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September 28, 2018
RC-18-0117

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Sir / Madam:

Subject: VIRGIL C. SUMMER NUCLEAR STATION (VCSNS), UNIT 1
DOCKET NO. 50-395
OPERATING LICENSE NO. NPF-12
FUKUSHIMA NEAR-TERM TASK FORCE RECOMMENDATION 2.1: SEISMIC
SEISMIC PROBABILISTIC RISK ASSESSMENT

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 November 2012 [ML12333A170]

3. Letter to the NRC, South Carolina Electric & Gas (SCE&G) 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 26, 2014 [ML14092A250]

4. NRC Letter, 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 No. MF3831), dated July 20, 2015 [ML15194A055]

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]

The Nuclear Regulatory Commission (NRC) issued a request for information pursuant to 10 CFR 50.54(f) regarding the Near-Term Task Force (NTTF) Recommendation 2.1 on March 12, 2012 (Reference 1). Enclosure 1 of Reference 1, Recommendation 2.1: Seismic, requested that each licensee perform a reevaluation of the seismic hazards at their sites using present-day NRC requirements and guidance to develop a Ground Motion Response Spectrum (GMRS). Additionally, licensees are requested to submit an interim evaluation and actions planned or taken to address the reevaluated hazard where it exceeds the current design basis.

Reference 2, developed by EPRI, provides industry guidance for conducting seismic evaluations as requested in Enclosure 1 of Reference 1. The SPID (Reference 2) was used to compare the reevaluated seismic hazard to the design basis hazard. In response to the 50.54(f) letter and following the guidance provided in the SPID (Reference 2), a seismic hazard reevaluation was performed (Reference 3). The reevaluation concluded that the GMRS exceeded the design basis seismic response spectrum in the 1 to 10 Hz and > 10 Hz ranges. Therefore, a seismic probabilistic risk assessment (SPRA) was required.

Reference 4 contains the NRC Staff Assessment of the VCSNS seismic hazard submittal. In this Assessment the NRC staff concluded that VCSNS's reevaluated seismic hazard is suitable for other actions associated with the NTTF Recommendation 2.1: Seismic.

Reference 5 contains the NRC letter "Final Determination of Licensee Seismic Probabilistic Risk Assessments." In that letter (Table 1 a - Recommendation 2.1 Seismic - Information Requests) the NRC instructed VCSNS to submit a SPRA by September 30, 2018.

The Enclosure of this letter contains the VCSNS SPRA in Response to the 50.54(f) Letter with regard to the NTTF 2.1 Seismic Summary Report. This Summary Report provides the information requested in Enclosure 1, Seismic Risk Evaluation Item (8)B. of Reference 1.

This letter contains no new regulatory commitments.

Should you have any questions concerning the content of this letter, please contact Michael S. Moore at (803) 345-4752.

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

9/28/18
Executed on

For
George A. Lippard

WHK/GAL/bb

Enclosure: VCSNS Seismic Probabilistic Risk Assessment Summary Report

c: without enclosure unless noted

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VIRGIL C. SUMMER NUCLEAR STATION (VCSNS)
DOCKET NO. 50-395
OPERATING LICENSE NO. NPF-12

ENCLOSURE

Virgil C. Summer Nuclear Station Seismic Probabilistic Risk Assessment in Response to 50.54(f) Letter with Regard to NTTF 2.1 Seismic

September 28, 2018
Summary Report

VIRGIL C. SUMMER NUCLEAR STATION SEISMIC PROBABILISTIC RISK ASSESSMENT IN RESPONSE TO 50.54(F) LETTER WITH REGARD TO NTTF 2.1 SEISMIC

September 28, 2018
Summary Report

VIRGIL C. SUMMER NUCLEAR STATION SEISMIC PROBABILISTIC RISK ASSESSMENT SUMMARY REPORT

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Executive Summary

In response to the 10 CFR 50.54(f) letter issued by the NRC on March 12, 2012, a seismic probabilistic risk assessment (SPRA) was performed for V.C. Summer Unit 1. The SPRA effort included performing a probabilistic seismic hazard analysis (PSHA) to develop seismic hazard and response spectra at the plant using state-of-the-art seismic source models and attenuation equations; seismic response analysis of structures, fragility analysis of structures, systems and components (SSCs); developing a logic model and performing risk quantification. The SPRA effort underwent a final peer review by a team of experts. The comments of the reviewers were addressed and incorporated into the SPRA as applicable.

The SPRA identified risk-significant sequences and SSCs with their risk rankings and showed that the point estimate seismic core damage frequency (SCDF) is 4.00×10^{-5}/yr, and the seismic large early release frequency (SLERF) is 3.65×10^{-6}/yr.

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 further reducing 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 insight 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 Nuclear Regulatory Commission (NRC) established a Near Term Task Force (NTTF) to conduct a systematic review of NRC processes and regulations and to determine if the agency should make additional improvements to its regulatory system. The NTTF developed a set of recommendations intended to clarify and strengthen the regulatory framework for protection against natural phenomena. Subsequently, the NRC issued a 50.54(f) letter on March 12, 2012 [1], requesting information to assure that these recommendations are addressed by all U.S. nuclear power plants. The 50.54(f) letter requests that licensees and holders of construction permits under 10 CFR Part 50 reevaluate the seismic hazards at their sites against present-day NRC requirements and guidance.

A comparison between the reevaluated seismic hazard and the design basis for the Virgil C. Summer Nuclear Station (VCSNS) has been performed, in accordance with the guidance in EPRI 1025287, “Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic” [2], and previously submitted to NRC [3]. That comparison concluded that the ground motion response spectrum (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 Probabilistic Risk Assessment (SPRA) has been developed to perform the seismic risk assessment for VCSNS in response to the 50.54(f) letter, specifically item (8) in Enclosure 1 of the 50.54(f) letter.

This report describes the SPRA developed for VCSNS 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 VCSNS, 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 SPRA as outlined in Section 2.0.

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

2.0 Information Provided in This Report

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

(1) The list of the significant contributors to Seismic Core Damage Frequency (SCDF) for each seismic acceleration bin, including importance measures (e.g., Risk Achievement Worth (RAW), Fussel-Vesely (F-V) and Birnbaum)
(2) A summary of the methodologies used to estimate the SCDF and Seismic Large Early Release Frequency (SLERF), including the following:
   i. Methodologies used to quantify the seismic fragilities of SSCs, together with key assumptions
   ii. SSCs 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 Probabilistic Risk Assessment (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 VCSNS Seismic Hazard Submittal [3]. Further, 50.54(f) letter Enclosure 1 paragraph 9 requests information on the Spent Fuel Pool. This information was submitted separately [54].

Table 2-1 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 [2] defines the principal parts of an SPRA, and the VCSNS SPRA has been developed and documented in accordance with the SPID. The main elements of the SPRA performed for VCSNS in response to the 50.54(f) Seismic letter correspond to those described in Section 6.1.1 of the SPID, specifically:

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

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

The VCSNS SPRA and associated documentation has been peer reviewed against the ASME/ANS PRA Standard in accordance with the process defined in NEI 12-13 [5], as documented in the VCSNS SPRA Peer Review Report. The VCSNS SPRA, complete SPRA documentation, and details of the peer review are available for NRC review.

This submittal provides a summary of the SPRA development, results and insights, and the peer review process and results, 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 3.0 provides information related to the VCSNS seismic hazard analysis.

Section 4.0 provides information related to the determination of seismic fragilities for VCSNS SSCs included in the seismic plant response.

Section 5.0 provides information regarding the plant seismic response model (seismic accident sequence model) and the quantification of results.

Section 6.0 summarizes the results and conclusions of the SPRA, including identified plant seismic issues and actions taken or planned.

Section 7.0 provides references.

Section 8.0 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 VCSNS SPRA peer review.

Table 2-1 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.	Section 5.0.
2	Summary of the methodologies used to estimate the SCDF and LERF.	Sections 3.0, 4.0, 5.0.
2i	Methodologies used to quantify the seismic fragilities of SSCs, together with key assumptions.	Section 4.0.
2ii	SSC fragility values with reference to the method of seismic qualification, the dominant failure mode(s), and the source of information.	Table 5.4-3 provides fragilities (A_m and β), failure mode information, and method of determining fragilities for the top risk significant SSCs based on standard importance measures such as Fussel-Vesely (F-V) or risk reduction worth (RRW). Seismic qualification reference is not provided as it is not relevant to development of SPRA.
2iii	Seismic fragility parameters.	Table 5.4-3 provides fragilities (A_m and β) information for the top risk significant SSCs based on standard importance measures such as F-V or RRW
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 internal events PRA model to produce the seismic PRA model and their motivation.	Sections 5.1 and 5.3 provide this information.
2vi	Assumptions about containment performance.	Sections 4.3 and 5.5 address containment and related SSC performance.

Table 2-1 Cross-Reference for 50.54(f) Enclosure 1 SPRA Reporting

50.54(f) Letter Reporting Item	Description	Location in this Report
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.
4	Identified plant-specific vulnerabilities and actions that are planned or taken.	Section 6.0 addresses this.

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

SPID Section 6.8 Item (1) Description	Location in this Report
A report should be submitted to the NRC summarizing the SPRA inputs, methods, and results.	Entirety of the submittal 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 submittal addresses this and identifies key methods of analysis and referenced codes and standards.
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 submittal 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 this submittal addresses this.
It is not necessary to submit all of 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.	Entire 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 Standard [4]. 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 VCSNS Seismic Hazard and Plant Response

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

The VCSNS site is located in Fairfield County, South Carolina, approximately 15 miles southwest of the county seat of Winnsboro and 26 miles northwest of Columbia, the state capital. The reactor building is located at latitude N34°17’54.1" and longitude W81°18’54.6" [14].

