEs is not a methodology change and had no impact on the results.	RESOLVED: MDN0009992017000043, Revision 1, PDF Pages 19 through 30 (Section 4.5), PDF Page 51 (Table 7-1), PDF Page 53 (Section 8.2), PDF Pages 66 through 67 (Table 9-1), PDF Pages 122-875 (HRA Calculator Sheets) were reviewed to confirm that the seismic recovery actions (including relay lockout reset actions) were addressed in the SPRA HRA. A review of recovery file "HEP-DEP-12-19-18" and MDN0009992017000045, Revision 1, PDF Pages 166-177) indicates that the lockout actions were included in the HRA dependency analysis.	CC I/II
3	SPR-A5	Met	8-8	For ISLOCA modeling, the SQN SPRA model replaces the pressure-induced rupture basic events (1(2)RUP*) with the fragility value for ISLOCA. Instead of replacing these basic events, the fragility failure mode should be added. Originated from SR SFR-A5: Met; Finding.	In Section 6.3.1.2 of MDN0009992017000044 , for the seismic induced ISLOCA, it is discussed that the difference between IE ISLOCA and S-ISLOCA is based on the piping rupture fragility modeled through basic events 1(2)RUP* The documentation states: The SQN IEPRA ISLOCA scenarios typically involve exposure of low pressure piping outside containment to reactor coolant system high pressure if isolation valves fail open. The SPRA ISLOCA events are different in that piping ruptures are seismically induced, such that either low- or high-pressure piping can be involved. This is reflected in the seismically induced rupture basic events (1(2)RUP*) that are replaced by FRANX during the quantification process. Except for the seismically induced rupture basic events, the ISLOCA failure logic transfers to the same logic gates as in the IEPRA. Sequence SEIS-009 transfers to the IEPRA ISLOCA event tree.	As suggested, the ISLOCA modelling was modified so that both the seismically induced pipe rupture probability and the normal random failure pipe rupture probability was included in the seismic model. Figure 14 through Figure 22 of SQN-0-18-121 show the revised modeling for Unit 2. The Unit 1 fault tree is similar except for minor nomenclature differences.	PRA Evaluation SQN-0-18-121, "SQN SPRA F&O Resolutions Summary"; “SQN Seismic PRA Human Reliability Analysis”, MDN0009992017000043, Revision 1.	Maintenance	The model change to include both random and seismic contributors to ISLOCA is not a methodology change and is an update to logic that provides improved basis for the expected seismic response.	RESOLVED: The resolution described in SQN-0-18-121 was reviewed, as was the ISLOCA sensitivity for the change described in SQN-0-18-161. In addition, the fault tree logic and quantification and recovery file were checked to determine that the changes have been appropriately incorporated. The resolution of the comment involved introducing a seismic level-specific piping seismic survival probability to correct the Boolean logic to account for the high seismic failure probabilities at the higher g levels. This process, and the values assigned, are in SQN-0-18-161, which also presents the results of a sensitivity to show that there is very little risk significance to seismic ISLOCA. Based on this review, the modified ISLOCA treatment resolves the finding.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
3	SPR-B4b	Met	8-9	For the lockout reset recovery actions, OP-LOCKOUT_EDG_S and OP-LOCKOUT_69kSDB_S, no procedural guidance for seismically induced relay reset is available. This lack of procedural guidance calls into question of operator action feasibility. Originated from SR SPR-B4b: Met; Finding.	Operator recovery actions, that were included in the SPRA for relay chatter related lockout of the EDGs and vital buses, are not based on earthquake specific procedural actions. Question Number 31 in MDN0009992017000048 for OP-LOCKOUT_EDG_S is: 'Does the crew believe that the instructions presented are appropriate to the situation (even despite any potential adverse consequences)? Do they have confidence in the effectiveness of the procedure for dealing with the current situation? In practice, this may come down to: have they tried it in the simulator and found that it worked?' and was answered with, 'No', based on the lack of procedural guidance on earthquake-induced spurious relay actuation'. Elsewhere in the interview documentation, a concern is raised regarding potential electrical damage. The operators felt that recovery may be delayed by up to 30 minutes due to these types of concerns.	Based on a condition report (CR 1323112), written during the model development process, guidance was added to procedure ECA-0.0 (Loss of All AC Power, Step 15) to address the recovery action OP-LOCKOUT_EDG_S. The HEP was re-evaluated. Based on a condition report (CR 1424214), the model development process, guidance was added to procedures ECA-0.0, AOP-N.05 (Appendix A), AOP-P.05 (Step 14), AOP-P.06 (step 14), and EPM-3-ECA-0.0 (Step 15) to address the recovery action OP-LOCKOUT_EDG_S. The HEP was reviewed based on the procedure revisions and was found to be adequate as is.	PRA Evaluation SQN-0-18-121, "SQN SPRA F&O Resolutions Summary"; "SQN Seismic PRA Human Reliability Analysis", MDN0009992017000043, Revision 1.	Maintenance	The re-evaluation of the HEP was done within the existing peer reviewed methodology, using new procedural guidance added to the EOPs, AOPs, and related procedures.	RESOLVED: MDN0009992017000043, Revision 1, PDF Pages 19 through 30 (Section 4.5), PDF Page 51 (Table 7-1), PDF Page 53 (Section 8.2), PDF Pages 66 through 67 (Table 9-1), PDF Pages 122-875 (HRA Calculator Sheets) were reviewed to confirm that the seismic recovery actions (including relay lockout reset actions) were addressed in the SPRA HRA. A review of recovery file "HEP-DEP-12-19-18" and MDN0009992017000045, Revision 1, PDF Pages 166-177) indicates that the lockout actions were included in the HRA dependency analysis. The HRA Calculator Sheets in PDF Pages 122-875, contain the procedural basis (with updated procedures based on cited CRs) for quantifying the failure probability of relay lockout reset actions.	Met
3	SPR-B6	Met	8-10	Accessibility to FLEX EDG rooms was not fully assessed. Specifically, the ex-control room block wall impact is not mapped to the HAESBO3MW_S FLEX action. In addition, the potential that a storage cabinet in the ADG Building may tip over and may block access was not assessed. Originated from SR SPR-B6: Met; Finding.	In the SQN SPRA, all ex-control room actions, except for the FLEX EDG action (HAESBO3MW_S), are impacted by the SEIS_BLOCKWALL failure. The FLEX EDG action is instead impacted by the ADG building failure (SEIS_ADGB). Although the action actually takes place in the ADGB building, operators would be required to go through the control room to obtain the relevant procedure. Because operators must travel to the control room prior to the ADGB, the potential for aux/control building block wall impact should be assessed. In addition, during the walkdown performed for the peer review, a review team member identified a storage cabinet located along the access pathway to the FLEX diesel that has the potential to obstruct access to the FLEX diesel room.	The block wall impact has been mapped to HFE HAESBO3MW_S FLEX. Figure 55 of SQN-0-18-121 shows the FRANX table (FireInitiatorHRA) where this mapping was performed.	PRA Evaluation SQN-0-18-121, "SQN SPRA F&O Resolutions Summary"; Model Files Pg 49 of 0-18-121 has the justification for excluding the Cabinet.]	Maintenance	The model change to address the missing impacts was done within the existing peer reviewed methodology.	RESOLVED: Section 2.2.3 of SQN-0-18-121 states that the block wall impact has been mapped to HFE HAESBO3MW_S FLEX and provides a clip of the FRANX file mapping the HFE to the block wall failure, and this was confirmed to be in the FRANX file FireInitiatorHRA. A review of the FRANX files confirms that the additional mapping of the block wall failure to failure of the FLEX diesel has been added to the model. Review of FRANX cutsets involving FLEX diesel failure indicates that the model also includes a 0.5 factor on block wall failure impact. This is in accordance with the overall approach taken in the SPRA for block wall failure impacts, as stated in Section 6.5.8 of MDN0009992017000044 Rev 1, SQN Seismic PRA Methodology, Inputs, and Model Notebook. So, this particular block wall impact has been incorporated consistent with the modeling of other block wall impacts. Pg 49 of SQN-0-18-121 provides the justification for excluding the impact of the noted Cabinet failure, based on assessment demonstrating there is an alternate pathway for the operators if the cabinet fails. The issues identified in this Finding have been resolved.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
3	SPR-E2	Met	8-11	Adequate justification for convergence is not documented. Originated from SR SPR-E2: Met; Finding.	MDN0009992017000045 Revision 0 was reviewed as the basis for this assessment Three criteria for determining adequate convergence are given in section 7.1: 1. The CCDP reached the 0.908 (the Unit 1 plant availability factor) or 0.932 (the Unit 2 plant availability factor) upper limit. 2. The change in total SCDF per decade truncation limit change was = 5%. 3. Quantification at a lower truncation was not possible due to either FRANX, ACUBE, or FTREX memory limitation issues. Criteria 1 and 2 are reasonable however, in cases where quantification at lower truncation levels was not possible, further justification of convergence should be provided.	PRA Evaluation SQN-0-19-003 shows that SCDF and SLERF converge for both Unit 1 and Unit 2.	PRA Evaluation SQN-0-19-003, SQN SPRA Convergence	Maintenance	The resolution is a clarification based on the original peer reviewed methodology.	Initial Disposition PARTIALLY RESOLVED: Review of SQN-0-19-003 shows that an extensive convergence assessment was performed to address the issue in the finding that additional justification is needed to demonstrate that the model converges. SQN-0-19-003, Section 2.2, describes the process undertaken to demonstrate convergence for each seismic hazard interval. Because the model could not be successfully run at truncations low enough to demonstrate a 5% per decade convergence, the approach to demonstrating that convergence is possible involved making adjustments to the model, including setting some alignment flags to TRUE (as opposed to having a probability of 1.0) and guaranteeing failure of fragility groups with failure probabilities greater than 0.7 for Seismic Bins %G04 through %G08. This results in inability to directly compare the results to the baseline model but was necessary to address the finding. Section 2.2.4 of SQN-0-19-003 presents and discusses the results of the convergence calculations using the adjusted model. A basis is provided for establishing that the adjusted model is converged (meeting the criterion in the PRA Standard) at 1E-11 for Unit 1 CDF and LERF and Unit 2 LERF, and at 1E-10 for Unit 2 CDF. The suggested resolution in the Finding is as follows: Where convergence is not demonstrated using criteria 1 or 2, show that convergence is reached by either: 1. For each seismic bin assess the remaining hazard frequency. If the remaining frequency is small enough, convergence may be justified 2. Quantify individual sequences separately to attain convergence 3. Use some other technically defensible approach to justify convergence. The approach taken in SQN-0-19-003 is the 3rd approach. TVA PRA staff explained that approach 2 was not productive since most of the sequences are seismically induced loss of offsite power and the same quantification limits occur. The approach used is judged to be a technically defensible approach and at least partially addresses the finding. Since the model used to demonstrate convergence differs from the baseline model (via the adjustments described above) the conclusion that the convergence sensitivity is directly applicable to the baseline model is indirect, i.e., it doesn't directly demonstrate convergence of the results at the same truncation levels in the baseline model (as shown in Table 8-1 of the Quantification Notebook MDN0009992017000045 Rev 1), which are significantly higher for seismic intervals 5 through 7. So, to completely close this finding, the following should be addressed: 1. Is the overall impact on CDF/LERF using the selected truncation level(s) for the base SPRA model vs the converged truncation levels sufficiently small; and 2. Is there confidence that there aren't any new/previously unreviewed accident sequences at the lower truncations, particularly for seismic intervals 5, 6, and 7? Addressing these questions would make the assessment in SQN-0-19-003 worthy of a best practice relative to SPR-E2. Final Disposition RESOLVED: On April 1, 2019, TVA provided SQN-0-19-003, Revision 1. Section 2.2 was reviewed to confirm the acceptability of the methods used to facilitate model convergence. Additional bases have been provided to justify that the model changes made to facilitate convergence are valid and have minimal	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
										impact on the results and contributors. Section 2.2.4 was reviewed to confirm that that the model could be successfully run at truncations low enough to demonstrate a 5% per decade convergence for the overall model results (i.e., summing over all seismic bins). There are several bins where the 5% criterion is not met, e.g., bins 1 – 3, which are not significant to the CDF and LERF results; and bin 6, which is more significant but is still shown to be trending correctly. The convergence criterion is met overall, i.e., the total seismic CDF and total seismic LERF meet the convergence criterion. Given the software limitations, the revised approach adequately demonstrates model convergence. Section 3 of SQN-0-19-003, Revision 1 further states that the model used in the convergence study will become the new base model, with the lower truncation levels used in the truncation sensitivity.	
3	SPR-F1	Met	8-12	Documentation of HRA Bin 3 upper bound acceleration. Originated from SR SPR-F1: Met; Finding.	Page 20 of MDN0009992017000048 states that the EPRI bin 3 upper bound is 0.8g, "The HCLPF of the ceiling panels exceeds the EPRI Bin 3 upper bound of 0.8 g, above which all HFEs are assumed unfeasible due to failure of instrumentation". Section 4.6 of the same document states that the boundary between bin 3 and 4 is 1.3g - "For SQN, an acceleration of 1.3g was chosen as the boundary between S3 and S4.". In addition, table 4-4 states that SQN uses 1.5g as the boundary between bin 3 and 4. Based on the value of the lower bound for the %G08 bin, the boundary should be 1.5g. In addition, the EPRI criteria for the bin 3 upper bound is stated in table 4-4 but the plant specific basis for use of 1.5g is not included (SQN staff was aware of the basis for this value and communicated it to the PR team).	The entry in the SQN SHRA calculation [MDN0009992017000043] stating the bin 3 and 4 boundary value is 1.3g was a typographical error. The correct value of 1.5g was applied in the damage state bin table. A note has also been added to the tables in the HRA SQN SHRA calculation to explain the justification for the 1.5g break point between EPRI bins 3 and 4]. Note the F&O should refer to document MDN0009992017000043 rather than MDN0009992017000048.	PRA Evaluation SQN-0-18-121, "SQN SPRA F&O Resolutions Summary"; "SQN Seismic PRA Human Reliability Analysis", MDN0009992017000043, Revision 1.	Maintenance	The resolution is a clarification based on the original peer reviewed methodology.	RESOLVED: MDN0009992017000043, PDF Pages 20 and 21 (Table 4-4), were reviewed to conform that an acceleration of 1.5g was chosen as the boundary between seismic bins S3 and S4 throughout the document.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
3	SPR-F1	CC I/II	8-13	Incorrect piping diameter used as upper bound definition for very small LOCA. Originated from SR SPR-F1: Met; Finding.	Table 6-6 of MDN0009992017000044 states that the very small LOCA upper bound break size is 3/4" in equivalent diameter. The SQN internal events success criteria notebook (MDN-000-000-2010-0207) identifies an equivalent break diameter of 3/8" as the upper bound for the internal events version of very small LOCA.	The SQN SPRA system notebooks [SQN Probabilistic Risk Assessment- Seismic PRA Methodology, Inputs, and Model Notebook". MDN0009992017000044] incorrectly stated that the maximum diameter for a Very Small LOCA (SLOCAV) was ¾". The correct value should be 3/8" and this has been corrected.	MDN0009992017000044, SQN Probabilistic Risk Assessment- Seismic PRA Methodology, Inputs, and Model Notebook	Maintenance	Documentation of the very small LOCA size is not a result of a methodology change and had no impact on the results.	RESOLVED: MDN0009992017000044, PDF Page 46 (Table 6-6), was reviewed to confirm that the maximum diameter for a Very Small LOCA is less than 3/8 inches.	CC I/II
3	SPR-F2	Met	9-1	While the Seismic Equipment List INCLUDES structures and passive components not included in the internal-events model, it does not appear that equipment that may have been screened on the SY-A15 criteria were systematically re-evaluated for seismic considerations. Originated from SR SPR-F2: Met; Finding.	While some of the system notebooks include listings of ASME screened components that did not appear to be systematically included back into the SEL for seismic evaluation. Typical system notebooks (MDN-000-003-2010-211 and 212) but not all have lists of screened equipment in Appendix F which may have resulted in components not being included in the SEL.	Systematic mapping of components in the SEL was done by outlining on P&IDs and electric drawings those components included in the SEL boundary. This mapping was done by system and documented in a filekeeper file for traceability. The information is described in section 2.3 of MDN0009992017000047.	MDN0009992017000047, SQN Seismic PRA Seismic Equipment List	Maintenance	Documentation of the systematic SPRA SEL development process is not a result of a methodology change and had no impact on the results.	RESOLVED: MDN0009992017000047 Revisions 1, PDF Pages 560 through 570, were reviewed to confirm that Systematic mapping of components in the SEL was done by outlining on P&IDs and electric drawings those components included in the SEL boundary. Sections 2.3 and 2.4 addresses the systematic approach used to develop the SEL, including equipment excluded from the internal events PRA. The table in PDF pages 27 through 39 provide the system screening process. Appendix H provides a listing of the final SPRA SEL, including the comments in the internal events PRA and components from other sources.	Met
3	SPR-A1	Met	9-2	Seismic group SEIS_23-2 results in multiple initiator groups being concurrently failed. This could result in the large, medium, and small LOCA sequences all being binned as excessive LOCA. This could result in these sequences being missed. Associated with SPR-C1 and SPR-A1. Originated from SR SPR-A1: Met; Finding.	The mapping of some seismic groups to failed components span multiple initiator groups. In the case of SEIS_23-2, this could result in only excessive LOCAs being included as a failure due to removal of nonminimal cutsets for the other LOCA classes. This could impact LERF results as excessive LOCAs would bin into the less challenging bin for RCS pressure low at core damage.	See SQN-0-18-121 R0 for the details on this response. A sensitivity case shows that the impact of splitting the various categories of LOCA into unique fragility groups has a negligible impact or CDF and LERF (see SQN-0-18-061, case #2). Even though the impact was minimal, separate fragility groups were created for the various types of LOCA (LLOCA, MLOCA, SLOCA, VSLOCA) rather than grouping them into the same fragility group. Each of the fragility groups have the same Am of 2.74 (Figure 50 of SQN-0-18-061), except for VSLOCA, which has an Am of 2.04.	PRA Evaluation SQN-0-18-061, case #2,	Maintenance	The model changes to create separate fragility groups for each LOCA size were not a result of a methodology change and is an update to logic that provides improved basis for seismic-induced initiating events.	RESOLVED: SQN-0-18-061, PDF pages 5 through 10 were reviewed to confirm that hat the effects of combining the resolution to F&O 9-2 and assuming that the small LOCA partitioning was FALSE (1(2)INMTLINE-PART1_S) and the Very Small LOCA partitioning was TRUE (1(2)INMTLINE-PART2_S) had very small impacts on CDF and LERF. SQN-0-18-121, PDF pages 46 through 48 were reviewed to confirm the replacement of split fractions with unique fragility groups for each LOCA size. Even though the impact was minimal, separate fragility groups were created for the various types of LOCA (LLOCA, MLOCA, SLOCA, VSLOCA) rather than grouping them into the same fragility group. Each of the fragility groups have the same Am of 2.74, except for VSLOCA, which has an Am of 2.04.	Met
3	SPR-D1	Met	9-3	FLEX equipment is not listed in the SEL. Refer to SPR-C1 and 7-6, 8-6, and 8-10. Originated from SR SPR-D1: Met; Finding.	FLEX equipment is not listed in the SEL. This could result in missing components that need to be evaluated for fragility.	The FLEX diesels and support components have been added to the SEL [MDN0009992017000047], based on logic model inputs presented in the FLEX System notebook [MDN0003602018000089].	MDN0009992017000047, SQN Seismic PRA Seismic Equipment List; MDN0003602018000089 "SQN PRA - FLEX DIESEL GENERATORS"; Model Files	0	Documentation of the FLEX equipment in the SPRA SEL is not a result of a methodology change and had no impact on the results.	RESOLVED: There is an assumption that although refueling is required within 10 hours, the operator action HEP can be ignored. The justification provided does not reconcile the FLEX DG mission time with FLEX DG start time and initial fuel capacity other than to claim that it is unlikely that refueling will be overlooked. However, resolution of Finding 8-6 provided the basis for the fuel capacity assumption.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
3	SPR-F3	CC II/III	9-4	The evaluation of the internal event model assumptions and sources of uncertainty for applicability to the seismic PRA was part of the documentation package and appeared to be performed after the review started. Originated from SR SPR-F3: Met; Finding.	While Section 5.1 of MDN0009992017000044 provides several assumptions that were assumed for the seismic PRA, the underlying assumptions and sources of uncertainty were not evaluated for inclusion to the list.	PRA evaluation SQN-0-18-058 documents a thorough review of the assumptions made in the internal events notebooks for their applicability to the seismic PRA. The content of this PRA evaluation has been added as an appendix to the Methodology, Inputs and Model notebook [“SQN Probabilistic Risk Assessment- Seismic PRA Methodology, Inputs, and Model Notebook”. MDN0009992017000044, Revision 1].	PRA evaluation SQN-0-18-058, Response to Peer Review Question AM-03"; MDN0009992017000044, Revision 1, “SQN Probabilistic Risk Assessment- Seismic PRA Methodology, Inputs, and Model Notebook”	0	Documentation of the evaluation of the internal event model assumptions and sources of uncertainty for applicability to the seismic PRA is not a result of a methodology change and had no impact on the results.	RESOLVED: SQN-0-18-058 Revision 1, PDF pages 3 through 54 were reviewed to confirm the documentation of the evaluation of the internal event model assumptions and sources of uncertainty for applicability to the seismic PRA. It would be useful to demonstrate that a reasonable HEP for this action would not contribute significantly to FLEX DG failure	CC II/III
3	SPR-F1	Met	9-5	The relationship between the SPRA inputs and the FRANX database tables is not documented. Originated from SR SPR-F1: Met; Finding.	The relationship between the SPRA inputs and the FRANX database Tables is not documented.	PRA evaluation SQN-0-18-058 documents the basis for the fragility mapping and FRANX data table relationship used in the SQN SPRA model. The content of this PRA evaluation has been added as an appendix to the Methodology, Inputs and Model notebook [MDN0009992017000044].	PRA evaluation SQN-0-18-058; MDN0009992017000044, Methodology, Inputs and Model notebook	Maintenance	Documentation of the relationship between the SPRA inputs and the FRANX database is not a result of a methodology change and had no impact on the results.	RESOLVED: SQN-0-18-058 Revision 1, PDF pages 54 through 208 (Table 5) were reviewed to confirm the documentation of relationship between the SPRA inputs and the FRANX database.	Met
3	SPR-C1	Met	20-2	The simplifying assumption of guaranteed failure of the Turbine Building is inconsistently modeled in the S-PRA and appears to be creating a distortion in the model, which is not currently justified. Originated from SR SPR-C1: Met; Finding.	The distortion does not appear to be significant to the risk results, however given the inconsistent treatment of Turbine Building failure given the current assumptions in the model (i.e., no credit is assumed for equipment located within the Turbine Building), this modeling approach should be reviewed to confirm that no significant distortions are present. Example of inconsistent treatment of Turbine Building failure: The S-PRA model assumes guaranteed failure of the Turbine Building but does not appropriately address closure of the MSIVs to prevent a steam line break for each seismic-induced initiating event. Given the assumed guaranteed failure of the Turbine Building, the S-PRA model should ensure that closure of the MSIVs to prevent a steam line break is addressed for each seismic-induced initiating event. Fragility group SEIS_13-1-4 triggers SSBO event tree sequences with failure of MSIV isolation and is included in the SIET logic under gate	As suggested in the possible resolution for F&O 20-2, the logic for MSIV closure was modified. A gate was added which results in direct core damage when an MSIV fails seismically along with a collapse of the turbine building. In addition, a new fragility for the turbine building called SEIS_BLD-TB was developed. Figure 24 and Figure 25 of SQN-0-18-121 show the modified logic for direct core damage upon seismic MSIV failure. For additional information on this F&O resolution, see SQN-0-18-061, case #3.	PRA evaluation SQN-0-18-058; MDN0009992017000044, Methodology, Inputs and Model notebook	Maintenance	The model changes to address seismic failure of MSIVs to close with failure of the Turbine Building, that lead to direct core damage, were not a result of a methodology change and had no impact on the results.	RESOLVED: SQN-0-18-121, PDF pages 23 through 26 (Figures 24 and 25) and the fault tree and results ??? were reviewed to confirm the model changes that address seismic failure of the MSIVS to close combined with the Turbine Building failure that result in direct core damage.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
					U1_S-SSBO . However, this fragility group only addresses correlated failure of all MSIVs to close. Partial closure of the MSIVs represented by fragility group SEIS_13-1-1 does not trigger the SSBO event tree and is not included in the SIET success logic. Additionally, the SIET success logic for the SSBO branch does not include independent failures of the MSIVs to close to prevent a steam line break.						
3	SPR-A2	Met	20-3	Seismic-induced Excessive LOCAs are missing from the SEIT. Originated from SR SPR-A2: Met; Finding.	Seismic-induced Excessive LOCAs, represented by fragility groups SEIS_23-5 (Reactor Vessel Supports), SEIS_23-4 (Pressurizer Supports), and SEIS_23-2 (NSSS piping) are modeled under gate U1_EXLOCA_S and represent direct to core damage scenarios. However, these failures are missing from the SIET success logic in the model (not included under U1_S-DCD) and are not discussed in Section 6.2.1 and 6.3.1 of MDN0009992017000044 Revision 0.	The original pre-peer review fault tree included logic developed for a Seismic Initiating Event Tree (SIET) where each seismically induced accident sequence was ordered from most severe (in terms of CDP) to least severe as pictured in Figure 1 of SQN-0-18-121. This was done according to guidance given in NUREG/CR-4840 in order to ensure that "…if one initiating event occurs, the occurrence of other initiating events further down the hierarchy [further right on the SIET] is of no significance in terms of the plant's response." [NUREG/CR-4840]. The peer team believed that this logic had no practical impact on the results of the seismic fault tree given the use of the ACUBE post-processing software. As a result of peer review F&O 20-6, the logic was modified to remove the SIET success branch logic. The resulting modified seismic initiator branches are presented in Figure 2 through Figure 11 of SQN-0-18-121. The figures showing the original initiator branches can be found in SQN-0-18-061, case #4. The figures show logic for the Unit 2 model, but the Unit 1 model is identical except for minor nomenclature differences. These changes had no noticeable effect on the cutset results [SQN-0-18-128].	PRA Evaluation SQN-0-18-121; PRA Evaluation SQN-0-18-061; Model Files	Maintenance	The model changes to remove the complementary success logic were not a result of a methodology change and had no impact on the results.	RESOLVED: The resolution for F&O 20-6 removed the requirement for complementary success logic, including the logic for seismic-induced failure of Reactor Pressure Vessel and Pressurizer supports that lead to direct core damage. However, the failure contribution of these items remains under gate U1_EXLOCA_S_NSSS for the Unit 1 model. The same is true for the Unit 2 model.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
						The resolution of 20-6 makes F&O 20-3 unnecessary, since the SIET logic was removed from the fault tree. Excessive LOCAs are included in the LLOCA branch of the fault tree. For comparison purposes, a sensitivity study done as part of the Watts Bar SPRA peer review F&O resolution [WBN-0-16-074] for a similar issue confirmed that for Watts Bar, re-ordering this SIET logic in the fault trees had no discernable impact on the cutset results generated.					
3	SPR-F1	Met	20-4	The application of offsite power success term given assumed guaranteed failure of equipment in the Turbine Building for every seismic event, which always results in a general transient due to a seismic-induced flood scenario in the Turbine Building, is misleading. Originated from SR SPR-F1: Met; Finding.	Offsite power success is addressed for each non-LOSP general transient initiator. Given the assumed guaranteed failure of equipment in the Turbine Building for every seismic event, a general transient due to a flood scenario in the Turbine Building will always propagate. Application of the offsite power success term to these scenarios is misleading. This is not deemed to be a significant issue given the low contribution of the seismic hazard bins where offsite power is successful.	The seismically induced flood scenario in the Turbine Building was marked as a guaranteed failure in the FRANX model, which resulted in a general transient whenever a flooding event occurred (which was always since it was a guaranteed failure event). For the revised model this event is no longer considered a guaranteed failure, as shown in Figure 63 of SQN-0-18-121.	PRA Evaluation SQN-0-18-121, "SQN SPRA F&O Resolutions Summary"; Model Files	Maintenance	The FRANX database mapping change to preclude guaranteed failure of the Turbine Building, due to seismic-induced flood scenarios in the Turbine Building, do not result in a methodology change.	RESOLVED: SQN-0-18-121, PDF page 78 (Section 2.5.2, Figure 63) was reviewed to confirm the FRANX database mapping change to preclude guaranteed failure of the Turbine Building for seismic-induced flood scenario in the Turbine Building. Event U0_FLOODPIPETB_S is mapped to Fragility Group SEIS_BLD-TB in FRANX Files "SQN_Seismic_Rev0_U1CDF-Post Peer Review". Also, Event U0_FLOODPIPETB_S checked in FRANX file. SUGGESTED SOLUTION: Remove mapping of Event U0_FLOODPIPETB_S to Fragility Group SEIS_BLD-TB for low hazard levels in the FRANX Files. MDN0009992017000044, Pages 84 through 6 (Section 5.5.9) was reviewed to confirm the method used to address the availability of offsite power for transient sequences. An offsite power available probability is applied to all transient sequences containing flag event FLG_NONLOSP_INIT. This probability was calculated using the median seismic capacity (Am) of 0.3g. A review of the FRANX model indicates that the balance of plant equipment was mapped to fragility group SEIS_HLF ("high likelihood of failure" group, with Am of 0.9), which effectively translates to guaranteed failure. The turbine building fragility group (SEIS_BLD-TB) is assigned an Am of 3.3. It is mapped to balance of plant equipment that have been characterized as rugged (i.e., manual valves). Basic event U0_TBFAIL_S is ANDED with the failure to close the MSIVs which results in direct core damage (U1_S-TBMSLB_DCD). The offsite power available probability is not modeled as a recovery of offsite power but is merely the availability of offsite power (split fraction) based on the seismic fragility of offsite power. Regardless of power is availability, the balance of plant equipment (except for the rugged equipment discussed above) have been assigned a high likelihood of failure.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
3	SPR-F1	Met	20-5	The very small LOCA fragility (SEIS_0-16-2) is partitioned between the VSLOCA and SLOCA sequences with a 50% split fraction. Originated from SR SPR-F1: Met; Finding.	Per MDN0009992017000044, “One or more instrument lines are likely to leak rather than beak in an earthquake. Although the leakage rate is very small, the leakage of one instrument line has potential of very small LOCA, and the aggregate leakage of multiple lines should not exceed a small LOCA. In this model, the seismic failure modes of the instrument lines are small LOCA and very small LOCA and are modeled as SEIS_0-16-1 and SEIS_0-16-2.” The median capacities assigned to SEIS_0-16-1 and SEIS_0-16-2 are 3.91 and 2.62 respectively. From review of the model, the very small LOCA fragility (SEIS_0-16-2) is partitioned between the VSLOCA and SLOCA sequences with a 50% split fraction. This split fraction is not documented in MDN0009992017000044. Given that the fragility analysis developed separate fragilities for SLOCA and VSLOCA the technical basis for further partitioning the VSLOCA contribution is unclear.	The peer review model used fragility group SEIS_0-16-2 for the fragility of SLOCAV and SLOCA. A sensitivity case was performed (see Reference SQN-0-18-061 case #2) which showed that the effects of combining the resolution to F&O 9-2 and assuming that the small LOCA partitioning was FALSE (1(2)INMTLINE-PART1_S) and the Very Small LOCA partitioning was TRUE (1(2)INMTLINE-PART2_S) had very small impacts on CDF and LERF. The current model therefore applies this fragility to both SLOCAV and SLOCA but does not partition them on a 50/50 percentage basis (1(2)INMTLINE-PART1_S=1 and 1(2)INMTLINE-PART2_S=1 in the fault tree model). Although the overall effect is to double-count these initiators, the result is a conservative over-estimation that does not significantly affect the results.	PRA Evaluation SQN-0-18-121, "SQN SPRA F&O Resolutions Summary"; Model Files	Maintenance	The model changes to create separate fragility groups for each LOCA size were not a result of a methodology change and is an update to logic that provides improved basis for seismic-induced initiating events.	RESOLVED: SQN-0-18-061, PDF pages 5 through 10 were reviewed to confirm that hat the effects of combining the resolution to F&O 9-2 and assuming that the small LOCA partitioning was FALSE (1(2)INMTLINE-PART1_S) and the Very Small LOCA partitioning was TRUE (1(2)INMTLINE-PART2_S) had very small impacts on CDF and LERF. SQN-0-18-121, PDF pages 46 through 48 were reviewed to confirm the replacement of split fractions with unique fragility groups for each LOCA size. Even though the impact was minimal, separate fragility groups were created for the various types of LOCA (LLOCA, MLOCA, SLOCA, VSLOCA) rather than grouping them into the same fragility group. Each of the fragility groups have the same Am of 2.74, except for VSLOCA, which has an Am of 2.04. Reviewed fault tree and FRANX files to confirm model and documentation are consistent.	Met

Table A-1 SQN SPRA F&O Closure Review Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
3	SPR-B5	CC I/II	20-6	Application of the SIET logic in the S-PRA model is inconsistent with its intention, which is to INCLUDE the complementary 'success' state (i.e., the SIET is currently applied as delete term logic rather than complement to success). Originated from SR SPR-B5: Met; Finding.	From a review of the model, the SIET is currently being applied as delete term logic rather than success term logic. For example, the following gates located under the SIET AANB gates should be 'AND' gates rather than 'OR' gates to correctly apply complementary 'success' state in the cutsets: - U1_SEIS-001N - U1_SEIS-002N - U1_SEIS-003N - U1_SEIS-004N - U1_SEIS-005N - U1_SEIS-006N - U1_SEIS-007N - U1_SEIS-008N - U1_SEIS-009N The current modeling could inadvertently be removing cutsets from the S-PRA results if an initiator fragility group propagates through both sides of the AANB gate. Correct application of success term logic would instead result in the addition of the complementary 'success' states for the initiator fragility groups in the cutsets (i.e., apply -SEIS_XXX in the cutset which equates to 1 minus the failure probability).	The original pre-peer review fault tree included logic developed for a Seismic Initiating Event Tree (SIET) where each seismically induced accident sequence was ordered from most severe (in terms of CCDP) to least severe as pictured in Figure 1 of SQN-0-18-121. This was done according to guidance given in NUREG/CR-4840. The peer team believed that this logic had no practical impact on the results of the seismic fault tree given the use of the ACUBE post-processing software. The logic was modified to remove the SIET success branch logic. The resulting modified seismic initiator branches are presented in Figure 2 through Figure 11 of SQN-0-18-121. The figures showing the original initiator branches can be found in SQN-0-18-061, case #4. The figures show logic for the Unit 2 model, but the Unit 1 model is identical except for minor nomenclature differences. These changes had no noticeable effect on the cutset results [SQN-0-18-128].	PRA Evaluation SQN-0-18-121, "SQN SPRA F&O Resolutions Summary"; Model Files	Maintenance	The model changes to remove the complementary success logic were not a result of a methodology change and had no impact on the results.	RESOLVED: SQN-0-18-061, PDF pages 17 through 32 were reviewed to confirm the original logic for complementary success logic, and a sensitivity analysis to demonstrate that removal of the complementary success logic gates had insignificant impact on the results. SQN-0-18-121, PDF pages 2 through 14 (Section 2.1.1, Figures 1 through 11) were reviewed to confirm the removal of the complementary success logic in the updated model. SQN-0-18-128, PDF pages 24 through 27 (Tables 1 through 9) were reviewed to confirm the removal of the complementary success logic did not result in a change in the risk calculation. Reviewed fault tree and FRANX files to confirm model and documentation are consistent.	CC I/II