The site is underlain by a complex sequence of metamorphic and igneous rock on the Charlotte Belt metamorphic zone. The overburden soils at the site are primarily residual, derived by the chemical weathering of the underlying metamorphic and igneous rock [14]. As discussed in [14], prior to and during construction, subsurface field investigations that included test borings and geophysical surveys were performed at the site. The results of the investigations indicate that the plant site and surrounding area were initially blanketed primarily by moderately thick residual soil derived by weathering of underlying rock. Many borings indicate that the overburden soils are underlain by a zone of highly weathered rock which sometimes is interlayered with decomposed rock of a granular soil-like consistency. Moderately weathered rock usually is present beneath these materials, and is directly beneath the overburden soils where the highly weathered rock zone does not occur. This, in turn, is underlain by fresh rock which contains some random thin zones of weathered and/or partially decomposed rock. Moderately weathered and/or fresh rock was encountered in borings at the principal plant structures at depths.

Extensive excavations to and into bedrock were made for plant structures. After the nuclear plant site was cleared, grubbed, stripped of topsoil and organic material, and graded to finish grade elevation of 435', excavations were made for the foundation mats of the Seismic Category 1 structures including the Reactor, Control and Auxiliary Buildings. These excavations extended into rock. The competency of the bearing rock was evaluated by geologic and engineering inspections during construction. Weathered or highly fractured rock was removed. Rock which was excessively fractured by blasting, or which failed to meet the minimum requirement for the Safe Shutdown Earthquake (SSE) and Operating Basis Earthquake (OBE) for the design of Seismic Category 1 structures supported on mat foundations founded on rock, was removed. After the foundation rock was inspected and approved, the excavations were backfilled with fill concrete [14].

Dynamic and static engineering properties of the foundation bedrock are presented in [14]. The shear wave velocity of the foundation sound rock is estimated approximately 9200 ft/sec using the compressional wave velocity and Poisson ratio provided in [14]. Moreover, the shear wave velocity for the foundation rock for Seismic Category 1 structures (including Reactor, Control and Auxiliary Buildings) is given as 10,000 ft/sec in [15]. On this basis, the site is categorized as a “hard rock” site, for purposes of developing seismic ground motions.

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 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.

The seismic hazard analysis is performed for the power block control point which is the foundation of the Seismic Category 1 structures (including Reactor, Control and Auxiliary Buildings). As indicated in the Section 3.0, the VCSNS power block is founded on hard-rock. The seismic hazard analysis for the VCSNS site is performed for the hard-rock condition using the 2012 Central-Eastern United States Seismic Source Characterization (CEUS-SSC) [16] and 2013 EPRI ground-motion models (GMMs) [17]. The 2013 EPRI GMMs are applicable to hard-rock conditions in the central and eastern United States (CEUS) defined as shear-wave velocities greater than 2.8 km/s or 9200 ft/s. Therefore, the 2013 EPRI GMMs without any further adjustments for site effects are used to calculate the seismic rock hazard at the VCSNS site.

Detailed information regarding the VCSNS site hazard was provided to NRC in the seismic hazard information submitted to NRC in response to the NTTF 2.1 Seismic information request [3]. That information was used in development of the VCSNS SPRA.

3.1.1 Seismic Hazard Analysis Methodology

For the VCSNS SPRA, the following method was used:

As reported in the VCSNS NTTF 2.1 Seismic Hazard submittal [3], the control point (power block) mean and fractile rock hazard curves were calculated at seven spectral frequencies of 0.5 Hz, 1.0 Hz, 2.5 Hz, 5.0 Hz, 10.0 Hz, 25.0 Hz, and 100.0 Hz, at which EPRI GMMs are available. The mean rock hazard curves were used to develop uniform hazard response spectra (UHRS) and the ground motion response spectrum (GMRS). The smooth UHRS were calculated using log-log interpolation from hard-rock spectral shapes to determine the spectral acceleration at each spectral frequency for the mean annual frequencies of exceedances (MAFEs) of 10^{-4} and 10^{-5}. The GMRS was calculated from the 10^{-4} and 10^{-5} UHRS at each spectral frequency. The site-specific ground motions were developed for a surface outcrop of the hard bedrock.

A set of 100 discrete hazard curves for PGA were calculated using the logic tree end branch hazard curves. The reduction of hazard curves down to 100 hazard curves is accomplished with an algorithm that uses a range of logarithmic accelerations that replicate the mean and uncertainty in ground motion at selected annual frequencies of exceedance (AFEs) of 10^{-4}, 10^{-5}, and 10^{-6}. The 100 discrete hazard curves are provided in [18].

The horizontal rock mean UHRS at MAFEs of 10^{-4} and 10^{-5} and the GMRS at the VCSNS site calculated for the SPRA are shown in Figure 3.1-1.

The methodology for obtaining the vertical response spectra is discussed in Section 3.1.4. Additional details regarding the Seismic Hazard Analysis Methodology are included in [18].

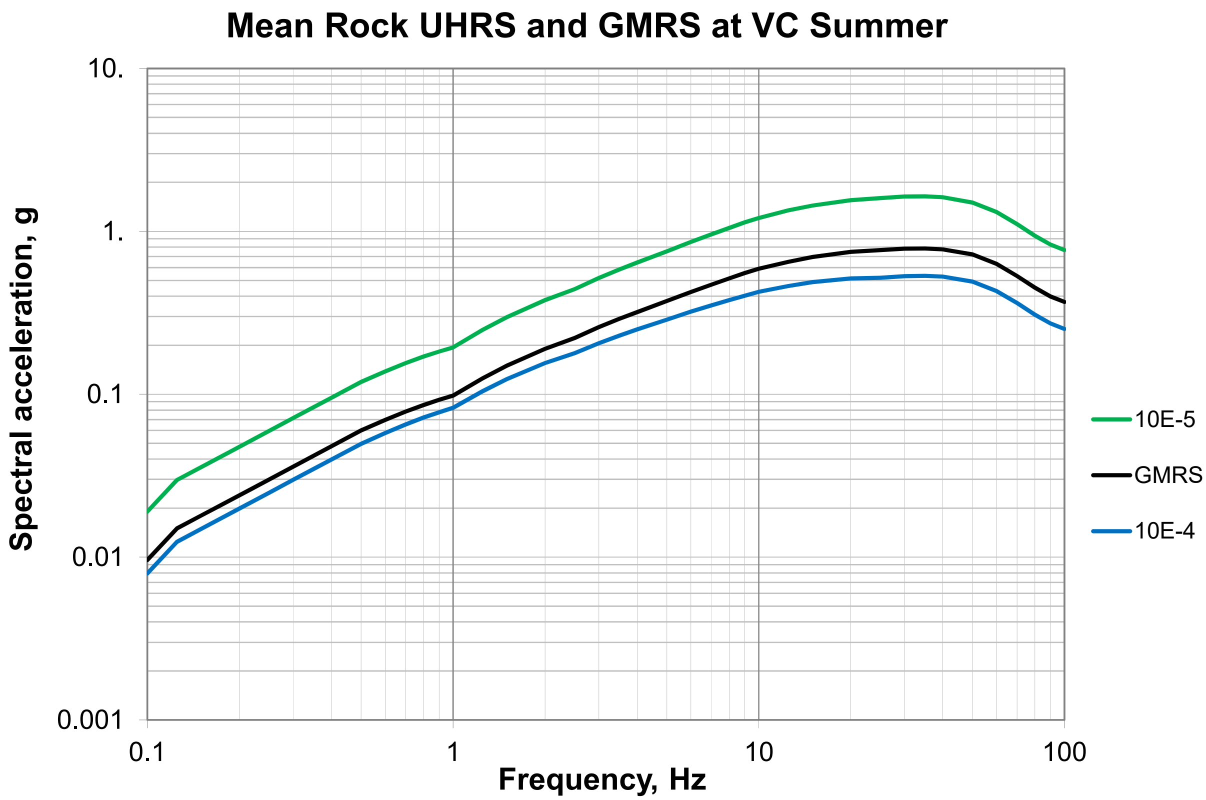

Figure 3.1-1 VC Summer Mean Horizontal Rock UHRS for MAFEs of 10^{-4} and 10^{-5} and GMRS

3.1.2 Seismic Hazard Analysis Technical Adequacy

The VCSNS SPRA hazard methodology and analysis associated with the horizontal GMRS were submitted to the NRC as part of the VCSNS Seismic Hazard Submittal [3], and found to be technically acceptable by NRC for application to the VCSNS SPRA [19].

The VCSNS hazard analysis was also subjected to an independent peer review [46] against the pertinent requirements in the ASME/ANS PRA Standard. The SPRA was peer reviewed relative to Capability Category II for the full set of requirements in the Standard and determined to be acceptable for use in the SPRA.

The peer review assessment, and subsequent disposition of peer review findings, are described in Appendix A.

3.1.3 Seismic Hazard Analysis Results and Insights

This section provides the final seismic hazard results used in the VCSNS SPRA.