Table A-2 Focused Peer Review Findings Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
1	SHA-I2	MET	1-2 (Finding)	Analyses of the depth to bedrock from borings within the footprint of the DGB indicate systematic differences. These systematic differences were not modeled and are likely to impact estimates of the differential settlement of the DBG and the settlement-induced strain in piping connected to the DGB.	The depth to bedrock is an important parameter in calculations for liquefaction-induced vertical settlement because (1) increasing depth results in greater settlement for a given strain and (2) the site profile indicates that there are more settlement prone soils at depth. The depth to bedrock under the footprint of the DGB was modeled as random with a median value of 65 ft and a beta of 0.31. These parameters were used to generate random samples of the depth to bedrock in the Latin Hypercube sampling approach. Analyses requested by the IAT indicate that there are systematic differences in the depth to bedrock, with the depth increasing to the west in Borings 34, 42, and 43 (Figure 3.3). These systematic differences across the DGB footprint that may impact estimates of differential movement between the DGB and piping and thus the settlement-induced strain in the piping.	Additional analyses of liquefaction-induced vertical settlement were performed to evaluate the potential impacts of the greater depth to bedrock on the west side of the DGB on buried piping. The additional analysis considered two deformation modes, (1) the DGB moves down relative to the surrounding soil and (2) the surrounding soil moves down relative to the DGB. The additional analysis shows that the information in the current governs.	Response to IAT Question. "Updated Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant (SQN)', Revision 1, 4/1/2019, CJC-SQN-C-001, Carl J. Constantino and Associates.	Maintenance	No new technology	RESOLVED: Appendix P (Section P.1) of Carl J. Costantino and Associates Report CJC-SQN-C-001, Revision 1 (April 1, 2019), Updated Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant (SQN) provides deformation sensitivity results for two modes of deformation to model the thicker depth to bedrock under the western edge of the DGB. The sensitivity results demonstrate that vertical deformations and pipe strains are not increased as a result of the thicker soil under the western edge of the DGB. The sensitivity results also included a 2-D finite element assessment of an extended region of liquefaction settlement; two variations of the thicker soil were assessed. The 2-D finite element assessment resulted in deformation profiles that were compared to those from the deformation sensitivity results, supporting the conclusion that the set of pipe strains described in Section 6.10 of the remain report remain conservative. Suggestion: The discussion presented in Appendix P of CJC-SQN-C-001, Revision 1, could be better integrated and cross-referenced with the main report to more clearly explain the context of the sensitivity analysis performed to address the F&O. For example, the additional discussion could cross-reference the section from the main report which summarizes the deformation from liquefaction. The discussion of the building settlement caused by reduced support on the west side of DGB could more clearly explain how the total deformation was quantified (the equations are presented after the results which causes confusion).	MET
1	SHA-J2	MET	1-3 (Finding)	The clarity and completeness of the documentation found in Carl J. Costantino and Associates Report CJC-SQN-C-001, Revision 0 (January 16, 2019), Updated Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant, should be improved to address specific technical issues identified during the focused scope peer review of the report. (This F&O originated from SR SHA-J2)	The documentation must be enhanced or augmented in the following areas: 1. The liquefaction assessment of vertical settlement uses Latin Hypercube sampling to generate profiles of engineering properties for use in estimating the distribution of liquefaction-induced vertical settlement and lateral spreading. Peer reviewers requested that the analyst confirm that the sampling procedure yields similar distributions of vertical settlement and lateral spreading as the eight "calibration" borings (i.e., borings with reasonably complete information available, Figure 3.3 of CJCSQN- C-001 R0) within the footprint of the DGB. 2. The extent of liquefaction associated vertical settlement is estimated using methods for the soil dependent on whether the soil layer is considered as sand-	1. An evaluation is performed using eight of the borings in the general area of the DGB as "calibration" borings (herein referred to as the "individual borings". These evaluations show that the vertical settlements and lateral deformations developed from evaluations using the generalized profiles produces settlements and deformations comparable to those developed from the individual borings. Differences in the results from these two sets of borings are attributed to the lack of adequate representation of the range of boring depths and (N1)60 values (i.e. residual shear strength) seen in the DGB area for the individual borings that is seen in the full data set for the DGB site, which was used to develop the general borings.	Response to IAT Question. "Updated Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant (SQN)', Revision 1, 4/1/2019, CJC-SQN-C-001, Carl J. Constantino and Associates.	Maintenance	No new technology	RESOLVED: Appendix P (Section P.2) of Carl J. Costantino and Associates Report CJC-SQN-C-001, Revision 1 (April 1, 2019), Updated Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant (SQN) provides augmented documentation to address the items identified during the focused scope peer review. An assessment of the liquefaction deformation from the 'calibration' borings demonstrated resulting settlements are comparable to those resulting from the individual borings as assessed in the main body of CJC-SQN-C-001, Revision 1. An assessment of the relative difference between sand-like and clay-like settlements determined that liquefaction settlement is controlled by clay-like soils. The documentation clarified (1) the use of the AF limit when assessing liquefaction settlement, (2) how the composite shear-wave velocity profiles were developed, (3) the lack of liquefaction settlement sensitivity to the possible correlation between Sa and rd, and (4) the lack of sensitivity to shear stresses considering the use of individual base case soil profiles versus the use of a composite soil profile as was used in the main body of CJC-SQN-C-001, Revision 1. Suggestion: The discussion presented in Appendix P of CJC-SQN-C-001, Revision 1, could be better integrated and cross-referenced with the main report to more clearly explain the context of the assessment performed to address the F&O. For example, the main report could refer to Appendix P relative to how the AF limit was used as part of the liquefaction assessment.	MET

Table A-2 Focused Peer Review Findings Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
					like or clay-like. As shown on Appendix E, Figures E34 to E36, the simulated soil profiles for the DGB include numerous layers that would behave as sand-like material. A summary of the DGB total vertical settlement is provided on Table 6-12 for the five peak ground accelerations values modeled. The peer reviewers requested that the analyst provide a measure of the impact on Table 6-12 for the contribution from sand-like layers to the total vertical settlements for the DGB for all five peak ground acceleration values. 3.Section 5 and Appendix G of the report describes the approach to assessing liquefaction settlement. Section G.2.4 and equation G.25 describe the liquefaction evaluations for clay-like soil. The approach taken computes the peak stress in the soil layer by Sa * rd. The peer reviewers requested that the following technical issues be described: A. The Sa is calculated by taking the rock peak ground acceleration value and multiplying by the site response AF. The text should be clarified to discuss whether the AF limit of 0.5 is used when applying equation G.25 for the set of LHS simulations performed. B. Appendix G indicates that a composite shear-wave velocity profile was developed considering the three individual shear-wave velocity profiles used for the site response analysis. The text should be clarified to describe or show the resulting composite shear-wave velocity profile used in the LHS. C. The peer reviewers questioned whether Sa and rd should be correlated. Experience would suggest that the Afs are profile dependent, and as applied here could have different mean and standard deviation if Afs versus PGA were assessed by individual profile versus using a composite profile. The analyst	2. Using the simulated profiles developed for the DGB, an evaluation was performed that excluded the contribution to settlement from sand-like layers. The settlements from this evaluation were compared with the settlement computed from the inclusion of sand-like and clay-like settlements in the layers. The contribution to total settlement from sand-like layers is between 20%-30%. By incorporating a Cetin and Bilge depth deduction factor into the DGB settlement evaluations that include the sand-like and clay-like materials and comparing with settlements considering only clay-like layers, the contribution to settlement from sand-like layers drops to approximately 5%-8%. Note that depth reduction factor for sand-like settlements was not included the evaluations documented CJC-SQN-C-001. 3A. The site response analyses performed for the PSHA limits the SAF to 0.5 but does not limit the SAF ± σLN to 0.5. Thus, the Sa computed for the liquefaction study as rock PGA value and multiplying by the site response AF is not limited to be consistent with the PSHA approach. A sensitivity study is performed to assess the impact of limiting the AF to 0.5 (i.e. setting a hard limit of 0.5 on the LHS distribution of AF) using the DGB area soil data. The results of this study show that the difference in estimated settlements are very small. Lateral displacements are slightly larger when the SAF is limited to 0.5. 3B. The text was clarified to describe and show the resulting composite shear-wave velocity profile used in the LHS. 3C. In response to the question of whether Sa and rd					

Table A-2 Focused Peer Review Findings Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
					was requested to assess whether the assumption of uncorrelated SA and rd has a significant impact on the results and whether the vertical settlement results would be sensitive to using each of the individual three soil profiles. 4. Within Appendix G of the report multiple symbols are provided for the probability of liquefaction (equation G.3 and G.8). 5. Within Appendix F (Section F.5) a more detailed definition or description of the slide ratio is needed.	should be correlated, Sa and rd are correlated through the rd equations in Figure G1. Two of the inputs to the G1 equations are Sa and Vs12 and the question arises as to whether these two input parameters should be correlated. As seen, in the sensitivity study, there doesn't appear to be a direct correlation between the average Vs for the individual profiles and Sa. To assess the importance of correlating Vs12 and Sa an additional sensitivity study was performed for 3 ground motion levels considering the parameters uncorrelated, positively correlated, and negatively correlated. The results of the study show that the computed shear stress is not sensitive to correlation, or lack thereof. Additionally, a sensitivity study was performed to address the reviewer's expectation that the AFs are profile dependent, and as applied as discussed in the report could have different mean and standard deviation if AFs versus PGA were assessed by individual profile versus using a composite profile. The study shows that the use of the composite section results in a slight conservative bias in the median shear stress (demand to the liquefaction assessment) and higher variability as compared to using the three individual profiles. 4. The symbol for probability of liquefaction was changed to SI throughout the document. 5. A more detailed definition/description of the slide ratio is added to the text in Section F.5.					

Table A-2 Focused Peer Review Findings Consensus Table

1 RU	2 SR	3 CC Assessment	4 Finding No.	5 Description	6 Prior Peer Review Assessment	7 Self-Assessment Closure Basis	8 Self-Assessment Reference Document(s)	9 Maint (M) or Upgrade?	10 Basis for Maint (M) or Upgrade	11 Independent Review Team Disposition	12 Independent Review Team SR Assessment
1	SHA-J2	MET	1-4 (Suggestion)	Carl J. Costantino and Associates Report CJC-SQN-C-001, Revision 0 (January 16, 2019), Updated Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant (SQN) provide a new approach and method for evaluating soil failure modes caused by liquefaction and cyclic softening. Suggestions are provided that the IAT believes will improve the documentation. (This F&O originated from SR SHA-J2)	The Latin Hypercube sampling (LHS) technique incorporates both epistemic and aleatory uncertainty. Conventionally, these two types of uncertainty are modeled separately.	The text of CJC-SQN-C-001 was modified in Revision 1 of the document as needed to incorporate the responses to each of the Items listed in the F&O.	Response to IAT Question. "Updated Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant (SQN)', Revision 1, 4/1/2019, CJC-SQN-C-001, Carl J. Constantino and Associates.	Maintenance	No new technology	RESOLVED: Section 5 of Carl J. Costantino and Associates Report CJC-SQN-C-001, Revision 1 (April 1, 2019), Updated Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant (SQN) provides enhanced documentation to distinguish between aleatory and epistemic uncertainty in the Latin Hypercube sampling technique used to assess liquefaction deformation.	MET

A.6 Summary of Technical Adequacy of the SPRA

The set of supporting requirements from the ASME/ANS PRA Standard [8] that are identified in Tables 6-4 through 6-6 of the SPID [3] define the technical attributes of a PRA model required for a SPRA used to respond to implement the 50.54(f) letter. The conclusions of the peer review discussed above and summarized in this submittal demonstrates that the SQN SPRA model meets the expectations for PRA scope and technical adequacy as presented in NRC RG 1.200, Revision 2 [27] as clarified in the SPID [3].

The main body of this report provides a description of the SPRA methodology, including:

- Summary of the seismic hazard analysis (Section 3)
- Summary of the structures and fragilities analysis (Section 4)
- Summary of the seismic walkdowns performed (Section 4)
- Summary of the internal events at-power PRA model on which the SPRA is based, for SCDF and SLERF (Section 5)
- Summary of adaptations made in the internal events PRA model to produce the SPRA model and bases for the adaptations (Section 5)

Detailed archival information for the SPRA consistent with the listing in Section 4.1 of NRC RG 1.200 Rev. 2 is available if required to facilitate the NRC staff's review of this submittal.

The SQN SPRA reflects the as-built and as-operated plant as of the cutoff date for the SPRA, January 2016. There are no permanent plant changes that have not been reflected in the SPRA model.

A.7 Summary of Technical Adequacy of the SQN Internal Events PRA

The PWROG performed a full scope peer review of the SQN internal events PRA and internal flooding PRA that forms the basis for the SPRA to determine compliance with ASME PRA Standard, RA-S-2008, including the *2009 Addenda A* [8] and NRC RG 1.200 [27], in the week of January 31, 2011. This peer review was performed using the process defined in NEI 05-04. The ASME/ANS PRA Standard contains a total of 325 supporting requirements for internal events and internal flooding in 14 technical elements and the configuration control element. Twelve of the SRs were determined to be not applicable to the SQN PRA. Of the 313 remaining SRs, 294 SRs, or 94%, were rated as SR Met, Capability Category I/II, or greater. Eight SRs were rated as Category I and eleven (11) SRs were not met.

A peer review findings closure review was performed for the SQN Units 1 and 2 Internal Events PRA from May 8 through May 10, 2017. The review evaluated how TVA addressed the F&Os that were classified as "Findings" from the 2011 peer review conducted by the Westinghouse Owners Group. The closure review was performed in accordance with the process documented in Appendix X to NEI 05-04, as well as the requirements published in the ASME/ANS PRA Standard (RA-Sa-2009) and NRC RG 1.200, Revision 2.

The assessment was performed by a team of three independent PRA experts and complied with the requirements for a findings closure review (as documented in

Appendix X to NEI 05-04 and other industry documents). Each F&O closure was reviewed by at least two team members and consensus sessions were held to determine whether the F&Os could be considered to be closed.

In addition to assessing the closure status, the changes made to the SQN PRA to address the F&O were also evaluated to determine whether the changes constituted a “PRA Upgrade” or if new PRA methods were introduced. The definition of PRA Upgrade as defined in the ASME/ANS PRA Standard was used. The performance of a PRA Upgrade or the use of new methods would require that a peer review be performed instead of a findings closure review. None of the changes made to the SQN PRA were considered by the review team to constitute a PRA Upgrade or the usage of a new PRA method.