The following main assumptions were made in the seismic hazard analysis:

- The specific background and repeated large-magnitude earthquake (RLME) sources determined in [16] and included in the seismic hazard calculation and the manner in which they are modeled characterize the hazard at the VCSNS site. No specific seismic sources other than those identified in [16] have been identified in the region of the site that would affect seismic hazard. We note that the Eastern Tennessee Seismic Zone (ETSZ) has been identified and studied by many researchers (e.g., [20]) as a zone of seismicity, not a specific seismic source. The ETSZ lies ~200+_km northwest of the site. The largest historical earthquake in the ETSZ had moment magnitude **M** 4.8 [20], and proposed maximum earthquakes for the ETSZ are **M**>6.0 [20]. The sources identified in [16] accommodate seismicity in the ETSZ by replicating historical rates of seismicity and by using $\mathbf{M}_{max}$ distributions exceeding **M** 5.9.
- The CEUS-SSC earthquake catalog [16] and updated version [22] for select background sources characterize the regional seismicity. New seismicity since the compilation of the CEUS-SSC catalog was evaluated for the nearby Vogtle power plant site, and it was concluded that an update to the activity rates was not necessary [21]. Because the VCSNS site lies in the same region as the study

for the nearby Vogtle site, it was concluded that updated seismic activity rates in the region would not affect seismic hazard for the VCSNS site.

- It is assumed that the mean rock UHRS at MAFEs of 10^{-4} and 10^{-5} are suitable to calculate vertical GMRS at the VCSNS site.

Table 3.1-1 provides the final seismic hazard used as input to the VCSNS SPRA, in terms of exceedance frequencies as a function of PGA level for the mean and several fractiles. Information on the vertical hazard is discussed in Section 3.1.4.

Table 3.1-1 VCSNS Mean and Fractile Exceedance Frequencies

	Exceedance Frequencies (/y)			
PGA (g)	**0.16**	**0.5**	**MEAN**	**0.84**
0.1	1.31E-04	3.09E-04	5.20E-04	8.23E-04
0.15	6.26E-05	1.53E-04	2.63E-04	3.95E-04
0.3	1.72E-05	4.37E-05	7.23E-05	1.05E-04
0.5	5.75E-06	1.64E-05	2.55E-05	3.95E-05
0.75	2.19E-06	6.83E-06	1.05E-05	1.72E-05
1	1.01E-06	3.42E-06	5.38E-06	8.98E-06
1.5	2.92E-07	1.15E-06	1.93E-06	3.23E-06
2	9.85E-08	4.49E-07	8.27E-07	1.36E-06
3	2.13E-08	1.20E-07	2.50E-07	4.01E-07

Uncertainties in the PSHA result from uncertainties in input models and parameters. The contributions of different parameters of the logic tree to the uncertainty in seismic hazard are investigated for MAFEs of 10^{-4} and 10^{-5} at 1 Hz and 10 Hz [18].

For 10 Hz spectral acceleration (SA) hazard, background sources were found to have the dominant contribution to hazard at amplitudes of interest, which is a common result for close, moderate sized (background) earthquakes. For 1 Hz SA hazard, the Charleston RLME source has a large contribution to hazard at amplitudes of interest, and this is a common result for sites at which large, distant RLME earthquakes contribute to low frequency hazard.

Regarding uncertainty in hazard, the main contributors are the EPRI GMMs used for hazard calculations, and the characteristic magnitude of the Charleston RLME source. The GMMs contribute to uncertainty at both high and low spectral frequencies, at spectral amplitudes corresponding to MAFEs of 10^{-4} and 10^{-5}. The characteristic magnitude of the Charleston source contributes to uncertainty primarily for low spectral frequencies at spectral amplitudes corresponding to MAFEs of 10^{-4} and 10^{-5}, for which the Charleston source has an important contribution to hazard.

In the SPRA plant model, described in Section 5, the hazard data in Table 3.1-1 was discretized into 12 intervals, with parameters as listed in Table 3.1-2.

Table 3.1-2 Acceleration Intervals and Interval Frequencies as Used in SPRA Model

Interval Designator	Interval Lower Bound (g)	Interval Upper Bound (g)	Representative Magnitude PGA (g)	Interval Mean Frequency (/yr)
GA	0.100	0.225	0.15	3.93E-04
GB	0.225	0.350	0.28	7.47E-05
GC	0.350	0.475	0.41	2.42E-05
GD	0.475	0.600	0.53	1.13E-05
GE	0.600	0.725	0.66	5.82E-06
GF	0.725	0.850	0.79	3.50E-06
GG	0.850	0.975	0.91	2.14E-06
GH	0.975	1.100	1.04	1.44E-06
GI	1.100	1.225	1.16	1.01E-06
GJ	1.225	1.350	1.29	7.23E-07
GK	1.350	1.475	1.41	5.19E-07
GL	>1.475	-	1.62	2.02E-06

3.1.4 Horizontal and Vertical GMRS

This section provides the control point horizontal and vertical GMRS.

The mean vertical UHRS at MAFEs of 10^{-4} and 10^{-5} are calculated by scaling those horizontal spectra by appropriate vertical-to-horizontal (V/H) ratios. The vertical GMRS is calculated using the obtained vertical spectra at MAFEs of 10^{-4} and 10^{-5}.

The V/H ratios from NUREG/CR-6728 [23] for the hard-rock sites in the CEUS, as recommended by Reg. Guide 1.208 [24], were used for to calculate vertical UHRS at the VCSNS site. The V/H ratios from [23] are given for three ranges in rock outcrop horizontal component PGA. The PGA values from horizontal UHRS at MAFEs of 10^{-4} and 10^{-5} were used to determine which V/H ratios to apply to each spectrum. PGA was used as a substitute for magnitude and distance which "are average of empirical relations" [23] and representative of nearby earthquakes. The V/H ratios at the 13 frequencies between 0.01 and 100 Hz, given in [23], are interpolated for the 40 frequency values in Table 3.1-3. The V/H ratios for MAFEs of 10^{-4} and 10^{-5} are applied to horizontal UHRS to calculate vertical UHRS. The vertical GMRS are calculated using vertical 10^{-4} and 10^{-5} UHRS. This is a more straightforward way to calculate the vertical FIRS than attempting to derive a V/H ratio for the GMRS.

The horizontal and vertical GMRS along with the V/H ratios are given in Table 3.1-3 and shown in Figure 3.1-2.

Table 3.1-3 Horizontal and vertical GMRS and V/H ratios for MAFEs of 10^{-4} and 10^{-5}

Frequency (Hz)	Horizontal GMRS (g)	10^{-4} V/H	10^{-5} V/H	Vertical GMRS (g)
100	3.68E-01	1.00	1.30	4.54E-1
90	3.98E-01	1.04	1.36	5.12E-1
80	4.51E-01	1.09	1.43	6.12E-1
70	5.31E-01	1.13	1.50	7.51E-1
60	6.31E-01	1.14	1.52	9.03E-1
50	7.22E-01	1.12	1.50	1.02E+0
40	7.77E-01	1.04	1.41	1.03E+0
35	7.86E-01	0.98	1.32	9.80E-1
30	7.84E-01	0.94	1.23	9.10E-1
25	7.67E-01	0.88	1.12	8.18E-1
20	7.47E-01	0.83	1.04	7.40E-1
15	6.96E-01	0.79	0.97	6.47E-1
12.5	6.51E-01	0.77	0.94	5.87E-1
10	5.89E-01	0.75	0.90	5.11E-1
9	5.53E-01	0.75	0.90	4.80E-1
8	5.14E-01	0.75	0.90	4.46E-1
7	4.71E-01	0.75	0.90	4.09E-1
6	4.24E-01	0.75	0.90	3.68E-1
5	3.73E-01	0.75	0.90	3.24E-1
4	3.20E-01	0.75	0.90	2.77E-1
3.5	2.90E-01	0.75	0.90	2.52E-1
3	2.58E-01	0.75	0.90	2.24E-1
2.5	2.22E-01	0.75	0.90	1.92E-1
2	1.90E-01	0.75	0.90	1.65E-1
1.5	1.50E-01	0.75	0.90	1.30E-1
1.25	1.25E-01	0.75	0.90	1.09E-1
1	9.81E-02	0.75	0.90	8.52E-2
0.9	9.26E-02	0.75	0.90	8.03E-2
0.8	8.59E-02	0.75	0.90	7.46E-2
0.7	7.83E-02	0.75	0.90	6.79E-2
0.6	6.96E-02	0.75	0.90	6.04E-2
0.5	6.00E-02	0.75	0.90	5.20E-2
0.4	4.80E-02	0.75	0.90	4.16E-2
0.35	4.20E-02	0.75	0.90	3.64E-2
0.3	3.60E-02	0.75	0.90	3.12E-2

Table 3.1-3 Horizontal and vertical GMRS and V/H ratios for MAFEs of 10^{-4} and 10^{-5}

Frequency (Hz)	Horizontal GMRS (g)	10^{-4} V/H	10^{-5} V/H	Vertical GMRS (g)
0.25	3.00E-02	0.75	0.90	2.60E-2
0.2	2.40E-02	0.75	0.90	2.08E-2
0.15	1.80E-02	0.75	0.90	1.56E-2
0.125	1.50E-02	0.75	0.90	1.30E-2
0.1	9.59E-03	0.75	0.90	8.33E-3