The results of the closure review show that all 31 Findings from the 2011 internal events PRA peer review can now be considered to be closed. In the case of one finding (6-5), the appropriate model changes have been made in the living PRA model and will be incorporated into the next Internal Events Model of Record revision. Finding 6-5 involves the modeling of support system initiating events, which does not affect the SPRA since the seismic event is the initiating event.

Table A-3 – SQN Internal Events F&O Closure Review Consensus Table

F&O Number	Applicable SRs	F&O Text	Closure Status	Actions to Address Finding	Acceptability Evaluation	Upgrade?	New Method?	Comments
1-4	IFQU-A6	MDN-000-000-2010-0203 does not document an assessment of the impact of flooding events on existing HFEs carried over from the internal events scenario used to represent the flooding event	Closed	The HRA notebook (MDN-000-000-2010-0204) contains a section discussing the new HFEs added to mitigate flooding events). A new section 9.3 was added to the Internal Flooding notebook (MDN-000-000-2010-0203) to discuss the internal events HFEs that could be impacted by the flooding event. The local operator actions from the internal events PRA were reviewed to determine which (if any) could be affected by flooding events. One such event, HAAC1, was determined to rely on instrumentation that would fail during some flooding scenarios. The cognitive portion of the HRA calculation was modified to reflect this instrumentation failure and the HRA calculator was used to generate a new HEP. This calculation is documented in Appendix B of the HRA analysis document.	Section 9.3 of the flooding notebook identifies each of the local actions credited in the internal events PRA and indicates the flood areas where the action takes place as well as pathways to perform the event. Actions that would be impaired by the flood were conservatively assumed to be failed. Other in-control room actions were assessed for impacts due to stress or due to failure of unqualified instrumentation. Human failure event HAAC1 from the internal events analysis was re-evaluated from a flooding perspective and new event HAAC1_FL created to reflect the instrumentation relied upon by the action that is failed during a flooding event. The cognitive portion of the HRA Calculator input was increased from 3.0E-03 in the internal events analysis to 6.0E-03 in the flooding analysis to account for the failed instrumentation. The new HEP is documented in Appendix B of the HRA analysis (MDN-000-000-2010-0204) and appears in multiple scenarios quantified for the flooding analysis as shown in Appendix F of the flooding analysis (MDN-000-000-2010-0203). Other than this one HFE, all other human actions were either assumed to be failed due to the flood, or were determined to be unaffected by the flooding impacts. The resolution of this F&O meets the CC-I/II/III requirements of SR IFQU-A6. The F&O is considered to be closed. The resolution of this F&O does not constitute an upgrade, as only documentation of the single modified HFE was added. This HFE, while appearing in multiple scenarios, does not appear to be a significant risk contributor. In addition, no new methods were used to resolve this F&O.	No	No	
1-7	HR-G7	Dependency analysis was performed for the post-initiator HEPs using the EPRI HRA Calculator. However, issues were identified including, 1) application of conservative dependency values between two HEPs where the same cue had been incorrectly specified for both (i.e., HARR1 and AFWOP3), and 2) combinations appearing that do not seem valid because the timing analysis for the first action indicates Tsw occurs before the cue is received for the following action (i.e., HARR2 and AFWOP3). In addition, it was noted that the dependency level of the cognitive recoveries were not entered in the HRA Calculator database for the post-initiators. This may underestimate or overestimate the HEP depending on the applicable dependence level. Some of these items were corrected during the review but they are documented in an F&O due to the need to look into the extent of the condition.	Closed	1. The cue for AFWOP3 was updated to its correct value. Review has been performed for all remaining actions to determine if any additional cues need to be updated. This review verified the accuracy of HRA cues and updated six of the identified cues. 2. The Tsw values for HFEs in dependency combinations were reviewed for overlapping timeframes. 3. Screening value HEPs were removed from the database if their values were set to 1.0. The HEPs that were originally in the model were no longer required and were deleted from the fault tree. 4. Dependency levels were entered for all post-initiators in the HRA Calculator database.	Cues for HFEs HARR1 and AFWOP3 (which had been incorrectly specified for both) have been corrected in the HRA calculator documentation (MDN-000-000-2010-0204). The review of the other HFEs resulted in the correction of six other HFEs. HFEs with screening values of 1.0 were removed from the model, however, the .RR file still contains basic event HARCW1 which as a value of 1.0 and is indicated in the HRA calculator documentation (MDN-000-000-2010-0204) to be a screening value. Dependency level of the cognitive recoveries have now been entered into the HRA Calculator database for post-initiator HFEs. The F&O resolution meets CC-I/II/III requirements for HR-G7 because it corrected errors in timing and cues for identified HEPs as well as making similar changes for additional HEPs based on extent of condition review. In addition, minor improvements to the HRA documentation were made. The resolution of this F&O does not involve any new methods and is not a PRA upgrade. Existing HEPs were reviewed to validate cues and timing relative to dependency considerations and model corrections were made based on the review. These actions are typical of those performed for model maintenance.	No	No	1. Confirmed that Tdelay for HARR1 and AFWOP3 reflected each action and were different. 2. Determined that HARR2 is used generally for failure to recover from auto swap over failure (i.e., not just Large LOCA as stated in HRA calculator). 3. Confirmed that database does not include HEPs with a value of one (with a few exceptions related to flooding). 4. Confirmed that dependency levels were entered for all post-initiators in the HRA Calculator database It would be good to list the six HEP whose cues were changed. HRA NB and TH analysis shows a value for Tdelay of 35.5 minutes (based on case AFWOP3-008). The HRA NB states that "Tdelay – For this analysis the point at which operator actions to depressurize and cooldown the RCS start when the RWST level reaches 8%." and the timing of that event is 35.5 minutes in Table 8.2-3 in the TH NB and stated as the same value in the HRA NB.
1-8	IFQU-A10	MDN-000-000-2010-0203 Section 9.5 only addresses quantification and results for CDF. There is no discussion of LERF for the flooding scenarios or documentation indicating that the flood scenarios were reviewed to determine if they would have an impact on the Level 2 CETs. The linked fault tree model should have the capability to produce LERF results, but this had not been done at the time of the review. In addition, there was no discussion in the Level 2 Notebook (MDN-000-000-2010-0206) that indicates the results include the internal flood scenarios.	Closed	Section 11 was added to the internal flooding notebook (MDN-000-000-2010-0203), as well as several appendices, to document the results of the LERF analysis for flooding. • Section 11.1 addresses the eighteen questions concerning LERF and their impact. • Section 11.3 and 11.4 address the LERF results due to flooding • To address the additional information the following Appendices were added to the model: • Appendix N - Significant Cutset Review	Section 11 of the flooding notebook provides documentation of the LERF quantification. Section 11.1 lists a series of questions used in the analysis to determine if an event qualifies as a LERF event. Each of these questions is addressed from a flooding perspective to determine flooding impacts on LERF. While this is not a systematic review of the Level 2 top events as documented in F&O 1-8, it does appear to disposition all the factors that could impact CETs. Sections 11.3, 11.4, and 11.5 were added to the flooding analysis (MDN-000-000-2010-0203) and document graphically (with pie charts and bar charts) the flooding contribution to LERF for each unit and the contribution to flooding LERF by system. These sections offer no discussion of the results however. Given that internal flooding contributes >80% to overall LERF, some discussion of these results is	No	No	

Table A-3 – SQN Internal Events F&O Closure Review Consensus Table

F&O Number	Applicable SRs	F&O Text	Closure Status	Actions to Address Finding	Acceptability Evaluation	Upgrade?	New Method?	Comments
					warranted. Appendix N identifies significant cutsets, but does not offer any discussion surrounding the validity of the cutsets identified. SR LE-F2 states that the contributors should be reviewed for reasonableness. Without some discussion there is no way to know that such a review has been performed. Subsequent to the on-site review, TVA provided a revised version of the notebook (Revision 4) that provided further discussion of the LERF results. It is suggested for future updates that further discussion be provided as to why these flood scenarios have a disproportionate impact on LERF. This F&O is considered to be closed. The resolution of the F&O satisfies the Category I/II/III requirements of IFQU-A10 as they pertain to the proper consideration of the effects of the flood on LERF. The resolution of this F&O is not considered to be a PRA upgrade. The flooding analysis did not result in any modeling changes specific to LERF and did not involve any new methods, did not create any new accident sequences, and did not result in a significant change in the risk results.			
1-10	LE-C4, LE-C13, LE-E3	MDN-000-000-2010-0206 Section 5.6 notes that credit was taken for scrubbing of releases from a ruptured SG. However, the technical justification for this credit needs to be strengthened. The current basis compares the zero power collapsed level to the top of the SG tubes. However, ES-3.1, Post- SGTR Cooldown Using Backfill allows the level in the ruptured SG to be between 20% narrow range and 75% narrow range during the cooldown (Step 7). The expected levels during SGTR recovery should be used to justify the scrubbing credit. It also appears that the analysis implicitly assumes that if FW will be applied to the ruptured SG if FW is available. No consideration of operator failure to provide FW flow to the ruptured generator is included in the analysis	Closed	Additional discussion of the basis for claiming SG scrubbing credit was provided in the Level 2 notebook (MDN-000-000-2010-0206). The L2 NB was updated to discuss the targeted water levels above the SG tubes during SGTR recovery actions. The response states that these water levels are between 4.7 and 9.8 feet, and concludes that should be sufficient to take credit for fission product scrubbing. The response states that the analysis assumes that the operator is successful in providing feedwater flow to the ruptured steam generator and maintaining water level.	While further discussion is provided in section 5.4 of the Level 2 notebook to demonstrate that the ruptured SG tubes would remain well-submerged with water if SG levels are maintained per procedure, the assumption that the operators will be successful in maintaining that level has not been addressed. Therefore, this F&O has only been partially addressed. Note that this credit for fission product scrubbing by water above the SG tubes only applies to sequences SGTR-28 and SGTR-29; it would also be helpful to add this fact to the documentation to clarify when scrubbing is being credited. Subsequent to the on-site review, TVA revised the discussion of SGTR scrubbing to specify applicability to FW success sequence SGTR-28 and SGTR-29, clarified likely SG water levels for these scenarios, and removed the operator action assumption. The latter change is appropriate since whether or not a water level control action occurs, the discussion shows that the SGTR opening will be covered by a significant water pool. These changes along with the previous changes made by TVA satisfy the intent of the F&O. As a result, this F&O can be closed. The F&O resolution meets CC II requirements for SR LE-C4 and meets CC II/III requirements for LE-C13 by improving the technical justification for crediting scrubbing for two particular SGTR sequences. The F&O resolution meets CC II requirements for SR LE-E3 by justifying a non-LERF outcome for certain sequences that can assure water scrubbing of radionuclide releases from a ruptured SG. The resolution of this F&O does not involve any new methods and is not a PRA upgrade. The event tree end-state changes involved are routine actions. The remaining changes involved only documentation enhancements.	No	No	It is agreed that EPRI TR-101869-V2 is a reasonable basis for scrubbing water above tubes. Since the value of DF that is being credited is not estimated, it would be reasonable to state that it makes the result no worse than a large late rather than a small early release. Note that this credit for fission product scrubbing by water above the SG tubes only applies to sequences SGTR-28 and SGTR-29. There does not appear to be an operator action regarding AFW in the SGTR sequences. Since other related actions fail in the two sequences of interest (i.e., identifying the ruptured SG for SGTR-29 and failing to stop the tube leakage or SGTR-28), it is likely that water level control in the ruptured SG will also fail. However, given the success of AFW and continuing inventory addition from the RCS, there should be a much higher water level in the SGs than the 20% NR that is discussed as providing scrubbing.
1-11	LE-F2	The total LERF is compared with other Westinghouse 4-loop plants and with other Ice Condenser plants. However, there is no comparison at the level of significant contributors or plant damage states. Without the contributor information, it is not really possible to determine how similar the LERF results are to other plants.	Closed	The Level 2 NB (MDN-000-000-2010-0206) has been updated to include Table 11-7, which presents a comparison between percentage of LERF contribution by initiating event for several other PWRs.	The additional documentation does not fully meet the requirements of the SR. The discussion added identifies trends in other plants contributors but does not identify Sequoyah differences (e.g., much higher LERF percentage from SLOCA) nor does it discuss whether the Sequoyah results seem reasonable based on the IE contributions. For example, SQN shows small LOCA as a 30% contributor while no other plant is	No	No	The use of percentages is not straightforward regarding inter-plant comparison since the plant CDF magnitudes are quite different.

Table A-3 – SQN Internal Events F&O Closure Review Consensus Table

F&O Number	Applicable SRs	F&O Text	Closure Status	Actions to Address Finding	Acceptability Evaluation	Upgrade?	New Method?	Comments
					greater than 2%. Subsequent to the on-site review, TVA updated Section 11.3 of the Level 2 Analysis to include a discussion of differences between SQN and the other plants, including small LOCA. The discussion shows that the SQN LERF results appear reasonable compared to other plants' results, which satisfies the intent of the F&O. The resolution of the F&O satisfies the Category I/II/III requirements of LE-F2 as they pertain to the reasonableness of LERF contributors. This F&O is considered to be closed. The resolution of this F&O is not considered to be a PRA upgrade. The review of LERF contributors did not result in any modeling changes and, therefore, did not involve any new methods, did not create any new accident sequences, and did not result in a significant change in the risk results.			
1-14	DA-C6	Demand data is obtained directly from the plant process computer for most components, as described in Section 7.3 of the data notebook (MDN-000-000-2010-0202). The status chance information from the computer is filtered and used to determine the number of demands. The use of automatic data collection, however, means that start and run events that occur in all modes of operation are included. In addition, post-maintenance test starts are also included in the data set. This is identified as a source of uncertainty in the sensitivities and uncertainties notebook (MDN-000-000-2010-0209) and a specific set of sensitivity studies were performed that assumed that various numbers of successful starts were invalid. The results show that the impact on CDF is relatively small, unless the number of successful starts is overestimated by a large amount. However, this SR is explicit in its requirement to not count post-maintenance test events.	Closed	The work orders for the components that were credited for success in the data analysis were reviewed to discover the number of post maintenance tests that were performed on the components. Table 15 in the data notebook was added to document the number of post maintenance tests that were removed from the analysis.	Table 15 of the data analysis (MDN-000-000-2010-0202) clearly identifies the number of post-maintenance starts that were mistakenly included in the development of hardware failure rates. Table 16 then was updated to remove the information from Table 15 and, thus, include only valid starts in the data analysis. These corrections made the sensitivity analysis moot since the analysis was corrected. Subsequent to the on-site review, TVA modified Section 7.3.4 to exclude the discussion of the sensitivity study to analyze the impact of post maintenance testing. This sensitivity study was moot since the hardware failure data analysis was corrected to exclude post maintenance starts. The F&O resolution meets CC-I/II/III requirements for DA-C6 because it corrected an error in data collection by removing credited start demands for post-maintenance testing (the SR explicitly requires such data to be excluded). This discrepancy was believed in the past to be remedied by a sensitivity case that showed the effect was non-significant. As part of the resolution, this sensitivity case was also removed. This F&O is considered to be closed. The resolution of this F&O does not involve any new methods and is not a PRA upgrade. Review and adjustment of component data are standard maintenance activities. All other changes required were related to correcting the documentation of the work and removing the sensitivity case.	No	No	
1-15	AS-A10, AS-B1, SC-B3	The super initiator "general transient" may overlook certain differences among its contributors. For example, the impact of specific IEs like LOSP and Loss of DC that may prevent PORV operation and challenge the Pressurizer Safeties do not appear to be captured. In addition, failure to provide a separate event tree for SBO may overestimate the success of power recovery by not addressing the operation of systems such as charging and AFW following power recovery	Closed	The GTRAN event tree was retained. However, the event tree structure was enhanced to question PORV operability and RCP seal cooling under SBO and loss of bus initiators. The underlying nodal fault trees include modeling to fail the appropriate ET nodes under Loss of DC and SBO conditions. The consideration of failure of mitigating systems following offsite power recover in an SBO was not added to the PRA model.	The modifications to the GTRAN event tree and supporting fault trees address the concerns of the first part of the F&O. However, it does not appear that the second portion of the F&O, pertaining to modeling of post-LOSP operation of mitigating systems, was addressed. The resolution response should be updated to provide some insight about the estimated numerical impact. As long as this sensitivity study continues to be maintained for future model updates, this approach is acceptable in lieu of developing a separate SBO event tree. It could postulates that a specific future risk application might show a higher sensitivity to this modeling simplification than the base case sensitivity study shows. However, the systems involved in post-power recovery (HPI and AFW) are already modeled in other events. So, if an application was changing something that impacted the risk significance of these two systems, we should already see that impact in the other initiating events in the PRA. Subsequent to the on-site review meeting, TVA performed a sensitivity study (documented in the Sensitivity and Uncertainty notebook) to estimate the impacts of possible mitigating system	No	No	An updated bounding case should be performed using the most recent PRA model and ensure that other scenarios (e.g., loss of AFW after recovery) are also considered. This was done as part of the subsequent revision, so this comment is closed.