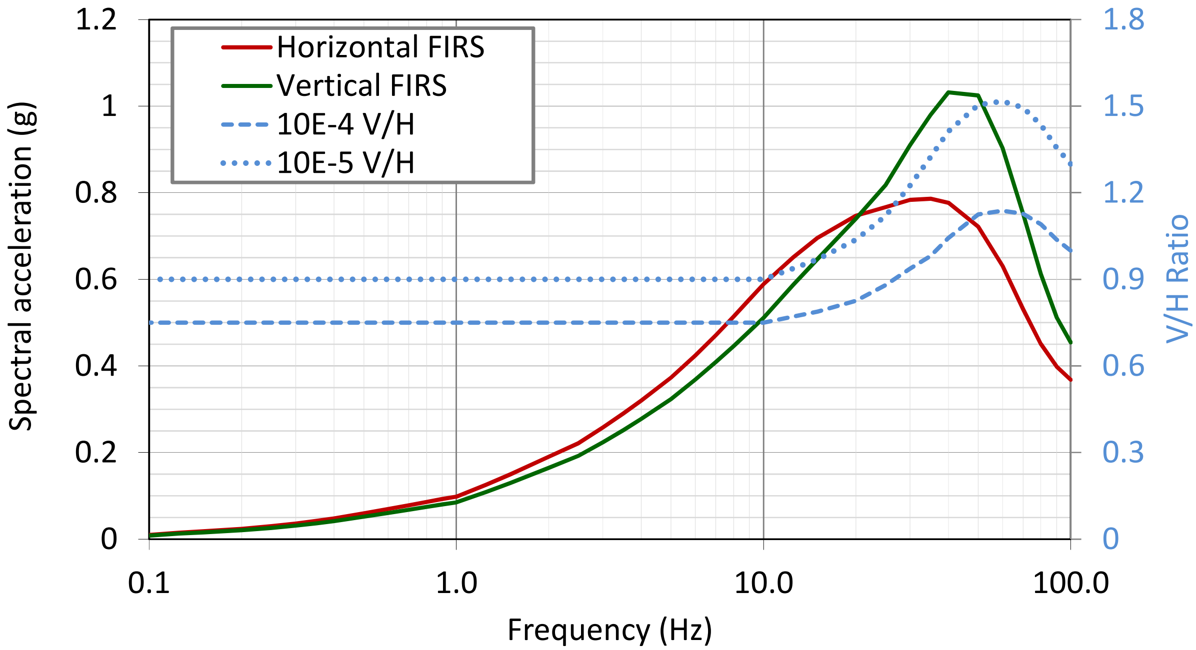

Figure 3.1-2 Plot of the Horizontal and Vertical Ground Motions Response Spectra and V/H Ratios

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 VCSNS SPRA. The subsections provide brief summaries of these elements.

4.1 Seismic Equipment List

A seismic equipment list (SEL) was developed that includes those SSCs whose seismic-induced failure could either give rise to an initiating event or degrade capability to mitigate a seismically induced initiating event. The SEL was developed for the end states of core damage and large early release. The methodology used to develop the SEL is generally consistent with the guidance provided in the EPRI SPRA Implementation Guide [10].

4.1.1 SEL Development

A preliminary SEL was developed based on seismic relevant portions of the internal events PRA model. This preliminary SEL was then supplemented by a series of reviews intended to identify seismically risk-significant components not modeled by the internal events PRA.

Based on the internal events PRA and review of other potential seismic initiators, the primary seismic initiators identified were Direct Core Damage and Large Early Release, Direct Core Damage, Interfacing Systems LOCA, Large LOCA, Medium LOCA, Small LOCA, Secondary Side Break Outside Containment, Support System Initiating Events (%LACA1, %LCC1, %LDCA1, %LDCB1, %LSW1), Loss of Offsite Power, and Very Small LOCA. The scope of the SPRA is power operation, therefore low power and shutdown states were not considered. For each of these initiating events, the plant SSCs whose seismic-induced failure could cause the initiating event were identified. The initial SEL was appended with these SSCs if they were not already included.

The internal events PRA was reviewed to identify all basic events associated with equipment failures that could be seismically-induced, and that are required to mitigate seismic-induced initiating events. A screening of all basic events was performed. Events surviving this screening process were assigned component IDs, which are later used for the development and mapping of fragilities.

The SEL was then supplemented by the following series of reviews intended to identify potential seismically risk-significant components not modeled by the internal events PRA:

- Review of distributed systems
- Plant parameter instrumentation review
- Interfacing systems loss of coolant accident review
- Plant structures review
- Flow diversion review
- Seismic-induced fire review
- Seismic-induced flood review
- Individual Plant Examination for External Events (IPEEE) review
- Seismic Walkdown Equipment List (SWEL) review

The plant structures that house SEL equipment include the following:

- Auxiliary Building (AB)
- Control Building (CB)
- Diesel Generator Building (DB)
- Fuel Handling Building (FB)
- Intermediate Building (IB)
- Reactor Building (RB)
- Service Water Pumphouse (SW)
- Condensate Storage Tank (CST)

The following types of equipment were not included on the SEL based on their having very high seismic capacity, and their passive nature:

- Check valves
- Manual valves and dampers
- Small passive in-line components supported by piping or ducting

The “rule-of-the-box” was applied to cases where components are housed within or otherwise integral to a larger component. In such cases the sub-component was not explicitly added to the SEL, provided that the containing component was already included on the SEL and its fragility considers equipment inside the “box.”

Fragilities are later assigned to components associated with unscreened basic events. Each component is assigned either to a detailed fragility performed for that component, or to a surrogate fragility group based on seismic class, building location, and/or elevation. Fragilities were initially mapped to all unscreened failures modes modeled by the internal events PRA for a given component. Refinement of this initial modeling is performed as needed based on risk significant components identified during model quantifications.

The internal events PRA applies a number of criteria to determine which containment penetrations to explicitly model and which to screen. The internal events screening criteria are considered applicable to the SPRA. Additionally, small line size penetrations were evaluated. All of the penetrations screened from the internal events PRA based on line size have been reviewed and are similarly screened from the SPRA.

The resulting SEL applicable to walkdowns includes approximately 710 items, not counting rule-of-the-box components.

4.1.2 Relay Evaluation/Spurious Breaker Trip Evaluation

During a seismic event, vibratory ground motion can cause relays to chatter. The chattering of relays potentially can result in spurious signals to equipment. Most relay chatter is either acceptable (does not impact the associated equipment), is self-correcting, or can be recovered by operator action. An extensive relay chatter evaluation was performed for the VCSNS SPRA, in accordance with SPID [2] Section 6.4.2 and ASME/ANS PRA Standard [4] Section 5-2.2. Either functional screening or fragility analysis was performed for relays with the potential to impact SEL component functions. The evaluation resulted in most relay chatter scenarios screened from further evaluation based on no significant impact to component function. Table 4.1-1 lists relays with significant contribution to risk, along with their function and disposition in the SPRA with appropriate seismic fragility or operator action.

An evaluation of spurious trips of breakers was also performed for low and medium voltage switchgear. The functionality of breakers was evaluated through evaluation of the site-specific seismic qualification testing performed for design basis and verification of EPRI NP-5223-SL [9] generic equipment ruggedness spectra (GERS) caveats. The major types of breakers at the plant are air breakers (medium voltage), draw-out type breakers (low voltage), and molded case circuit breakers. Molded case circuit breakers inherently have high seismic capacity. The fragilities for switchgear which house breakers are evaluated through EPRI NP-6041-SL [7] conservative, deterministic failure margin (CDFM) criteria. The seismic capacities used in this evaluation are based on the site-specific qualification testing and EPRI generic equipment ruggedness spectra (GERS) in EPRI NP-5223-SL [9]. This evaluation addresses fragility for high frequency sensitive components as discussed in Section 6.4.2 of the SPID [2].

Table 4.1-1 lists relays with significant contribution to risk, along with their function and disposition in the SPRA with appropriate seismic fragility. Note however that Table 4.1-1 lists only risk significant relay groups, and the SPRA includes a very large volume of additional relay impacts that are modeled explicitly but do not contribute significantly to seismic risk.

Table 4.1-1 Summary of Disposition of Risk Significant Relays

Relay Group	Function	Treatment
Relay_0.11AC, Relay_0.11BD	Affects emergency diesel generators and powering of 7.2 kV safety-related switchgear following loss of offsite power	Fragility analysis performed and incorporated explicitly into the SPRA model
Relay_0.52	Chatter within solid state protection system broadly affecting automatic-actuation important mitigating equipment	Fragility analysis performed and incorporated explicitly into the SPRA model
Relay_0.14D	Affects containment isolation valve XVG-6066	Fragility analysis performed and incorporated explicitly into the SPRA model

4.2 Walkdown Approach

This section provides a summary of the methodology and scope of the seismic walkdowns performed for the SPRA [11]. Walkdowns were performed by personnel with appropriate qualifications as defined in EPRI NP-6041-SL [7] Section 2 and the requirements in the ASME/ANS PRA Standard [4] Section 5-2.2. Each seismic review team (SRT) utilized for the SPRA included seismic engineers with extensive experience in fragility assessment and seismic walkdown training. Walkdowns of those SSCs included on the seismic equipment walkdown list were performed to assess the as-installed condition of these SSCs for use in determining their seismic capacity and performing initial screening, to identify potential II/I spatial interactions and to look for potential seismic-induced fire/flood interactions. The seismic walkdowns were performed in accordance with the criteria provided in EPRI NP-6041-SL [7].

The information obtained was used to provide input to the fragilities analysis and SPRA modeling (e.g., regarding correlation and rule-of-the-box considerations).