Table A-3 – SQN Internal Events F&O Closure Review Consensus Table

F&O Number	Applicable SRs	F&O Text	Closure Status	Actions to Address Finding	Acceptability Evaluation	Upgrade?	New Method?	Comments
					failure. The sensitivity study showed the impacts to be small. This F&O is therefore considered to be closed. The F&O resolution meets CC II requirements for SR AS-A10 because it improves the plant response modeling to distinguish differences in system requirements and operator actions for the various event trees. Explicit consideration of the potential for random equipment failure following OSP recovery following SBO represents an additional improvement but the omission is non-significant for the base model. The F&O resolution meets CC I/II/III requirements for AS-B1 because the improved modeling better represents the impact of the SBO and Loss of DC events on the mitigating systems. The F&O resolution meets CC I/II/III requirements for SC-B3 because the improved modeling uses existing success criteria that are appropriate for the SBO and Loss of DC events. The resolution of this F&O is not a PRA upgrade. While it included modifications to the GTRAN tree, these modifications did not change the overall accident sequences (the changes moved some items treated in the nodal fault trees up to the event tree node level). The remaining changes pertained to documentation updates. The sensitivity study demonstrated that the inclusion of the additional modifications suggested by the F&O would not have a significant impact. In addition, no new methods were employed.			
1-19	LE-C7	It was noted that HFE HAPZR (discussed in Section 6.8 and Section 7.2) is not calculated using HRA Calculator. This event seems to have been carried over from the Watts Bar analysis and is treated as basic event U1_L2_NOTRCSDEPNOSBO. In addition, although Section 6.8 says that the No RCS Dep branch is set to a value of 1 for SBO cases, the value of basic event U1_L2_NOTRCSDEPSBO in the provided MASTERL2.CAF fault tree was set to 0.9995. This also appears to be a carryover from Watts Bar.	Closed	Removed basic event U1_L2_NOTRCSDEPNOSBO from the model. HFE HAPRZ was added to the HRA calculator. A success branch value, HAPRZ_SUC was also added to the L2 Event Trees. SBO sequences were changed such that early depressurization is a guaranteed failure.	Confirmed that Watts Bar Basic Events are no longer used. Section 6.8 indicates that gate U1_DP has a value of 1.0 for SBO cases and this appears to be reflected in the fault tree. Similarly, for non-SBO cases U1_DP has a value of 0.0 which also is reflected in the fault tree. HFE HAPZR is now calculated using HRA Calculator as documented in the HRA analysis (MDN-000-000-2010-0204). Basic event U1_L2_NOTRCSDEPNOSBO no longer appears in the SQN model. Reviewed L2 ET NHD-NON SBO and confirmed that the sequences developed on the Question 7 failure branch all include Basic Event HAPRZ-SUC which is assigned a value of 0.988. This F&O is considered to be closed. The F&O resolution meets CC-I/II/III requirements for LE-C7 because it corrected errors in the Level 2 HRA related to specific HEPs. The corrections included adding one of the HFEs to the HRA Calculator. In addition, minor changes to the Level 2 documentation of these HFEs were made. The resolution of this F&O does not involve any new methods and is not a PRA upgrade. An existing HEP was re-analyzed using the HRA Calculator and added to joint HEP combinations which is an existing method. Other minor related documentation and modeling changes were made. These actions are typical of those performed for model maintenance.	No	No	The Level 2 analysis (MDN-000-000-2010-0206) needs to be updated to include the latest CETs. This is a documentation issue only.
2-1	IE-C12, IE-D2	Section 7.0 of the Initiating Events Analysis observes a decreasing trend in initiator frequency in the more recent generic data sources. However, there is no comparison of the SQN results against the generic results nor an explanation of any significant differences.	Closed	A comparison table was added to Section 9 of the Initiating Event Notebook (MDN-000-000-2010-0210) to compare the latest initiating events frequencies to those used in the last PRA revision, as well as to compare against the frequencies noted in the PLG PRA database, NUREG/CR-5750, and NUREG/CR-6928.	The newly added Table 15 provides an adequate comparison of SQN IE frequencies to those of generic sources. While no significant differences are noted in this table, the IE document (MDN-000-000-2010-0210) offers no explanation of the results. In fact, the text of the IE document never introduces or references the table at all. Conclusions drawn from the comparison of the information in Table 15 needs to be added to the document.	No	No	

Table A-3 – SQN Internal Events F&O Closure Review Consensus Table

F&O Number	Applicable SRs	F&O Text	Closure Status	Actions to Address Finding	Acceptability Evaluation	Upgrade?	New Method?	Comments
					Subsequent to the completion of the on-site review, TVA issued Revision 2 of this notebook, which included an appropriate discussion of the results of the comparison of the various initiating event data sources. Therefore, this F&O is considered to be closed. The resolution of the F&O satisfies the Category I/II/III requirements of IE-C12 and IE-D2 as they pertain to the comparison of results and explanation of differences between the plant-specific and generic initiating event frequencies. The resolution of this F&O is not considered to be a PRA upgrade. The comparison of plant-specific and generic initiating event frequencies and the subsequent discussion of differences did not involve any new methods, did not create any new accident sequences, and did not result in a significant change in the risk results.			
2-3	DA-D1, DA-D3	Section 4.3.1 of the Data Analysis notebook discusses the basic event probability model methodology. Generic data sources selected for use are applicable for SQN. For those components which had a failure during the analysis time period (1/1/03 - 11/30/09), the distributions are updated via the Bayesian update program built into CAFTA program. However, the intent of this supporting requirement is to assure realistic parameter estimates are calculated for SIGNIFICANT basic events based on relevant generic and plant-specific evidence, not just those for which failures have occurred. Where no failures have occurred, use of the generic data may be conservative since it includes failures from potentially less reliable components across the industry.	Closed	Plant-specific data for an additional failure type codes with no observed failures were documented in Table 10 of the data notebook (MDN-000-000-2010-0202) and were included in the Bayesian update of the PRA dataset.	The combined Bayesian-updated dataset for the Sequoyah PRA adequately reflects all major component types. Table 17 was added to the data notebook (MDN-000-000-2010-0202) and lists those significant contributors for which there were no recorded failures and also lists their demands and operating hours. Table 21 documents the posterior failure rates based on the information in Table 17. The resolution of the F&O satisfies the Category II requirements of DA-D1 and DA-D3 as they pertain to the development of significant component failure rate distributions. This F&O is considered to be closed. The resolution of this F&O does not constitute a PRA upgrade. While basic data values were updated, the change in basic event data is equivalent to a routine PRA data update. No new methods were used, as Bayesian updating was already employed, as noted in the F&O text.	No	No	
2-4	DA-D4, DA-E2	Appendix F of the Data Analysis notebook provides graphs that show the prior and posterior distributions. Table 19 lists generic and Bayesian-updated mean values, along with a ratio of the posterior to prior mean value. However, there are no conclusions drawn about whether or not the posterior distributions are reasonable given the relative weight of evidence provided by the prior and the plant-specific data. (Note: the statement that "There are no significant differences between the industry data from NUREG/CR-6928 and the posterior distributions for the SQN failure rates" in section 11.0 is not judged to be sufficient. For example, the ratio of the posterior to prior mean for the AHUFR type code in Table 19 is 10.6. For type code LSTFR, the ratio is 4.3. The significance of these differences should be discussed.)	Closed	Appendix F of the Data Notebook (MDN-000-000-2010-0202) has been updated to include an evaluation of each of the reasonableness of the prior and posterior distributions. The posterior distributions were validated using the following process. Using a Monte Carlo simulation, the posterior distributions were sampled to see the probability of having a recurrence in the number of events observed in the data window given the number of successes in the data window. If the mean value was within 0.05 to 0.95 the resultant distribution was used within the model. Appendix F was re-written to address this analysis as well as to present the prior, posterior, and plant specific distributions. One case was identified in which the posterior distribution was not valid for use.	Appendix F of the data notebook (MDN-000-000-2010-0202) was rewritten in response to this finding. Each entry in Appendix F now provides a brief summary of the validity of the posterior distribution based on Monte Carlo analysis of both the prior and posterior distributions. The resolution of this F&O meets the CC-II/III requirements of SR DA-D4 and the CC-I/II/III requirements of SR DA-E2. This F&O is considered to be closed. The resolution of this F&O is not a PRA upgrade since it was primarily a documentation enhancement to validate the Bayesian update process. While one data value was changed to conservatively use only the plant specific data (i.e., for service water system pressure switches), the impact of this data change does not constitute a change in methodology nor would it significantly impact the results.	No	No	
2-5	DA-D4, DA-E2	The method from NUREG/CR-6823 is used to Bayesian-update a Jeffrey's noninformative prior distribution with plant-specific experience. However, there is no comparison of the posterior means to plant-specific means. (See the last sentence in NUREG/CR-6823, section 6.7.1.2.)	Closed	Assumption 10 of the data notebook (MDN-000-000-2010-0202) was added to indicate that a Jeffrey's prior was used to calculate plant-specific unavailability distributions since there is no prior information from which to Bayesian update. Therefore, the methodology used was to use a Jeffery's non-informative prior (0.5) as the foundation for the update process. All of the available data that was used was from plant specific data collection, therefore the posterior mean and plant specific mean are directly correlated.	When using Bayesian analysis to develop a failure rate or frequency, the posterior mean and plant specific mean will always be correlated since both methods make use of the same data. SR DA-D4 clearly states that the posterior results from a Bayesian approach must be verified as reasonable when compared to the prior and the plant data. Comparison of the posterior results to a Jeffrey's uninformed prior would be meaningless, but comparison to the plant specific mean could offer insights as to the validity of the posterior results. It appears that no such comparison has been performed. Subsequent to the on-site review, TVA issued Revision 4 to the notebook, which included a new Appendix L which performed the comparisons of the posterior unavailability distributions to the plant specific mean values. Section 8.3 of the notebook was also revised to discuss the results of this comparison. The resolution of the F&O satisfies the Category II/III requirements of DA-D4	No	No	

Table A-3 – SQN Internal Events F&O Closure Review Consensus Table

F&O Number	Applicable SRs	F&O Text	Closure Status	Actions to Address Finding	Acceptability Evaluation	Upgrade?	New Method?	Comments
					and Category I/II/III of DA-E2 as they pertain to the validity of posterior distributions through comparison to the plant-specific mean. This F&O is considered to be closed. The resolution of this F&O is not considered to be a PRA upgrade. Documentation of the validity of distributions through comparison to the plant-specific means did not involve any new methods, did not create any new accident sequences, and did not result in a significant change in the risk results.			
2-8	QU-D7	The importance of components and basic events are identified in sections 5.1 and 5.7 of the Accident Sequence notebook, respectively. However, documentation that determined the importance results make logical sense could not be identified.	Closed	Added discussion on Section 9 "Conclusions" of Quantification NB. A review of the importance of components and basic events has been performed to determine that they make logical sense. The review shows that the risk significant components are consistent with the model results and limitations. Significant contributors include basic events associated with diesels, ERCW, Component Cooling, RHR, Atmospheric Relief Valves (ARVs) and Air Compressors. In SQN, failure of the auxiliary control air headers impacts the ARVs that are needed to cooldown/depressurize in LOCA scenarios since the condenser is unavailable from a Phase B isolation. The emergency diesel, ERCW, RCP breakers, and RHR are important since their failure result in scenarios involving SBO and RCP seal LOCAs. The review is documented in the Quantification Notebook Section 4.6.	The discussion that was added to the quantification notebook added provides minimal basis for acceptance and meeting SR QU-D7. Tables 13 and 14 of the quantification notebook (MDN-000-000-2010-0208) list the importance measures of all significant components. Appendix F of the same document lists the importance measures for all plant components. While the resolution of the F&O states that the review was performed and that the risk significant components are consistent with the model results, this is not as well documented as they could be. Subsequent to the on-site review, TVA modified the documentation in the conclusions section of the QU notebook to show why the most important systems and components make sense given how they contribute to accident mitigation. While this discussion is minimal, it satisfies the intent of the F&O. As a result, this F&O can be closed. The resolution of the F&O satisfies the Category I/II/III requirements of QU-D7 as they pertain to the review of components and basic events to assure they made logical sense. This F&O is considered to be closed. The resolution of this F&O is not considered to be a PRA upgrade. Review of components and basic events for logical sense did not involve any new methods, did not create any new accident sequences, and did not result in a significant change in the risk results.	No	No	
3-1	LE-D7	Section 4.5, 'The calculation above provides that the containment "hole" size must lie between a 1 inch equivalent path and a 4 inch path. Therefore, it is acceptable to use the NRC value of 2 inches.' Based on the statement, the 1" equivalent hole should have been considered.	Closed	The discussion in section 4.4 of the Level 2 Notebook (MDN-000-000-2010-0206) was enhanced to better justify the usage of the 2-inch lower limit.	The evaluation of the 2-inch diameter minimum containment hole size appears to be reasonable and is consistent with other NRC determinations of the required size of a containment hole to constitute a large release (100% of containment volume released per day). Therefore the concerns of this F&O have been resolved. The resolution of this F&O meets the CC-II requirements of SR LE-D7. The F&O is considered to be closed. The resolution of this F&O is not a PRA upgrade since it consisted of a documentation enhancement to justify the selection of the 2-inch hole size. This improves the realism in the basis for the minimum size of containment penetration failure leading to LERF. There were no model or data changes, and no new methods were used.	No	No	
3-7	SC-B3	Several areas were identified that need additional discussion with respect to the Success Criteria Analysis. For example: 1) The differences between plant response to a pipe-break SLOCA and a consequential PORV LOCA are not fully discussed. Given the differences in break location, there should be some discussion in the Success Criteria Notebook of why the pipe-break SLOCA analyses bound the consequential PORV LOCA. In addition, while there is a discussion in the TH Notebook comparing the values of some key parameters for the pipe-break SLOCA and the consequential PORV LOCA, this does not fully explore differences in plant response that may affect the success criteria.	Closed	The F&O response provides a discussion of each of the topics but the discussions related to Stuck-open RV, 480 gpm seal LOCA classification and SGTR flow rate do not appear to have been incorporated into documents. The medium LOCA ET was modified to consider the use of the accumulators in response to Item 4.	This F&O remains open because model documentation does not appear to be updated for items 1 through 3. Subsequent to the on-site review meeting, TVA revised the TH Notebook and Success Criteria Notebook to classify all 480gpm RCP seal LOCAs as small breaks. The TH notebook was revised to provide justification for the 700 gpm used for the SGTR event. The Success Criteria Notebook added further discussion of the consequential PORV LOCA and why is it bounded by the small LOCA pipe break event. Therefore, this F&O is now closed. The resolution of this F&O meets the CC-I/II/III requirements of SR SC-B3. The resolution of this F&O is not a PRA upgrade since primarily	No	No	- There is no discussion of grouping of the stuck open RV with the SLOCA in the success criteria NB or in the accident sequence NB. Sec. 4.4.11 in the TH NB includes a discussion providing a basis for treating a stuck-open PORV as a small LOCA, but the basis is different than provided in the resolution response. This is a documentation issue only. The comments below support the F&O closure conclusion and require no action: - The Accident Sequence NB states: "If a RCP Seal LOCA occurs the GTRAN is exited to the SLOCAV or SLOCA tree." Also, "GTRAN-28, -29,