The seismic walkdowns were conducted on all accessible SEL equipment including equipment inside the Reactor Building. The seismic walkdowns included the evaluation of seismic interactions, including the effects of seismic-induced fires and flooding. The SPRA walkdowns were performed in parallel with walkdowns for the Expedited Seismic Evaluation Program (ESEP) [12]. Walkdown information was applied for both ESEP and the SPRA.

In addition to evaluating individual components and associated systems on the SEL, the walkdown reviewed the fire protection system. The fire protection piping was found to be well supported and not susceptible to anchorage failures.

A concern for seismic-induced fires is from flammable gases and liquids. Thus, the walkdown included these sources and their proximity to components on the SEL. Potential fires due to hydrogen piping in SEL buildings and transformers in SEL buildings are examples of scenarios that were evaluated.

The potential for seismically-induced flooding was also evaluated. During the walkdowns, potential spray and flooding scenarios from piping systems and SEL components were reviewed. Flood sources, including the fire-protection system, were evaluated. Identified scenarios that were evaluated for seismic fragility include the following:

- Anchorage failure of Control Building Fire Protection Hose Station cabinet with potential to produce spray and flooding.
- Failure of Intermediate Building Fire Service piping at threaded joints with potential to produce flooding.
- Failure of Diesel Generator Building Fire Service piping at cross tie to Service Water piping.

4.2.1 Significant Walkdown Results and Insights

Consistent with the guidance from EPRI NP-6041-SL [7], no significant findings were noted during the VCSNS seismic walkdowns. A limited number of findings related to EPRI NP-6041-SL [7] or GERS [9] screening caveats were identified for disposition under ESEP program [12] and were addressed through the plant corrective action process. For the SPRA, implementation of modifications was verified and credited in fragility analyses as appropriate.

Components on the SEL were evaluated for seismic anchorage and interaction effects in accordance with SPID [2] guidance and ASME/ANS PRA Standard [4] requirements. The walkdowns also assessed the 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 seismic-induced fire and flooding scenarios were assessed, and potential internal fire and flood scenarios were incorporated into the VCSNS SPRA model. The walkdown observations were used in developing the SSC fragilities for the SPRA.

4.2.2 Seismic Equipment List and Seismic Walkdowns Technical Adequacy

The VCSNS SPRA SEL development and walkdowns were subjected to an independent peer review against the pertinent requirements (i.e., the relevant SFR and SPR requirements) in the ASME/ANS PRA Standard [4].

The peer review assessment, and subsequent disposition of peer review findings, is described in Appendix A, and establishes that the VCSNS 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 Soil Structure Interaction Analyses.

4.3.1 Fixed-base Analyses

The VCSNS site is a hard rock site with a shear wave velocity of approximately 10,000 feet/second for foundation rock [13]. This meets the SPID [2] Section 6.3.3 criterion for fixed-base analysis. Major structures within the power block area are founded on hard rock. The Service Water Pumphouse is outside the power block area and is founded on soil and is discussed below. Relevant foundation conditions and analysis options for structures are listed in Table 4.3-1 [26].

Fixed-base analyses were performed for the Control Building, Intermediate Building and Diesel Generator Building. New detailed 3-dimensional models were created for the Control Building and Intermediate Building as discussed below. The Diesel Generator Building is founded on piles that are socketed into hard rock. The existing design-basis lumped-mass stick model (LMSM) for the Diesel Generator Building was enhanced and coupled to a new 3-dimensional finite element model of the pile foundation.

ANSYS was used for fixed-base analysis. Natural modes analysis was first performed to obtain natural frequencies and mode shapes. Mode superposition transient analysis was then performed using imposed rock time history data. The in-structure response spectra (ISRS) for structures considered in the seismic PRA were developed using the time-history analysis results. Both horizontal and vertical ISRS were computed from time-history motions at various floors or other important locations. Incoherence of input ground motions was not considered in the fixed-base response analysis.

The Auxiliary Building and Reactor Building are founded on hard rock and fixed-base models are acceptable for those buildings. As an enhancement, analysis was performed for each using the EKSSI program described below so the effect of rock foundation compliance on response could be investigated. Building responses were found to be equivalent to fixed-base analysis with some reduction in high-frequency response. The effects of soil embedment were investigated and found to be insignificant and no soil effects were included. Variation of rock stiffness was investigated and was found to have no significant effect on response.

Upper Bound (UB) and Lower Bound (LB) stiffness cases were analyzed by imposing an assumed ±15% natural frequency variation on the fixed-base structure models. The 15% variation was per ASCE 4-13 [29] Section 6.2.3.

Ground motion input was based on the site-specific 1E-5 Uniform Hazard Response Spectrum (UHRS) shapes anchored to a 0.70g horizontal peak ground acceleration (PGA). A single set of artificial time histories were generated based on the criteria presented in ASCE 43-05 [30] Section 2.4 and NUREG-0800, Section 3.7.1 [31] Acceptance Criterion 1B, Option 1 Approach 2. Additional checks were performed for statistical independence, strong-motion duration, power spectral density and Arias intensity. The checks ensure resulting time histories are suitable without any deficiencies of power across the frequency range of interest. The adequacy of the artificial set over the applicable range of structure damping was verified by comparing to the response spectra for 5 sets of time histories generated from real earthquake seeds and spectrally matched to the site-specific UHRS [26] anchored to a 0.70g horizontal PGA.

The generated ISRS provide 84% NEP ISRS and are appropriate for CDFM analysis per guidance of EPRI NP-6041-SL [7] Section 2 and EPRI 1019200 [32] Appendix A. In addition, a series of sensitivity studies were performed for rock founded structures to support median-centered fragility analysis [26]. Sensitivity studies included analysis of the Reactor Building for 5 sets of time histories generated from real earthquake seeds and analysis of spatial incoherence of ground motion.

ISRS with amplified narrow frequency content were clipped for comparison to broad-banded test response spectra, typical of most NPP components.

The guidance in EPRI TR-103959 [33] Section 3 was performed for the peak clipping process. Additional response variability (uncertainty) for median-centered fragility analysis was added for use of a single TH set in response analysis per recent industry expert guidance [34].

No response analysis was performed for the Fuel Handling Building. The Fuel Handling Building is founded on piles that are socketed into hard rock. The SEL items in the Fuel Handling Building were limited to a small group of valves and rigidly mounted pressure transmitters. Based on walkdown and screening, the items were judged to be relatively rugged and detailed response analysis was not required for initial capacity screening. No subsequent response analysis was required based on low risk significance of these components.

4.3.2 Soil Structure Interaction (SSI) Analyses

The Service Water Pumphouse is founded on soil with depth to hard rock approximately 135 feet below grade. The structure is embedded in the soil and seismic response was found to be low compared to other buildings due to the combination of high soil damping and the attenuation of rock motion by the soil column [26].

Soil-structure interaction (SSI) analysis was performed for the Service Water Pumphouse using the EKSSI program. EKSSI provides a frequency domain solution to a dynamically loaded structure that rests on compliant soil. The EKSSI program performs the SSI analysis by combining the fixed-base building model and the foundation impedance matrix, then subjecting the model to the input acceleration time history motions. Soil degradation curves were used to develop strain-compatible soil properties including shear modulus and damping. Embedment effects were included in the analysis.

Uncertainty in the Service Water Pumphouse seismic analysis was accounted for by evaluating the SSI model for three soil columns: best estimate (BE), upper bound (UB), and lower bound (LB). The LB and UB soil property variation values were taken as 1/2 and 2 times the large strain BE shear modulus values, respectively, as suggested in ASCE 4-13 [29] Section 5.1.7 for the highest level of uncertainty of soil properties. An SSI analysis was performed for each of the BE, UB and LB cases. These three soil profiles account for the variation in the measured site-specific soil properties and account for most of the uncertainty in seismic response of the Service Water Pumphouse.

Free-field time histories were generated for the surface of the soil profile using SHAKE. Horizontal time histories were convolved to the surface using large-strain shear wave velocities and the vertical time history was convolved to the surface using compression wave velocity. A unique set of surface time histories was analyzed for each soil case.

In-structure response spectra (ISRS) were developed using the time-history analysis results. Both horizontal and vertical ISRS were computed from time-history motions at various floors or other important locations.

SSI analysis was also performed for the Condensate Storage Tank (CST) using EKSSI. The CST is founded on soil in the power block area with depth to hard rock approximately 68 feet below grade. Analysis was performed for BE, UB and LB soil properties in the manner described above for the Service Water Pumphouse, except the CST was treated as a surface founded structure. The results were used to perform a fragility analysis of the CST. The CST SSI analysis was originally performed for ESEP using the site-specific GMRS [12]. The GMRS shape is very similar to the 1E-5 UHRS shape and results were considered suitable for updated fragility analysis of the CST.

As described above, EKSSI was also used for time history analysis of the Auxiliary Building and Reactor Building to facilitate sensitivity studies. The analysis showed that the buildings are effectively fixed-base [26]. The effect of soil on the Diesel Generator Building was addressed by using an elastic foundation element for lateral displacement of piles within the fixed-base analysis [26]. BE, UB and LB responses for the Diesel Generator Building were developed by considering uncertainty in pile effective lateral stiffness.

4.3.3 Structure Response Models

Structure response models used in the response analyses include both 3-dimensional finite element models and lumped-mass stick models (LMSMs). All models were initially developed using the ANSYS analysis program. A fixed-base natural modes analysis was performed for each building. Where needed, ANSYS modal results were converted to EKSSI format for SSI analysis.

A detailed finite element model was developed for the Control Building. The Control Building is a symmetric box-like structure and global response could possibly be well-captured by a LMSM. However, initial review of the structure indicated vertical flexibility of the internal structure and horizontal flexibility of floor diaphragms would be difficult to realistically capture with a LMSM. The internal structure features concrete floor slabs with relatively long spans on steel columns. The building contains a significant amount of active SEL electrical equipment at multiple elevations including control cabinets containing relays. The finite element model was developed from plant drawings and related documents.

A detailed finite element model was also developed for the Intermediate Building. The building geometry is irregular with respect to lateral load paths and the global lateral load path would be difficult to realistically model with a LMSM. Also, review indicated the vertical flexibility of floor slabs and horizontal flexibility of floor diaphragms would be difficult to realistically capture with a LMSM. The building contains a significant amount of active SEL electrical and mechanical equipment at multiple elevations including batteries, switchgear, and transformers. The finite element model was developed from plant drawings and related documents.

LMSM were developed for the other structures as indicated in Table 4.3-1. Design-basis LMSM were obtained for these structures and enhanced. Enhancements included adding elements to represent the load path through concrete fill above the foundation rock, adding outrigger nodes to automatically capture torsional and rocking effects in time histories, and enhancing torsional response capabilities. Enhanced LMSMs met the modeling criteria of SPID [2] Section 6.3.1 and were judged suitable for SPRA response analysis. Criteria were verified by review of drawings, review of plant design basis calculations, and by supplemental calculations [26].

The guidance of ASCE 4-13 [29] Section 3.2 was applied for structural damping of buildings. The applied damping values are intended to have a conservative bias to meet the intent of EPRI NP-6041-SL [7] Table 2-5 for estimation of 84% NEP in-structure response. Buildings are primarily reinforced concrete shear-wall structures and applied damping was in the range of 4% to 7%. Based on a concrete cracking analysis, a 4% damping value was applied for the Reactor Building and 7% damping value was applied for other rock-founded structures. The Service Water Pumphouse is embedded in soil and very high damping is provided by the soil load path. A nominal 7% structure damping was applied for the Service Water Pumphouse and in-structure response was found to be insensitive to structure damping. Per a cracking analysis model element stiffnesses were adjusted per the guidance of ASCE 4-13 [29] Section 3.3.

For EKSSI analysis the effective foundation damping is per the frequency dependent impedance functions used in the analysis. The Service Water Pumphouse was found to exhibit high damping due to soil foundation and embedment. Foundation damping for the Auxiliary Building and Reactor Building was found to be insignificant overall, with limited benefit in the high-frequency region [26].

A simplified LMSM was developed for the Condensate Storage Tank to capture the seismic response using SSI analysis. Effective masses for the sloshing and impulsive modes were per EPRI NP-6041-SL [7] Appendix H. A 5% structural damping value was applied per EPRI NP-6041-SL [7].

Table 4.3-1 summarizes the type of analysis and model used for each of the structures modeled in the SPRA.

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

Structure	Foundation Condition	Type of Model	Analysis Method	Comments/Other Information
Auxiliary Building	Rock	LMSM	Deterministic SSI	BE, UB, LB cases analyzed, single TH set, shear wave velocity ≈ 10,000 feet/second. Per analysis, structure is effectively fixed base
Control Building	Rock	3D FEM	Fixed Base	BE, UB, LB cases analyzed, single TH set, shear wave velocity ≈ 10,000 feet/second
Diesel Generator Building	Rock	LMSM + 3D FEM	Fixed Base	BE, UB, LB cases analyzed, single TH set, 3D FEM for pile foundation
Intermediate Building	Rock	3D FEM	Fixed Base	Similar to Control Building
Reactor Building	Rock	LMSM	Deterministic SSI	Similar to Auxiliary Building
Service Water Pumphouse	Soil	LMSM	Deterministic SSI	BE, UB, LB cases, unique TH set for each soil case
Condensate Storage Tank	Soil	LMSM	Deterministic SSI	Similar to Service Water Pumphouse

4.3.4 Seismic Structure Response Analysis Technical Adequacy

The VCSNS SPRA Seismic Structure Response and Soil Structure Interaction Analysis were subjected to an independent peer review against the pertinent requirements in the ASME/ANS PRA Standard [4].

The peer review assessment, and subsequent disposition of peer review findings, is described in Appendix A, and establishes that the VCSNS SPRA Seismic Structure Response and Soil Structure Interaction Analysis are suitable for this SPRA application.

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, such as peak ground acceleration (PGA), peak spectral acceleration, floor spectral acceleration, etc. The SSC seismic fragility evaluations performed for VCSNS anchor the probability of each SSC failure to the horizontal PGA at the rock surface. The fragilities of the SSCs that participate in the SPRA accident sequences, i.e., those included on the seismic equipment walkdown list (SEL) are addressed in the model. Seismic fragilities for the significant risk contributors, i.e., those which have an important contribution to plant risk, are intended to be generally 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, identifies the tabulation of the fragilities (with appropriate parameters i.e., Am, βc, βr, βu), and the calculation method and failure modes for those SSCs determined to be sufficiently risk important, based on the final SPRA quantification (as summarized in Section 5). Important assumptions and important sources of uncertainty, and any particular fragility-related insights identified, are also discussed.

4.4.1 SSC Screening Approach

The seismic logic model (described in Section 5) was developed in parallel with the fragility analyses, starting with the initial judgments of seismic capacity. The SSC screening approach for fragility analysis was to perform fragility calculations in stages to make use of the plant response model feedback. With this approach the level of analysis effort applied for an SSC item was tied to the best understanding of its importance to plant seismic response. Risk-based SSC screening was not performed for SSC fragility screening. Fragility values were initially assigned to all SEL equipment and structures based on representative or bounding seismic capacity calculations using the conservative, deterministic failure margin (CDFM) criteria of EPRI NP-6041-SL [7] or the application of earthquake experience-based capacity described in EPRI 1019200 [32]. That method uses experience-based capacity and plant-specific demands. Refined analyses were then performed for important risk contributors per the direction of the Seismic Plant Response (SPR) team. The refined analyses included more detailed CDFM calculations and separation of variables (SOV) analyses per guidance of SPID [2] Section 6.4.1.

Initial CDFM calculations addressed a large group of SSC identified by the seismic review team during the walkdown and capacity screening phase of the project. This list was based on SRT judgment with respect to potential seismic vulnerability as well as potential equipment importance. The walkdown and capacity screening effort also verified minimum experience-based capacities that were applicable for initial CDFM analysis. The initial list of fragility targets was subsequently expanded to include CDFM-based fragility data for a large population of electrical equipment. Plant specific seismic capacities were also developed during the fragility refinements to supplement experience-based capacities. Plant-specific capacities were based on seismic qualification testing and analyses performed for design basis.

Initial calculations focused on passive structural items, equipment anchorage, potential seismic interactions and verification of screening caveats. Equipment functionalities were addressed using simplified CDFM analysis. Initial calculations showed that passive structural items and anchorage capacities were relatively robust. Subsequent refinements focused on equipment functionality, including relay chatter, and high-importance equipment such as NSSS equipment [35].

SSC items not assigned an item-specific fragility were grouped and assigned a lower bound fragility. Each grouping corresponded to specific plant area and considered both the seismic demand in the area and the type of equipment located therein. Each reported value is based on the lower bound capacity of all credited equipment in the corresponding area, excluding the equipment that was assigned a specific fragility. Per the direction of the SPR team, SSC items were broken out of the groupings during refinement iterations. The approach allowed for an orderly process to refine the plant response model. The approach is consistent with the SPID [2] Section 6.4.3 approach of retaining SSCs using a capacity-based screening level.

A separate systematic screening and analysis process was implemented to deal with the large number of relays and similar devices that were associated with functionality of SEL items. A comprehensive relay database was created that contains SEL relays, relay locations, and SEL components potentially affected by relay chatter. Initial CDFM-based capacities were calculated for relays using the methodology of EPRI NP-6041-SL [7] Appendix Q. Important relays were then identified by the plant response model team. Chatter analyses were performed for important relays as identified by the SPR team. Each chatter analysis was performed using a failure modes and effects analysis to examine the consequences of spurious changes of states on device contacts within a component's control circuit. The technical approach and criteria for this analysis were per EPRI NP-7148-SL [36].

4.4.2 SSC Fragility Analysis Methodology

Seismic fragility evaluations were performed for VCSNS SSCs contributing to CDF and LERF. The SSC fragility analysis was performed in accordance with Section 6.4 of the SPID [2] and the requirements defined in Section 5-2.2 of the ASME/ANS PRA Standard [4]. For fragility evaluation guidance, the SPID [2] recommends Seismic Fragility Applications Guide Update (EPRI 1019200 [32]), Seismic Fragility Application Guide (EPRI 1002988 [37]), Methodology for Developing Seismic Fragilities (EPRI TR-103959 [33]), and A Methodology for Assessment of Nuclear Plant Seismic Margin (EPRI NP-6041-SL [7]). The VCSNS fragility analysis is based on these documents, among other industry codes and standards.