Table A-3 – SQN Internal Events F&O Closure Review Consensus Table

F&O Number	Applicable SRs	F&O Text	Closure Status	Actions to Address Finding	Acceptability Evaluation	Upgrade?	New Method?	Comments
		2) There needs to be more discussion of why the 480 gpm per pump RCP Seal leaks are included in the Medium LOCA (MLOCA) grouping. It is stated in Section 4.4.10 of the TH Notebook that the 480 gpm seal LOCA meets the MLOCA requirement of not requiring AFW for accident mitigation, but there is no documentation of success criteria analyses that support this statement. 3) The basis for assuming a SGTR flow of 700 gpm in Section 7.2.10 of the TH Notebook needs to be discussed in more detail than simply noting that no historic SGTR has been of the magnitude of a double-ended guillotine rupture of a SG tube. 4) The LOCA analysis is limited to the upper and lower end of the break range for each class. TH analysis at the middle of the break range within the Large, Medium, and Small LOCA categories may provide insights that have not been revealed by the upper and lower end of the break. For instance, it is not clear if sequence MLOCA-011 can be a success path for a break in the 3 to 5 inch range.			documentation changes were made to the T/H and Success Criteria notebook to provide further justification of the assumptions used. The 480 gpm RCP seal LOCA was stated to have been re-classified to be a small LOCA; however, this does not constitute the use of a new method. Initiator contributions to risk between MLOCA and SLOCA might have changed; however, overall CDF and the relative importances of component and HRA basic events would not have been significantly impacted. Per the QU documentation, the primary contributor to SLOCA importance is actually stuck-open safety relief valves, and not RCP seal LOCAs.			and -30 A transient initiating event occurs resulting in a reactor trip. Reactor coolant pump seal injection or cooling is not maintained leading to a RCP seal leak. The WOG 2000 RCP Seal LOCA model is used to determine the leak flow rate. These sequences transfer to the small LOCA event trees." - The SC NB states: "When AC power recovery occurs before core damage, the remaining event for success may be covered in SLOCAV (21 gpm RCP seal leak), SLOCA (182 gpm RCP seal leak), and MLOCA (480 gpm RCP seal leak)." - Verified that the Medium LOCA ET was re-configured to include the cold-leg accumulators. These comments were addressed in the subsequent revised notebook provided by TVA and are now closed.
3-9	SC-A3	All mitigation strategies credited in the accident sequence model when the high pressure recirculation has failed are not prescribed by the corresponding EOPs. In other words, the mitigation credit in the event tree model has no basis. This issue has been self-identified by the SQN PRA staff and a corrective action report has been written for the EOP group to resolve this issue. At this stage the PRA group "firmly" believes that the EOP will be modified, not the model. Thus it is a tracking issue.	Closed	PRA group wrote a CR to have appropriate EOP update performed. CR was closed by implementing EOP ES-1.3 changes to address failure of high pressure recirculation. The EOP revisions were approved at the SQN PORC meeting on May 6th 2011.	Plant response is acceptable and this F&O may be closed. This was considered a tracking issue. Its resolution preserves meeting SR SC-A3. The F&O resolution meets CC I/II/III requirements for SC-A3 because it brought the plant operation (i.e., EOPs) back into conformance with the PRA modeling of the EOPs (specifically regarding actions after high-pressure recirculation has failed). The resolution used existing plant processes (i.e., the Corrective Action Program) and did not involve any modeling or documentation changes. The resolution of this F&O does not involve any new methods and is not a PRA upgrade. No PRA changes were required. The modeling of accident mitigation when high pressure recirculation fails was based on expected changes to existing EOPs to address the strategy. The EOP was indeed revised to reflect those expectations. This F&O is considered to be closed.	No	No	
3-14	SC-C1	Several documentation issues were noted in the Success Criteria and TH Notebooks. Specifically, 1) Figures 7-60 and 7-61 of the TH Notebook (MDN-000-000-2010-205) need to be replaced with updated results. 2) The discussion of accident sequence node LPH in Section 7.3.1 of the TH Notebook (MDN-000-000-2010-205) states that “The time for switchover to hot leg recirculation is specified in the EOP E-1 as 3 hours after the initiation of a large LOCA (Reference 4, Step 31c).” In the paragraph immediately below this statement, the calculation of the time available for recovery from a failure of recirculation uses a switchover time of 5 hours. Discussion with TVA personnel indicated that the 3 hour value was copied from the WBN notebook. The actual time specified in the SQN procedures is 5 hours. 3) Table 7-13 of the TH Notebook (MDN-000-000-2010-205) does not include success path ISLM-014 as shown in Figure 6.4-10 of the Accident Sequence Notebook (MDN-000-000-2010-0201). In addition, success path ISLM-017 in Table 7-13 of the TH Notebook is not shown in Figure 6.4-10 of the Accident Sequence Notebook. 4) Section 4.4.11 of the TH Notebook (MDN-000-000-2010-205) discusses the classification of a Stuck Open PORV as a small LOCA. The basis needs to be provided.	Closed	While figures 7-60 and 7-61 appear to be replaced, it doesn’t appear that items 2) and 3) noted in this F&O were not fully resolved. A discussion of the basis for item 4) concerning the classification of a PORV LOCA as a SLOCA appears to have been expanded.	Figures 7-60 & 7-61 figures were changed, but the text describing these plots needs to be corrected to match the new figures. In the 2nd paragraph below Table 7-5, the SLOCAV-001 description says the sequence includes "RCS depressurization through ARVs" which is consistent with the ET sequence progression but not included in the MAAP run. Also, the Figure 7-60 description should note that the SG pressure remains at the ARV set pressure since operator depressurization was conservatively neglected. Item 2) still has not been addressed. The reference to 3 hours still exists in the TH notebook (MDN-000-000-2010-205, Rev. 1 (see p. 150). For Item 3), it appears that the sequence numbering in Table 7-13 still does not match the success paths shown in the accident sequence notebook for the ISLOCA event tree. The success paths ISLM-010, 011 and 014 should be re-numbered to ISLM-009, 010 and 013, respectively, in Table 7-13. For Item 4), additional information included in section 4.4.11 of the TH notebook provides adequate justification of the classification of a Stuck Open PORV as a small LOCA. Subsequent to the on-site review, TVA updated the text before Figures 7-70 & 7-61 in the TH notebook (MDN-000-000-2010-205), and changed 3 hr to 5 hr for initiation of hot leg recirculation. TVA revised the success path numbers in Table 7-13 in the accident sequence notebook (MDN-000-000-2010-0201). These changes satisfy the intent of the F&O. As a result,	No	No	

Table A-3 – SQN Internal Events F&O Closure Review Consensus Table

F&O Number	Applicable SRs	F&O Text	Closure Status	Actions to Address Finding	Acceptability Evaluation	Upgrade?	New Method?	Comments
					this F&O can be closed. The resolution of this F&O meets the CC-I/II/III requirements of SR SC-C1 and improves the T/H NB documentation. The resolution of this F&O is not a PRA upgrade since it consisted of documentation corrections. Existing T/H documentation was corrected or improved as identified in the F&O and no modeling changes were performed. There were no new methods used.			
3-19	HR-H2	Section 7.2 of the HRA Notebook (MDN-000-000-2010-0204) does not explicitly discuss how the required and available manpower is addressed in the analysis. Manpower requirements are included in the operator interview checklist as item 37. However, it is not clear how this information was used in the development of the HEPs since some instances were observed where the operator interview responses were not used in the HRA calculator (see HFE HARR1).	Closed	A discussion of the required and available manpower to perform the actions and equipment manipulations was documented in sections 7.1 and 7.2 of the HRA notebook. Also, HARR1 was revised to match the operator interview for the manpower requirements.	Sections 7.1 and 7.2 of the HRA notebook (MDN-000-000-2010-0204) do indeed list manpower as one of the factors used in the HRA Calculator and one of the factors influencing execution errors. The HRA Calculator entry for event HARR1 clearly lists available manpower. The response is acceptable and this F&O may be closed. The F&O resolution meets CC-I/II/III requirements for HR-H2 because it improved documentation of the required and available manpower in the HRA NB, as well as corrected a specific identified documentation error related to manpower requirements. The resolution of this F&O is not considered to be a PRA upgrade. The finding of evidence of the use of manpower availability in the development of HEPs did not involve any new methods, did not create any new accident sequences, and did not result in a significant change in the risk results.	No	No	
3-20	HR-I1	Several issues related to the TH analyses used to support the HRA were identified. Specifically, 1) Some time windows are buried in MAAP output files which are not included in the TH Notebook and take time to review. For example, the time window for AFWOP5 is not easily available. 2) TH Notebook MDN-000-000-2010-205 Section 7.3.3 discusses the actions required following a failure of high pressure recirculation. The required action related to failure of the automatic recirculation alignment (HARR1) has two big pieces. The first is to stop the pump to avoid pump damage. If the pumps are damaged, high pressure recirculation can't be successful. The time window is short for this action and is related to RWST depletion. If the pumps are stopped on time the next action is to manually establish recirculation. The time window for that action is based on the RCS inventory depletion which is, relatively speaking, much longer. If HP recirculation is not successful, the RCS is depressurized to facilitate low pressure recirculation (AFWOP3). These two actions (HP recirculation and RCS depressurization and establish LP injection/recirculation) are for the same mitigation function. Therefore, it is unclear why there are big differences between the time windows for these two actions. In addition, the HRA Calculator input for these actions appears to be different from the descriptions in Section 7.3.3 of MDN-000-000-2010-205. 3) The use of bounding analyses for the HFEs results in non sequence specific timing information in the HRA. For example, HARR1 is used in the accident sequences after AFWS success in SSBO and SSBI accident sequences. However, the timing window of HARR1 is based on the medium LOCA and it is conservative for these sequences.	Closed	For Item 1), Timing Tables were added to each sub-section in the Thermal Hydraulics notebook (MDN-000-000-2010-205) to include separate discussions of each applicable HFE and the supporting TH analysis. An overall table was added to present the key times used for these HFEs based on MAAP analyses For Item 2), the HEPs were reviewed to determine that the time windows that are assumed were correct. HFE HARR2 is defined as the time to recover from auto-swap over failure prior to requiring placing the pumps in pull to lock and its timing estimate is based on the DBA LOCA with 2 CS trains in operation. HFE HARR1, is the time to achieve LP recirc if the HP pumps are not stopped in time and is analyzed separately for a 3" LOCA. Revised AFWOP3 discussion to clarify basis and usage. The timing for AFWOP3 in the HRA NB was made consistent with the TH NB value. For Item 3), no changes were made because the analysis used is conservative. The timing analysis is for the most time-limiting break for which the action is applied. This conservative timing selection addresses all potential scenarios/break sizes and would only reduce HEP and add additional margin to the analysis. This is considered to be appropriate due to the ranges of break sizes included in the broad bands of initiating event groupings. Evaluation of the recovery of additional margin from developing lower HEP individual analyses for each application of HARR1 will be completed in future revisions of the SQN PRA model.	The changes made to the Thermal hydraulic notebook and the HEP calculations adequately address this F&O. The conservatism identified related to HARR1 was not addressed at this time, although the response indicates that more sequence-specific values will be determined in future revision. The resolution of this F&O meets the CC-II requirements of SR HR-I1. The F&O is considered to be closed. The resolution of this F&O is not a PRA upgrade since the changes made were primarily to the HRA documentation. Existing HRA time intervals for several HEPs were reviewed and clarified and made consistent with the T/H NB as needed. In addition, the T/H NB documentation for HEP timing was improved. Some HEP value changes were made as a result of resolution of item #2 of the F&O; however, the overall impact on the PRA results should be small. In addition, no new methods were used.	No	No	- Note that Table 8.2-3 shows that AFWOP3 actions are delayed for cases AFWOP3-005, -006, -007 & -008 by 30 minutes, 70 minutes, 60 minutes & 50 minutes, respectively. This is not consistent with Table 8.2-2 which states that AFWOP3 actions are delayed for cases AFWOP3-005, -006, -007 & -008 by 40 minutes, 80 minutes, 70 minutes & 60 minutes. This comment assumes that the base case for timing is case AFWOP3-003. - Rev. 3 of the HRA NB (ca. 2013) specifically notes that HARR1 timing was changed and also states “Changed various timings in the HRAs for updated MAAP T/H analysis and updated tables 7.3.1, 7.3.2, and 10-2.” AFWOP is included in the HEPs listed in tables 7.3-1 and 7.3-2. The F&O identification of HRA timing conservatism specifically cites %1SSBO and %1SSBI. Review of QU NB identifies Sequence SSBO-6 as a 1.06% contributor to CDF; SSBO-7 also is a contributor to the Top 100 cutsets but less significant. Review of the AS NB shows that both SSBO-6 & 7 have AFW success but they also have a stuck-open Pzr PORV. That is much closer to a Medium LOCA (low end) sequence and implies that this potential source of conservatism is not significant.
3-25	AS-C1	Several documentation issues were noted. For example: 1) Sequences ISLM-008 and ISLM-017 were deleted from the ISLOCA event tree. However, there is no discussion of why this was done.	Closed	Updates were made to the Accident Sequence notebook (MDN-000-000-2010-0201) to address the identified issues. The ISLOCA sequences were re-numbered and the sequence descriptions now agree with the event tree. Note that ATWS is now documented in section 5.4.9 and no longer credits manual	This F&O primarily addresses documentation issues. These have been satisfactorily resolved. 1. Reviewed ISLOCA event tree in AS NB and sequences 8 and 17 appear to be valid.	No	No	Noted a typo in Section 5.4.9 "Step 2" discussion of Turbine Trip implies that is part of MRI and is actually part of AM. This typo has no impact on the evaluation.

Table A-3 – SQN Internal Events F&O Closure Review Consensus Table

F&O Number	Applicable SRs	F&O Text	Closure Status	Actions to Address Finding	Acceptability Evaluation	Upgrade?	New Method?	Comments
		2) Paragraphs in Section 6.4.7 need to be revised. Specifically, the first sentence in the first paragraph on page 62, starting with "If the temperature of the RCS is 557°F and dropping, the steam dumps, S/G PORVs and blowdown isolation valves are closed." needs to be finished. There is the "if" but no "then." It is also unclear how this sentence is related to the accident sequence event tree or the following statements in the paragraph related to the PORVs. The second paragraph on page 62 has grammatical errors (e.g., "…the possibility of have a RCP Seal LOCA…"). 3) The discussion of manual control rod insertion following ATWS in Section 7.9 needs to be revised to reflect the intent to remove credit for this action from the model.		rod insertion. T 1. Response states that no change was made to the ISLOCA ET and this will be fixed in upcoming revision. 2. Response states that grammatical errors have been corrected. 3. Response states that the identified section has been updated to state that only failure of automatic control rod insertion and mechanical failure of the control rods to insert are addressed by MRI.	2. Confirmed that current version is reasonable and grammatical errors have been corrected. 3. Confirmed that Table 5-12 identifies that MRI only considers failure of automatic control rod insertion and mechanical failure of the control rods to insert. The resolution of this F&O meets the CC-I/II/III requirements of SR AS-C1. This F&O is considered to be closed. The resolution of this F&O is not a PRA upgrade since it consisted of documentation corrections. There were no model or data changes, and no new methods were used.			
4-3	IFSO-A1	Non-water flood sources are excluded on the basis of Assumption 11 of the notebook. However, the Standard states (in Note 1 for this SR) that non-water sources should be considered, A more detailed basis for excluding these sources should be developed to meet the requirements of this SR.	Closed	Assumption 11 in the internal flooding notebook (MDN-000-000-2010-0203) was updated to better document the basis for only including water floods in the remainder of the analysis. All sources of fluid within the plant were analyzed for flooding considerations. However, the glycol system is the only non-water system which could have an impact on the flooding analysis. All other sources such as resin did not have enough volume to cause impact to plant operation. The glycol system also has a minimum volume, but the location of the piping, in the control rod drive rooms, causes system to be a source of spray initiating events.	The revised wording in assumption 11 adequately explains why non-water sources do not need to be evaluated. However, Table 6-3 lists various sources, such as the lube oil system, using the words "outside of the scope of the internal flooding analysis" as the basis for screening these non-water sources. This wording is incorrect (since no flood sources were out of scope) and should be updated to list which of the screening criteria defined in Section 6.2 was used. Subsequent to the on-site review, TVA corrected Table 6-3 to replace "out of scope" with "screened." A new screening criterion was added to Section 6.2 that states: Any non-water fluid source that is not the glycol system, as discussed in Assumption 11. This new criterion refers to Assumption 11 where it is documented that all non-water sources except glycol were analyzed and found to be of no consequence in the flooding analysis. The resolution of this F&O meets the CC-I/II/III requirements of SR IFSO-A1. This F&O is considered to be closed. The resolution of this F&O is not a PRA upgrade since it consisted of documentation enhancements. There were no model or data changes, and no new methods were used.	No	No	Stating in Assumption 11 that the glycol spray frequency is so small when compared to that of a transient that it is easily 'subsumed' by the transient frequency would probably clarify this response.
4-7	IFEV-B3	No discussion of sources of uncertainty associated with the flooding initiating events is currently provided in the flooding notebook (MDN-000-000-2010-0203). It is noted that the notebook includes documentation of sources of uncertainty for other portions of the flooding analysis. Sources of model uncertainty for internal flooding are also documented in MDN-000-000-2010-0209, Uncertainty And Sensitivity Analysis; however, again flood initiator uncertainties are not discussed. If no uncertainties are identified for the flood initiator frequency evaluation, then the notebook should state this to be consistent with the approach used for the IFPP, IPSO, and IFSN tasks.	Closed	Section 8.10 was added to the Internal Flooding Notebook with the following: The internal flooding frequency calculation has several different uncertainties associated with the calculation. The current model uses a summation of three different frequencies, passive pipe break failures, human induced floods, and maintenance induced flooding. Each of these flooding events has its own inherent uncertainties.	The text added to Section 8.10 of the Internal Flooding Analysis (MDN-000-000-2010-0203) appears to sufficiently address the uncertainties associated with internal flooding initiating events. However, it doesn't appear that these additional uncertainties were propagated into the overall uncertainty analysis notebook (MDN-000-000-2010-0209), appendix A. Subsequent to the on-site review, TVA added Section 5.2.10.4.1 to the Uncertainty Analysis to discuss the uncertainties associated with internal flooding initiating events. This was previously presented in the flooding analysis but not carried over into the uncertainty analysis notebook. This F&O is now closed. The resolution of this F&O meets the CC-I/II/III requirements of SR IFEV-B3. The resolution of this F&O is not a PRA upgrade since it consisted of documentation enhancements to the sources of uncertainty evaluations. There were no model or data changes, and no new methods were used	No	No	Maintenance-induced floods were defined as being those resulting from random internal ruptures of closed isolation valves (along with the exposure time required to perform the maintenance, plus possible isolation following the boundary failure. The documentation doesn't discuss flooding resulting from maintenance errors. The generic initiating event frequencies used for pipe rupture data include maintenance-related events. The EPRI Report states that "the pipe induced flood frequencies developed in this report account for all reported floods caused by human errors that involve failed or damaged piping system components." So, while human-induced flooding is included within the initiating event data, it is suggested that the next revision of the notebook include a brief discussion of human-induced floods and how they are reflected in the initiator data.
4-11	DA-C14	While the PRA model considers the possibility of two PORVs being blocked at the same time, there does not appear to have been an investigation of whether coincident maintenance can occur in the various SQN systems (or if coincident inter-system maintenance can occur). Therefore this SR is not met. It was also observed that the PORV blocking basis events noted above did not appear to be documented in either the data notebook or the appropriate system notebook.	Closed	Section 7.4.4 was added to the data notebook (MDN-000-000-2010-0203) to specifically discuss coincident maintenance unavailability. The evaluation credits the plant's System Operability Checklists as well as the Maintenance Rule a(4) risk assessment process to guard against removal of multiple SSCs that could result in high risk conditions. Appendix J in the Data NB documents the data collection and an estimate of Basic Event probability.	The additional documentation provided in section 7.4.4 and Appendix J largely address the requirements of SR DA-C14 and this F&O, however, additional information is needed to meet DA-C7 for coincident maintenance. The documentation changes made to address the F & O concern regarding PORV block valve closure meets the intent of DA-C7. Subsequent to the on-site review meeting, TVA revised section 7.4.4 of the Data notebook to provide a further discussion of the	No	No	

Table A-3 – SQN Internal Events F&O Closure Review Consensus Table

F&O Number	Applicable SRs	F&O Text	Closure Status	Actions to Address Finding	Acceptability Evaluation	Upgrade?	New Method?	Comments
					administrative processes that limit the occurrence of coincident maintenance that could have a risk impact. A review of maintenance history since 2012 was also performed to confirm that no significant instances of coincident maintenance occurred. This F&O is therefore closed. The F&O resolution meets CC I/II/III requirements for DA-C14 because it adds information that documents an examination of the specific coincident maintenance unavailability item noted in the Finding (I.e., Pzr PORVs) as well as a broader review of the governing processes and review of plant records for any potential data evidence of the practice. The resolution also satisfies the CC-II/III requirements of SR DA-C7. The resolution of this F&O is not a PRA upgrade since it consisted of documentation enhancements to validate the treatment of coincident maintenance. The changes made as part of this resolution are based on work skills such as procedure reviews and plant work records reviews that are standard for data development. There were no model or data changes, and no new methods were used.			
5-2	HR-G1	Screening analysis performed by setting the probability of some HFEs at 0.0 prevents logic from propagating properly. For example, HACI1 and HAAE1 ANDed with automatic signals are recovery actions for the signal logic. However, the HRA screening analysis sets the HEP probability to 0.0 based on an analysis that the operator action is not required. This screening approach, combined with the model structure, removes the auto actuation contribution to mitigating system failure during quantification.	Closed	For those events where 0.0s were used in the model, the fault tree was updated to remove the events so that the conflict concerning an AND gate and a zero event will no longer be encountered during normal quantification.	Review of the HRA NB confirmed that no HFEs are assigned zero values. HACI1 and HAAE1 were confirmed to be removed from the model. Review of the RR file, the flag file, and HRA recovery file confirmed that no HFEs are assigned zero values in the current fault tree model. The F&O resolution meets CC II requirements for HR-G1 because it corrected identified errors related to use of screening HEP values, as well as performed an extent of condition review to assure/correct any similar errors that might exist in the model. This F&O is considered to be closed. The resolution of this F&O is not considered to be a PRA upgrade. A simple model logic change was made to fix the identified error. Existing HEPs were reviewed to validate that there was no inappropriate usage of 0.0 for HEPs. The removal of HEPs of 0.0 did not involve any new methods, did not create any new accident sequences, and did not result in a significant change in the risk results.	No	No	
6-2	IE-C2, IE-D2	The justification for excluding plant data prior to July 2002 in the calculation of plant specific IE frequencies is not documented well enough to support IE-C2.	Closed	Plant response states that Section 7.2 of the Data NB was changed to include a discussion of the data collection period and its basis: "The data review period encompasses a total of 18.11 reactor-years of data which represents an adequate number of years to capture a good sample of recent plant data without including older data that may skew it by including data reflecting past operating and maintenance practices that may no longer apply to the plant."	Acceptable for SR IE-D2. Additional justification is needed to meet the SR IE-C2 requirements, which are "JUSTIFY excluded data that is not considered to be either recent or applicable (e.g., provide evidence via design or operational change that the data are no longer applicable)." For example, plant efforts to improve system health, performing single point vulnerability studies and other trip reduction efforts, or unavailability of prior data could be possible justifications for not looking back beyond 2002. Subsequent to the on-site review, TVA modified Section 7.2 of the initiating events analysis to explain the exclusion of historical events as required by SR IE-C2 of the ASME Standard. This F&O is now closed. The F&O resolution meets CC I/II/III requirements for both IE-C2 and IE-D2 because it added a justification statement for excluding data prior to 2002; although the statement is not very specific or detailed. The resolution of this F&O is not considered to be a PRA upgrade. The justification of the exclusion of plant data did not involve any new methods, did not create any new accident sequences, and did not result in a significant change in the risk results.	No	No	
6-3	IE-A6, IE-C10	The alignment flags in the ERCW system are not fully implemented to represent the system alignment within the	Closed	The current flag alignment for ERCW has been revised so, for the baseline model, without setting a specific configuration, the	The implementation of flags in the current model appear to be appropriate. In addition, illegal combinations of logic flags (e.g.	No	No	

Table A-3 – SQN Internal Events F&O Closure Review Consensus Table

F&O Number	Applicable SRs	F&O Text	Closure Status	Actions to Address Finding	Acceptability Evaluation	Upgrade?	New Method?	Comments
		Initiating event portion of the tree. For example, the gates under U0_AEX_G006 should contain flags to indicate which pump is running and which two pumps are not, so that the two non running pumps would have considerations for failures to start.		flag files were set to the respective time in each configuration so that a probability is now used not a true or false value.	Pump A both running and in standby) are removed using the mutually exclusive logic. However, the fault tree model file shows split fraction values for the various alignments. While TVA had used numerical split fractions at one point in their modeling, they no longer use these fractions and overwrite them with TRUE/FALSE using the master flag file. The master flag file appears to be correct and is documented in 5.2.1 of the quantification notebook. The flag settings are also documented in Table 12 of the ERCW system notebook. The fractions no longer appear in the cutsets so the quantification appears to be correct regarding the new flag events. Therefore, this F&O is judged to be closed. The resolution of this F&O meets the CC-II requirements of SR IE-A6 and the CC-I/II/III requirements of SR IE-C10. The resolution of this F&O is not a PRA upgrade. No new methods were used. Corrections were made to the alignment logic for the ERCW system were corrected, and the master flag file was updated to include all alignment settings are correct. The Loss of ERCW initiating event is a minor contributor to total risk (less than 4% for either unit). Therefore, these changes would not be expected to have a significant impact on the risk results or the risk contributors.			
6-5	IE-A6, IE-C10, SY-B3	The support system initiating event trees for the most part include provisions for common cause failures and routine system alignments. There are some discrepancies in the modeling of common cause failures in the ERCW and CCS models that require attention, however. For example: 1) While a common cause event for all 3 of the 1A, 1B, and C-S pumps failing to run exists, there are not events for the 1A and C-S pumps or the 1B and C-S pumps. 2) The structure of the ERCW tree is such that pump common cause failures could result in a pump failing due to an independent failure as well as a common cause failure in a single cutset. (See gate U0_AEX_G001) 3) The common cause initiating event group U0_ERW08POEFRI is not valid, since it is entirely based on 8760 hour exposure time for all the components. The common cause failure frequencies are therefore overestimated. The CCS tree uses a different approach than the ERCW tree for common cause initiating events. An alternate approach is also given in EPRI reports 1013490 and 1016741.	Closed	Item 1) was not yet fully addressed. For Item 2), it appears that the current structure of the ERCW fault tree (including the mutually exclusive logic) would prevent erroneous (but conservative) cutsets from being generated in the final result. For item 3) the CCF probabilities were re-calculated based on the EPRI methodology	Confirmed that failures for each pump under the identified gate are tagged as occurring either when the pump "is running" or "is in standby," so that two failures of the same pump cannot occur (e.g., in a product of a single pump failure and a CCF that includes the pump). Also, reviewed many cutsets for the initiator and found no instances of a single pump failure in a cutset with a CCF event that included the same pump; there were many instances of single pump failure in a cutset with a CCF event that included the other trains' pump but not the pump represented by the single pump failure event. Also confirmed that ERCW and CCS CCF values were calculated in Appendixes of the ERCW and CCS system NBs using information provided by EPRI 1013490. Items 2 and 3 are adequately addressed. For Item 1, the CCS system design is such that failure of the C-S CCS pump would not cause an initiating event. However, it appears that CCF failure to run for the 1A and 1B pumps should be considered. Also, because the Alpha factor CCF method is being used, all possible CCF combinations of double and triple pump failures may need to be included to properly capture the entire pump failure probability. Subsequent to the on-site review meeting, TVA modified the "living model" version of the SQN PRA to include the appropriate CCF events for the CCS pumps. The inclusion showed a negligible increase in base case risk. Per TVA PRA procedures, the living model changes (with the CCF events included) will be incorporated into the next update of the model of record. The F&O resolution meets CC II requirements for IE-A6 because the model changes incorporated address a specific CCF modeling omission that was not consistent with IE-A6. The F&O resolution meets CC I/II/III requirements for IE-C10 because the model changes incorporated address a specific CCF modeling error involving use of an 8760-hour exposure time for all components in a common cause initiating event. The F&O resolution meets CC I/II/III requirements for SY-B3 because the model changes incorporated correct an error in common cause grouping for the CCS initiating event definition. Therefore this F&O is considered closed.	No	No	While the C-S pump cannot contribute to a loss of CCS Train A, it appears that there is no CCF for failure to run of both Pumps 1A and 1B which would cause a loss of CCS Train A, and a plant trip shutdown. Addressing this F&O should consider updating the CCF development to include the 1A&1B (2A&2B) pumps. This issue can be addressed as part of the incorporation of the updated CCF modeling into the model of record.

Table A-3 – SQN Internal Events F&O Closure Review Consensus Table

F&O Number	Applicable SRs	F&O Text	Closure Status	Actions to Address Finding	Acceptability Evaluation	Upgrade?	New Method?	Comments
					The resolution of this F&O is not a PRA upgrade. Some of the changes are documentation related. Changes in the CCF events were demonstrated to have only a minimal effect on the overall results. The changes to the system initiating event trees common cause modeling correct errors in the application of techniques that have been adopted at TVA. No new methods were used.			
6-6	IE-C4	Section 5 of the IE notebook shows a Bayesian process was used to combine plant specific and generic data. However, LOCA frequencies from NUREG-1829 were also updated with plant specific data. Since the frequencies in NUREG-1829 were based on expert judgement and not actual industry data, and it is not expected that a plant would experience such an event, it does not seem appropriate to use the Bayesian update process for these events. The update did not appear to significantly alter the IE frequencies, however, so there is little impact on CDF.	Closed	No action taken to address the F & O.	IE NB treatment of industry data for initiating events obtained by expert elicitation in a consistent manner. For example, EX and Medium LOCA values are not Bayesian updated while Large LOCA, Small LOCA and SGTR values are Bayesian updated. Subsequent to the on-site review, TVA updated Section 8.1.2.3 of the initiating events analysis to reflect that excessive LOCA and medium LOCA frequencies are based on expert elicitation. Sections 8.1.4.1 and 8.1.4.4.1 were updated to reflect that Bayesian updates were not performed for excessive LOCA, medium LOCA, or large LOCA. The resolution of the F&O satisfies the Category I/II/III requirements of IE-C4 as they pertain to the Bayesian updating of LOCA frequencies. This F&O is considered to be closed. The resolution of this F&O is not considered to be a PRA upgrade. The elimination of Bayesian updates of LOCA frequencies did not involve any new methods, did not create any new accident sequences, and did not result in a significant change in the risk results.	No	No	Section 8.1.4.1 of the IE NB states: "Those prior distributions that were based only on expert elicitation are not Bayesian updated." Based on that statement, IE values for EX and Medium LOCA were not Bayesian updated in the IE NB because their values are based on expert elicitation. However, the Large LOCA IE value was Bayesian updated even though the cited reference for the value is NUREG/CR-6928 Table 8.1; NUREG/CR-6928 Table 8.1 cites Ref. 69 as its source for the Large LOCA value presented. NUREG/CR-6928 Reference 69 is NUREG-1829. NUREG-1829 is described as follows in NUREG/CR-6928: "Reference 69 is a draft report addressing LOCA frequencies for BWRs and PWRs. Frequencies were estimated using the expert elicitation process. LOCA events include large, medium, and small LOCAs and steam generator tube ruptures." These comments were resolved in the subsequent revision that was presented.
6-7	IE-C11	Section 6 of the Initiating Events Analysis, the associated system notebooks, and the HRA notebook document the use of plant-specific information in the assessment and quantification of recovery actions where available, in a manner consistent with the applicable HR SRs. An issue was noted with the ERCW initiating event tree. Event HAAEIE "OPERATOR FAILS TO START ERCW PUMP (INITIATING EVENT)" has been set to zero based on an analysis that found one pump was sufficient to cool plant loads, so if one of the two running pumps trips, operator action is not required to start another pump. Operator action to start a standby pump would be required, however, if flow was to be lost from both running pumps. The current model essentially assumes a successful operator action to start both of those pumps	Closed	The ERCW initiating event model was updated to reflect revised success criteria used in the initiating event model. Due to the change in the success criteria, the initiating event model was updated to require the failure of two running ERCW pumps as well as failure of both standby ERCW pumps to start. The HRA action HAAEIE was added to the model under the appropriate failure to start OR gate. HFE HAAEIE has a non-zero value determined using the HRA calculator.	The ERCW fault tree and system notebook (MDN-000-067-2010-0222) were reviewed. The model changes described above were made were incorporated. The concern identified in the F&O has been addressed. The F&O resolution meets CC-I/II/III requirements for IE-C11 because the model changes incorporated address a specific modeling omission that was not consistent with IE-C11. This F&O is considered to be closed. The resolution of this F&O is not an upgrade. Minor changes were made to the ERCW initiating event fault tree. The Loss of ERCW initiating event is a minor contributor to total risk (less than 4% for either unit). Therefore, these changes would not be expected to have a significant impact on the risk results or the risk contributors. In addition, no new methods were used.	No	No	Confirmed that HAAEIE value was developed in HRA NB and a non-zero value was determined. Confirmed that the CAFTA logic used HAAEIE as described.
6-10	QU-E4, QU-F4	Tables 42 and 43 of MDN-000-000-2010-0209 contain a list of modeling assumptions and their impact on the PRA model. However, the majority of items in Table 43 have an impact of 'Unknown'. Classification of model impact for these assumptions is necessary to meet this SR.	Closed	The Uncertainty and Sensitivity Analysis calculation has been updated in the following ways: Text concerning the discussion of Unknown impacts and performing a respective uncertainty analysis was removed from Section 5.0. Appendix A (formerly Table 43) was updated to provide the "Model Impact" and Assumption Dispositions.	Documentation of assumptions and uncertainties meets the intent of SR QU-F4. All assumptions listed in Appendix A (formerly Table 43) have "Model Impact" dispositions of either Low, Medium, or High. None of them is dispositioned as Unknown. The resolution of the F&O satisfies the Category I/II/III requirements of QU-E4 and QU-F4 as they pertain to identifying the effects of uncertainties on the PRA model. This F&O is considered to be closed. The resolution of this F&O is not considered to be a PRA upgrade. The documentation of uncertainty effects on the PRA model did not involve any new methods, did not create any new accident sequences, and did not result in a significant change in the risk results.	No	No	Documentation is adequate for QU-E4, but could be improved by making explicit statements about specific impacts (i.e., changing an assumption would imply increasing a BE value).
6-12	QU-F6	From the results presented in sections 5.2 and 5.7 of MDN-000-000-2010-0208, it can be inferred that the definition of significant basic event and significant accident sequence are consistent with those listed in Part 2 of the standard. This is not explicitly stated in the documentation, however. The definition of significant cutset is not provided, nor does the 100 cutset list provided in the	Closed	Definitions consistent with those in the PRA Standard were added to section 4.5 of the quantification notebook (MDN-000-000-2010-0208) for the terms "significant basic event" and "significant accident sequence".	The definitions added to the Quantification notebook for significant basic events and significant accident sequences meet the CC-I/II/III requirements of SR QU-F6. However, there is not a definition of "significant cutset" noted; however, this term does not appear to be used in the notebook. This F&O is considered to be closed.	No	No	Note that the PRA Standard roadmap incorrectly states that Section 5.0 of the notebook addresses SR QU-F6 (should be section 4.5). This is a documentation issue for the roadmap only. As noted in the F&O, it doesn't appear that the

Table A-3 – SQN Internal Events F&O Closure Review Consensus Table

F&O Number	Applicable SRs	F&O Text	Closure Status	Actions to Address Finding	Acceptability Evaluation	Upgrade?	New Method?	Comments
		documentation imply that the part 2 definition was used, as the 100 cutsets do not represent 95% of the risk.			The resolution of this F&O is not a PRA upgrade since it consisted of documentation enhancements to define the terms for significant basic events and accident sequences. There were no model or data changes, and no new methods were used.			"significant cutset" criteria was used to determine which cutsets to report, etc. The notebook should either utilize that criteria or should provide an explanation as to why the criteria wasn't used.