VCSNS fragility parameters for SSCs were developed based on the following:

- Plant-specific design information.
- Use of conservative generic fragilities.
- The hybrid method outlined in the Seismic Fragility Application Guide (EPRI 1002988 [37]) and in Section 6.4.1 and Table 6-2 of the SPID [2].
- The more-detailed separation of variables approach outlined in Methodology for Developing Seismic Fragilities (EPRI TR-103959 [33]) per guidance of SPID [2] Section 6.4.1.

Critical failure modes were identified, and seismic fragility calculations were performed to develop three important fragility parameters: median capacity (Am), and logarithmic standard deviations for randomness and uncertainty (βr and βu). These three parameters provide sufficient information to construct a family of fragility curves for use in the SPRA logic model. In instances where a fragility estimate resulted in the SSC's contribution to SCDF and/or SLERF being significant, refinement was performed to better estimate the median capacity.

Structures

The following discussion applies to the following SEL buildings and structures

- Auxiliary Building
- Control Building
- Diesel Generator Building
- Fuel Handling Building
- Intermediate Building
- Reactor Building
- Service Water Pumphouse
- Service Water Pond Dam
- Condensate Storage Tank
- Refueling Water Storage Tank (RWST)
- Reactor Makeup Water Storage Tank (RMWST)

A detailed screening analysis was performed for SEL buildings to assign initial fragilities [38]. The screening criteria of the EPRI NP-6041-SL [7] Table 2-3 were evaluated, and the analysis confirmed a minimum experience-based 1.2g spectral acceleration capacity was applicable. A fragility was assigned to buildings based on the experience-based capacity. A relatively high seismic capacity is produced by experience-based capacity because the frequency range of interest (FROI) for damaging motion to plant buildings is the low frequency region of the UHRS. In this region the ground motion has relatively low spectral accelerations. As a result, further refinements to building fragility data were not required.

For the Service Water Pond Dam a detailed CDFM analysis was performed. Using the design basis analysis as input, a detailed scaling analysis was performed to evaluate the soil stress state for the UHRS motion. The scaling analysis accounted for the primary response modes of the dam.

A detailed CDFM analysis was performed for the Condensate Storage Tank using the method of EPRI NP-6041-SL [7] Appendix H. The seismic demand was obtained from the Condensate Storage Tank SSI analysis described in Section 4.3 above. Detailed CDFM analyses were also performed for the Refueling Water Storage Tank and Reactor Makeup Water Storage Tank using the method of EPRI NP-6041-SL [7] Appendix H. Those tanks are anchored to an Auxiliary Building floor slab and the appropriate ISRS were applied.

In general, seismic capacities for passive structures were high relative to components. To a significant extent, this results from the shape of the UHRS, which has low spectral accelerations in the frequency range of interest (FROI) for structures. Of the structures listed, the Reactor Makeup Water Storage Tank had the lowest seismic capacity. This tank is a back-up water source with low risk significance.

Based on walkdowns, there are no masonry wall hazards near safety-related SEL components [38]. From that result, it is concluded that the use of masonry wall structures was conservatively controlled during original plant design.

<u>Components</u>

The VCSNS component fragilities were derived using a multi-step approach. Simplified CDFM analysis was used to develop and assign fragilities to many components as an initial step, which included some conservative simplifying assumptions. The fragility parameters for certain risk-significant components (i.e., important contributors to SCDF and/or SLERF) were then refined to become more plant-specific and realistic. When experience-based capacities were utilized for mechanical and electrical components, the EPRI NP-6041-SL [7] Appendix F equipment caveats were confirmed to be satisfied. More recent experience-based capacities identified in EPRI 3002002933 [39] were applied where appropriate.

Realistic component failure modes included anchorage, lateral supports, functional failures, and failure due to seismic interactions. Anchorage capacities typically were calculated based on standard practice and functional capacities were extracted from existing quantification reports. The seismic demands for both anchorage and functional evaluations come from the ISRS. The ISRS is component specific and depends on the location of the component within the building and is generated from the seismic analysis building models.

Seismic fragility calculations for relays were performed and made use of GERS [40] and [41] and plant specific testing for seismic capacities. Relay GERS were verified to be applicable per relay vintage and model number. Full scale shake table testing with electrical monitoring was available for many control and electrical panels. The capacities used are lower than, or the same as, the capacities of these relays in the high frequency range [42] and [43]. Therefore, this evaluation addresses fragility for high frequency sensitive components as discussed in Section 6.4.2 of the SPID [2].

The nuclear steam supply system (NSSS) was evaluated for fragility variables. The primary system includes the reactor vessel, the steam generators, the reactor coolant pumps, a pressurizer, and the piping that connects these components to the reactor vessel. The fragility evaluation of these components was based on review and scaling of the existing safety analysis results, in accordance with SPID [2] Section 6.4 guidance.

Correlation

Correlation of components (or common cause failure) was considered in accordance with the ASME/ANS PRA Standard [4]. For the VCSNS SPRA, if the equipment was similar in design, with similar anchorage, and located in the same building on the same elevation, then the equipment was assumed to be fully-correlated.

In order to model the potential correlated failures of like components during an earthquake, the following general correlation rule was used:

- If the equipment is similar in design, with similar anchorage, and located in the same building on the same elevation, then it is treated as a correlated failure. That is, all of the similar equipment is modeled to fail with the same likelihood from a given challenge. For example, if one 480-v emergency switchgear fails given a particular seismic initiator, then the other also fails. In the PRA model, as discussed in Section 5, this one seismic failure would fail both trains of the switchgear.

- Otherwise, there is no correlation. For example, the 480-v switchgear failures are not correlated with the 480-v motor control center failures.

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, Table 5.4-3 (for SCDF) and Table 5.5-3 (for SLERF). By a process of successive refinement, detailed separation of variables (SOV) calculations have been performed for risk significant SSCs, as well as for selected other components.

Fragility refinements were executed in an iterative fashion. During early iterations the SOV calculations addressed NSSS equipment. Additional SOV refinements addressed functional failures and relay chatter.

4.4.4 SSC Fragility Analysis Technical Adequacy

The VCSNS SPRA SSC Fragility Analysis was subjected to an independent peer review against the pertinent requirements in the ASME/ANS PRA Standard [4].

The peer review assessment, and subsequent disposition of peer review findings, is described in Appendix A, and establishes that the VCSNS SPRA SSC Fragility Analysis is suitable for this SPRA application.

5.0 Plant Seismic Logic Model

This section summarizes adaptation of the VCSNS internal events at-power PRA model to create the seismic PRA plant response (logic) model.

The seismic plant logic model includes combinations of structural, equipment, and human failures that could give rise to significant core damage and large early release sequences. Quantification of this model yields total SCDF and SLERF, including contribution of both seismic-induced and random failures, and the identification of important seismic risk contributors. The quantification process also includes an evaluation of uncertainty, which provides perspective on how modeling and parametric sources of uncertainty affect the SPRA insights.

5.1 Development of the SPRA Plant Seismic Logic Model

The VCSNS seismic logic model was developed from the internal events at-power PRA model of record as of July 20, 2017. The internal events model was adapted in accordance with the EPRI Seismic PRA Implementation Guide [10] guidance and ASME/ANS PRA Standard [4] requirements. This process included adding seismic fragility events to the logic model, eliminating portions of the logic model irrelevant to seismic risk, and adjusting the human reliability analysis to account for response during and following an earthquake.

The final seismic logic model is a large fault tree with single top events for SCDF and SLERF. It uses the EPRI FRANX software, which is a scenario-based PRA calculation tool. It is used widely for hazards requiring definition of many scenarios (often hundreds or thousands), each representing one of 12 ground motion intervals with a certain plant impact for which the PRA logic model is quantified. The following subsections summarize each of the major changes made to the internal events logic model during SPRA model development.

<u>*General Approach*</u>

Seismic-induced initiating events that could give rise to significant accident sequences were first identified, including seismic-unique initiating events such as building failure. Next, SSCs whose seismic failure could cause an initiating event, or degrade plant response to an initiating event, were identified and consolidated into an SEL. Initially conservative fragility groups and lognormal fragility parameter estimates were identified for each SEL item.

The seismic hazard was discretized into twelve (12) uniform intervals, each with a representative ground motion level and occurrence frequency. Fragility events representing seismic failure of individual components, or groups of components, were developed for each ground motion interval. These fragility events were inserted into the fault tree using as targets the internal basic events associated with components in each fragility group. Internal human failure events relevant to seismic sequences were quantified using a screening process accounting for earthquake impact to performance shaping factors, and the resulting seismic human error probabilities were incorporated into the SPRA quantification process. A detailed seismic human reliability analysis was performed for one operator action related to containment isolation.

The resulting SPRA model is capable of quantifying at-power seismic-induced CDF and LERF, including the contributions of both seismic-induced and random hardware failures, as well as seismic impact to human actions.

Initiating Events and Accident Sequences

The initial step of the S-PRA was to systematically identify earthquake-caused initiating events that have the potential to give rise to significant accident sequences. In the initiating event identification process, a hierarchy was developed to ensure earthquakes exceeding an OBE are modeled, and that their frequency is apportioned to an appropriate induced initiating event. The VCSNS SPRA includes the following seismic-induced initiating events:

- Direct Core Damage and Large Early Release
- Direct Core Damage
- Interfacing Systems LOCA
- Large LOCA
- Medium LOCA
- Small LOCA
- Secondary Side Break Outside Containment
- Support System Initiating Events (120VAC, 125VDC, component cooling water, service water)
- Loss of Offsite Power
- Very Small LOCA following Seismic-Induced Non-LOCA Initiating Event

The direct core damage and release initiating events include seismic-unique failures such as building collapse. The potential for seismic-induced very small LOCA is modeled following non-LOCA initiating events. The initiating event and mitigating systems impact of seismic-induced fires and floods is also included quantitatively in the SPRA.