A.8 Identification of Key Assumptions and Uncertainties

The ASME/ANS PRA Standard [8] includes several requirements related to identification and evaluation of the impact of assumptions and sources of uncertainty on the PRA results. NUREG-1855 [11] and EPRI 1016737 [43] provide guidance on assessment of uncertainty for applications of a PRA. As described in NUREG-1855, sources of uncertainty include “parametric” uncertainties, “modeling” uncertainties, and “completeness” (or scope and level of detail) uncertainties.

- Parametric uncertainty was addressed as part of the SQN SPRA model quantification (see Section 5 of this submittal).
- Modeling uncertainties are considered in both the base internal events PRA and the SPRA. Assumptions are made during the PRA development to address a particular modeling uncertainty because there is not a single definitive approach. Plant-specific assumptions made for each of the SQN SPRA technical elements are noted in the SPRA documentation that was subject to peer review, and a summary of important modeling assumptions is included in Section 5.
- Completeness uncertainty addresses scope and level of detail. Uncertainties associated with scope and level of detail are documented in the PRA but are only considered for their impact on a specific application. No specific issues of PRA completeness were identified in the SPRA peer review.

A summary of potentially important sources of uncertainty in the SQN SPRA is listed in Table A-4.

Table A-4 Summary of Potentially Important Sources of Uncertainty

PRA Element	Summary of Treatment of Sources of Uncertainty per Peer Review	Potential Impact on SPRA Results
Seismic Hazard	The SQN SPRA peer review team noted that both the aleatory and epistemic uncertainties have been addressed in characterizing the seismic sources. In addition, uncertainties in each step of the hazard analysis were propagated and displayed in the final quantification of hazard estimates for the SQN site.	The seismic hazard reasonably reflects sources of uncertainty.

A.9 Identification of Plant Changes Not Reflected in the SPRA

The SQN SPRA reflects the plant as of the cutoff date for the SPRA, which was January 2016. All plant changes have been reviewed since the 2016 cutoff date and there are no significant plant changes subsequent to this date.

Appendix B

NRC Generic Concerns on Responses to NTTF 2.1 Seismic 50.54(f) Letter

The purpose of this Appendix is to provide a response for each of the generic observations associated with the staff's review of seismic probabilistic risk assessments (SPRA) reports provided in response to the March 12, 2012, 50.54(f) letter associated with reevaluated seismic hazards.

1. *Resolution of finding level Facts and Observations (F&Os) for internal events probabilistic risk assessment (PRA)*

   <u>NRC Observation 1:</u> The internal events PRA forms the base for the SPRA. To date the staff has ensured that internal event F&Os are resolved/closed AND that the SPRA reflects those resolutions/closures through the audit process. The staff is looking for a more efficient way of addressing this issue. For those plants that have already dispositioned the internal events F&Os and the disposition have been fed into the SPRAs, the staff believes (1) adding a statement in the cover letter transmitting the SPRA submittal that this was done, and (2) adding a statement that the findings were closed through an NRC-accepted process or pointing to docketed information providing the dispositions would obviate the need for the staff trying to determine this through the audit process.

   <u>SQN Response</u>

   The cover letter transmitting this submittal has the following statement: "The SQN internal events PRA has had all finding-level peer review F&Os being resolved and the updated internal events model has been used as the basis for the SQN SPRA.

   Section A.4 of Appendix A describes how each of the finding level SQN SPRA F&Os were closed using an NRC-accepted process. Section A.6 of Appendix A describes how each of the finding level SQN Internal Events F&Os were closed using an NRC-accepted process.

2. *Consideration of Staff Comments on Industry Documents*

   <u>NRC Observation 2 (i):</u> The staff had several comments on the industry guidance for crediting Diverse and Flexible Coping Strategies (FLEX) equipment and actions in PRAs (Nuclear Energy Institute [NEI] 16-06), which were documented in a publicly available memorandum dated May 30, 2017 (ADAMS Accession No.ML17031A269). To date the staff has used the audit process to review the credit for FLEX equipment and actions with the intent of ensuring that the credit considers those comments. The staff is looking for a more efficient way of addressing this issue and focusing its review. A potential path that can gain efficiency would be a discussion in the SPRA submittal about the specific credit for FLEX equipment and actions included in the SPRA, how the staff's comments on NEI 16-06 were appropriately considered, and the basis of as well as results from relevant sensitivity studies.

   <u>SQN Response</u>

There are 13 conclusions reached in the NRC memorandum dated May 30, 2017 (ADAMS Accession No.ML17031A269). Each of these conclusions is addressed below;

NRC Conclusion 1: NEI 16-06 has not provided accepted human reliability analysis methods for inclusion of offsite portable equipment to take quantitative risk credits in risk-informed applications that should meet the guidance of RG 1.200; therefore, claiming quantitative credits for offsite equipment is not appropriate until evaluations consistent with the guidance of RG 1.200 or improvements in the NEI guidance or state-of-art methods address the technical gaps

SQN Response to NRC Conclusion 1

No credit was taken for any portable FLEX equipment (offsite or onsite) in the SQN SPRA. Only the permanently installed FLEX diesels (6.9 kV and 480 V) were credited in the analysis.

NRC Conclusion 2: For any new risk-informed application that has incorporated mitigating strategies and should meet the guidance of RG 1.200, the licensee should either perform a focused-scope peer review of the PRA model or demonstrate that none of the following criteria is satisfied: (1) use of new methodology, (2) change in scope that impacts the significant accident sequences or the significant accident progression sequences, (3) change in capability that impacts the significant accident sequences or the significant accident progression sequences.

SQN Response to NRC Conclusion 2

This is not applicable to this submittal, however, a peer review has been performed on the SQN SPRA model that includes the permanently installed FLEX diesels. This is documented in Section A.2 of Appendix A.

NRC Conclusion 3: Licensees may incorporate mitigating strategies in PRA models after the issuance of amendments for applications that use PRA models to exercise self-approval for a plant change. For such applications, the licensee should, in addition to conforming with specific license condition(s) associated with those applications, either perform a focused scope peer review and resolve the focused scope peer-review findings before using the new models to support any risk-informed decision-making or document an evaluation demonstrating that none of the upgrade criteria is satisfied. NRC will monitor those evaluations and their documentation, along with evaluations and documents related to other items identified in this assessment, through appropriate regulatory processes (e.g., inspections).

SQN Response to NRC Conclusion 3

This is not applicable to this submittal. It is not a risk-informed application.

NRC Conclusion 4: The use of expert judgment consistent with the ASME/ANS PRA Standard as endorsed by RG 1.200 is acceptable for estimating parameter values under certain conditions and the rationale for estimated values should be documented. In reviewing future risk-informed *applications, the staff may request additional information to understand the rationale for parameter values. Using the appropriate regulatory processes, the NRC will review the rationale for parameter values added to PRA models after issuance of applications that use PRA models to exercise self-approval for a plant change.*

SQN Response to NRC Conclusion 4

This is not applicable to this submittal. It is not a risk-informed application.

NRC Conclusion 5: The NRC staff does not agree with crediting spare portable equipment not modeled in the PRA in lieu of using appropriate failure rates because this approach is not consistent with the ASME/ANS PRA Standard and RG 1.200. Furthermore, the potential impact of underestimating failure rates could be larger than the unquantified risk benefits of spare equipment not modeled in PRAs.

SQN Response to NRC Conclusion 6

This is not applicable to this submittal. It is not a risk-informed application.

NRC Conclusion 6: The failure rates of permanently installed equipment cannot be used for portable equipment even if sensitivity analyses are performed. Licensees should use plant-specific of generic data collected and analyzed using acceptable approaches to estimate the failure rates for portable equipment.

SQN Response to NRC Conclusion 6

No credit was taken for any portable FLEX equipment (offsite or onsite) in the SQN SPRA. Only the permanently installed FLEX diesels (6.9 kV and 480 V) were credited in the analysis.

NRC Conclusion 7: NEI 16-06 and risk-informed applications should address whether and how the analysis described in Supporting Requirement DA-D8 is performed.

SQN Response to NRC Conclusion 7

This is not applicable to this submittal. It is not a risk-informed application.

NRC Conclusion 8: The uncertainty associated with failure rates of portable equipment should be considered in the PRA models consistent with the ASME/ANS PRA Standard as endorsed by RG 1.200. Risk-informed applications should address whether and how these uncertainties are evaluated.

SQN Response to NRC Conclusion 8

No credit was taken for any portable FLEX equipment (offsite or onsite) in the SQN SPRA. Only the permanently installed FLEX diesels (6.9 kV and 480 V) were credited in the analysis.

NRC Conclusion 9: The NRC staff does not have access to and has not reviewed PWROG-14003. At this time, the NRC staff treats approaches proposed by that PWROG document as unreviewed methods.

SQN Response to NRC Conclusion 9

No credit was taken for any portable FLEX equipment (offsite or onsite) in the SQN SPRA. Only the permanently installed FLEX diesels (6.9 kV and 480 V) were credited in the analysis.

NRC Conclusion 10: Without any additional data or evaluations, the currently available common cause failure (CCF) parameter values should be used which should appropriately reflect the higher CCF failure rates of the portable equipment when applied to the higher independent failure rates.

SQN Response to NRC Conclusion 10

No credit was taken for any portable FLEX equipment (offsite or onsite) in the SQN SPRA. Only the permanently installed FLEX diesels (6.9 kV and 480 V) were credited in the analysis.

NRC Conclusion 11: The staff finds that using surrogates for specific actions or engineering judgement to estimate the failure probability do not adequately address the elements needed for a technically acceptable human reliability analysis described in the ASME/ANS PRA Standard (e.g., the impact of the environment under which the operators work). Until gaps in the human reliability analysis methodologies are addressed by improved industry guidance, HEPs associated with actions for which the existing approaches are not explicitly applicable, such as actions described in Sections 7.5.4 and 7.5.5 of NEI 16-06, along with assumptions and assessments, should be submitted to NRC for review.

SQN Response to NRC Conclusion 11

No credit was taken for any portable FLEX equipment (offsite or onsite) in the SQN SPRA. No credit is taken for debris removal, transportation of portable equipment, installation of equipment at a staging location, or routing of cables and hoses. Only the permanently installed FLEX diesels (6.9 kV and 480 V) were credited in the analysis. The operator actions associated with aligning and starting the FLEX diesels are not significantly different than other ex-control room operator actions associated with other permanently installed equipment. These actions are properly evaluated using the existing HRA tools (HRA Calculator).

NRC Conclusion 12: If procedures for initiating mitigating strategies are not explicit and the associated failure probabilities are not directly analyzed by accepted approaches, technical bases for probability of failure to initiate mitigating strategies should be submitted to NRC for review.

SQN Response to NRC Conclusion 12

No credit was taken for any portable FLEX equipment (offsite or onsite) in the SQN SPRA. Only the permanently installed FLEX diesels (6.9 kV and 480 V) were credited in the analysis. The procedures associated with the FLEX diesel are explicit and the associated failure probabilities are directly analyzed by accepted approaches.

NRC Conclusion 13: Until acceptable guidance is provided for identifying and assessing unique aspects of pre-initiator human failure events for mitigating strategies, the staff may request additional information regarding assessment of those human failure events.

SQN Response to NRC Conclusion 13

No credit was taken for any portable FLEX equipment (offsite or onsite) in the SQN SPRA. Only the permanently installed FLEX diesels (6.9 kV and 480 V) were credited in the analysis. There are no unique aspects of pre-initiator human failure events for the permanently installed flex diesel generators when compared to the emergency diesels.

Observation 2(ii): The staff issued a formal acceptance letter for NEI 12-13 dated March 7, 2018 (ADAMS Accession No. ML18025C022), which included specific comments. The letter stated that the use of NEI 12-13 was acceptable when supplemented by the staff's comments. To date the staff has used the audit process to ensure that the implementation of NEI 12-13 was appropriately supplemented by the staff's comments. A potential path for efficiency in this area would be a discussion in the SPRA submittal about the consideration of the staff's comments in the aforementioned acceptance letter provided such confirmation exists in the peer-review report (an excerpt from the peer-review report that states as much would also be beneficial).

SQN Response

This peer review was performed by an experienced, independent team, using the process defined in Nuclear Energy Institute (NEI) guidelines NEI-12-13 as amended by the Nuclear Regulatory Commission (NRC) on March of 2018 (ADAMS access ML18025C024 and ML18025C025).

3. Combining potential improvements during detailed screening:

   Observation 3: In alignment with the discussion in the Enclosure to letter dated September 21, 2016 (ADAMS Accession Nos. ML16237A108 [letter] and ML16237A114 [enclosure]) the staff's evaluation of each licensee's SPRA submittal includes a determination "whether additional regulatory actions are necessary (e.g., updating the design basis and structures, systems, and components (SSCs) important to safety." The staff uses guidance documents that have been developed to facilitate consistent and objective decisionmaking (ADAMS Accession No. ML17146A200). To date, in accordance with the cited guidance, the staff has engaged with the licensee to request information and insights, as necessary, as part of the audit process. A potential path for efficiency in this area would be the consideration of the enclosure and guidance document mentioned above and communication of the results therefrom in the submittal.

SQN Response

Prior to the completion of the SPRA, SQN did select plant modifications to reduce seismic risk based on IPEEE results and early SPRA quantifications. The modifications include 1) replacing the anchorage on the 480V shutdown transformers to high strength bolts and 2) installing fasteners on removable grating above the diesel generator intake and exhaust dampers. There are no further potential improvements that could be made that would significantly reduce risk without costly modifications.

The GE HEA relays (SEIS_0-30-5) fail by chatter and are a significant top risk contributor. The team has noted that these relays may be improved to a 40% increase in fragility based on experience data. However, the 40% improvement of the relays combined with a few other slight analytical improvements only reduced risk by approximately 3% when evaluated in sensitivity studies.

The next significant contributor is the 480V Reactor MOV Board which currently fails by block wall with A_m=1.4g. If block wall was modified to not fail, the functional fragility is the next governing failure mode with A_m=1.71g. The 22% increase is not expected to significantly decrease the seismic risk.

Several NSSS components are on the importance list. TVA used the best available information provided from the NSSS vendor to determine the fragility of the NSSS components. The fragility of these components may be slightly conservative since detailed analysis by the NSSS vendor was not performed.

The seismic impact on MCR instrumentation is accounted for in the fault tree logic by combining the HFEs with the hazard bin specific probability of failure of instrumentation (modeled as SEIS_HINST). The probability of instrumentation failure for each hazard bin is obtained by combining the fragilities of instrumentation failure with the fragility of the instrumentation power supply. MCR instrumentation, and therefore the cues for operator actions, is assumed to be unavailable (failure probability of 1.0) for the upper two seismic bins (%G08 and %G09). Sensitivity Studies 9 and 10 (see Table 5.7-1) showed that doubling and halving the MCR instrumentation failure probabilities had very little effect on the overall SCDF and SLERF, respectively.