While the VCSNS SPRA uses a largely unmodified version of the internal events accident sequence and system modeling, some changes were required to reflect potentially risk significant seismic sequences that were not already included in the base internal events model. Table 5.1-1 identifies significant modifications to the internal events model that were required to support the SPRA.

Table 5.1-1 Summary of Modifications to the Internal Events CDF and LERF Models that were Required for the SPRA

Added logic to reflect seismic failures that lead directly to core damage and large early release
Added logic to reflect seismic failures that lead directly to core damage
Added logic to reflect seismic-induced very small loss of coolant accident following non-LOCA initiating events
Modified the Level 2 logic to allow seismic failures leading directly to core damage to contribute to the LERF quantification
Disabled credit to recovery of offsite power
Updated the mutually exclusive logic to prevent nonsensical failure combinations
Added flag events to enable portions of the fault tree developed by the fire PRA, but also relevant to seismic risk, to contribute to the SPRA quantification
Modified the pre-quantification flag file to prepare the fault tree for seismic risk quantification
Modified the recovery rule file to apply the seismic human reliability analysis
Used FRANX to create and insert logic reflecting seismic-induced initiating events and mitigating equipment failures

Modeling of Correlated Components

Treatment of correlation of modeled components is discussed in Section 4.4.2. Fully correlated components were assigned to correlated component groups so that all components in the group fail at the same time with the same probability based on the seismic magnitude for each hazard bin. The model assumes fully correlated response of same or very similar equipment in the same structure, elevation, and orientation. Correlated component groups were developed consistent with the above-mentioned correlation criteria and based on insights from component walkdowns.

Modeling of Human Actions

The VCSNS seismic HRA consists of the following tasks:

- Operator action identification and definition
- Feasibility assessment

- Screening quantification
- Detailed quantification
- Model integration

Operator actions to be modeled by the SPRA were identified and defined using the guidance of EPRI 3002008093 [8] Chapter 3. First, all operator actions modeled by the internal events PRA were identified. Pre-initiator HFEs are independent of the initiating event, and therefore there is no seismic impact to these actions. Then, HFEs not required to mitigate seismic-induced initiating events were screened. Next, a review was performed to identify seismically risk-relevant operator actions not already modeled by the internal events PRA (no new seismic-specific mitigating actions were identified in this analysis). Finally, all HFEs identified for inclusion in the SPRA model were defined.

The feasibility of each identified operator action was assessed using the guidance of EPRI 3002008093 [8] Section 4.2. The purpose of the feasibility assessment was to determine if successful completion of each operator action is even possible in the event of an earthquake. All actions determined to be infeasible must either not be modeled by the SPRA or have their associated HEPs set to 1.0. The feasibility assessment considered seismic impact on the following PSFs: time, manpower, cues, procedures and training, accessible location and environmental factors, and equipment accessibility, availability and operability.

All operator actions carried forward from the internal events PRA were determined feasible. The conclusion was that the internal events feasibility assessment of each action remains valid in the context of seismic events, and any exceptions are addressed by the screening quantification process.

Data required for the seismic HFE screening quantification were extracted from the internal events HRA Calculator file.

The following two human failure events have contributed significantly to the overall seismic PRA results throughout model development and its final quantification: OAEFC and OAESF3. OAEFC represents operator failure to continue EFW following battery depletion during a station blackout, and OAESF3 represents operator failure to start A and B train ESFAS equipment following failure of automatic ESFAS. Recognizing the conservative nature of the seismic HFE screening quantification, a more detailed assessment of these two actions was performed using the guidance of EPRI 3002008093 [8] Sections 5 and 6.

The assessment of HFE dependencies is discussed in Section 5.3.1.

Seismic LERF Model

The internal Level 2 fault tree logic did not require significant modification because seismic fragility events were generally mapped to individual SSCs, or groups of SSCs, either in the initiating event or mitigation logic. The associated seismic core damage sequences come up through the sequence level top gates and into the Level 2 logic via their corresponding plant damage states.

Seismic-induced failures leading directly to core damage were modeled above the sequence level logic. Each seismic-induced direct core damage sequence is grouped into a plant damage state, and the fault tree changes made enable the direct core damage events to generate seismic LERF sequences.

5.2 SPRA Plant Seismic Logic Model Technical Adequacy

The VCSNS SPRA seismic plant response methodology and analysis were subjected to an independent peer review against the pertinent requirements in the ASME/ANS PRA Standard [4].

The peer review assessment, and subsequent disposition of peer review findings, is described in Appendix A, and establishes that the VCSNS seismic plant response analysis is suitable for this SPRA application.

5.3 Seismic Risk Quantification

In the SPRA 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

The SPRA quantification generates a wealth of data, which generally require post-processing to extract meaningful insight. The SPRA estimates, at a minimum, the following metrics:

- Total seismic CDF and LERF
- Fractional CDF and LERF contributions of each ground motion interval
- CCDP and CLERP for each ground motion interval
- Occurrence frequency for each ground motion interval
- Ground motion at which the estimated likelihood of core damage and release are 1.0
- Fussell-Vesely importance for each fragility and human failure event

- Documented cutset review, including sampling of non-significant cutsets

To assess HRA dependency all unique combinations of HFEs that appear in the seismic CDF and LERF cutsets were identified. Using a Fussell-Vesely importance threshold, the dependency for all HFE combinations were screened based on low risk significance (i.e., HFE dependency was assessed but excluded from the final model due to low risk significance).

5.3.2 SPRA Model and Quantification Assumptions

Significant assumptions and sources of uncertainty for the VCSNS SPRA are summarized as follows:

1) Correlation: Seismic failures of multiple components sharing the same fragility group are assumed to be fully correlated. The VCSNS SPRA seeks to reduce conservatism associated with assuming full correlation by minimizing the number of large fragility groups occurring in significant accident sequences. Fragility groups were generally subdivided and refined until the point where their individual constituents were no longer significant. Part of this process is breaking correlation for components of differing design and location.

2) Building Failure: The fragility for SC-I buildings is mapped directly to core damage and large early release. This is conservative because the fragilities represent the onset of structural failure, as opposed to catastrophic collapse. The approach could potentially be refined by estimating fragilities for various levels of structural failures, although there is currently limited guidance and data that would support such an analysis. An exception is taken to this approach for the turbine building, which does not house a significant number of safety-related systems. While catastrophic turbine building collapse could cause rupture of multiple steam lines downstream of the main steam isolation valves (MSIVs), the %SLBO initiating event (representing single line rupture) is used instead.

3) Soil Failure: Although VCSNS is generally a hard rock site, the following localized soil failure concerns exist:

    - Diesel generator building foundation
    - Service water intake structure foundation
    - Service water pump house foundation

- Service water pond earthen dams

The service water intake, pump house, and earthen dam soil failures are modeled to cause core damage directly. The core damage mechanism is an unrecoverable loss of the ultimate heat sink.

The diesel generator building foundation fragility is mapped to failure of both emergency diesel generators.

This conservative approach is implemented given uncertainty in the extent and location of building displacement given soil failures. The fragility for soil failure more likely represents the onset of building foundational failure, as opposed to a catastrophic collapse failing all equipment in the building.

Further refinement may be possible; however, it is unclear that such effort would change the insights and subsequent decision-making based on the SPRA.

4) NSSS Component Support Failures: Seismic failure of structural supports for the steam generators and reactor coolant pumps is assumed to cause unrestrained motion and subsequent RCS rupture. This is conservative because the motion would likely be impeded by other structural elements, and pipe ductility may accommodate motion before rupturing.

5) Offsite Power: Loss of offsite power (LOOP) is assigned a generic fragility with a low capacity, and the SPRA does not credit recovery of offsite power. This treatment being conservative is underscored by operating experience where offsite power was not lost, and where offsite power was lost but recovered following an earthquake affecting North Anna.

6) Large LOCA: Seismic failure of the pressurizer supports is conservatively assumed to cause unrestrained motion of the pressurizer and subsequent failure of the surge line. This is conservative because the pressurizer motion would be impeded by the pressurizer cubical and associated structural supports, and pipe ductility may accommodate motion before generating a full line break. This modeling conservatism may artificially increase the importance of seismic-induced large LOCA, which is assumed for surge line rupture.

7) LOCA Location: The SPRA assumes that severe unrestrained movement of the pressurizer causes a large LOCA initiating event by rupture of the 14 inch surge line. Seismic large LOCA is therefore postulated to occur on RCS Loop A. Seismic-induced medium, small, and very small LOCAs are assumed to occur at any RCS loop location. The internal events LOCA accident sequence modeling is used by the SPRA, and the postulated seismic-induced LOCA location does not have any practical significance or modeling impact.

8) Seismic Initiating Event Tree: Development of the seismic initiating event tree involves a ranking of seismic initiating events from greatest to least in terms of potential risk significance, with the purpose of ensuring each interval is assigned to the most challenging initiating event that could credibly occur at that ground motion level. This ranking involves judgment and is a source of epistemic uncertainty.

   This source of uncertainty is mitigated by the VCSNS SPRA, which does not explicitly model the event tree success branches, and therefore the frequency assigned to each initiating event is not altered by the non-occurrence of preceding event tree events.

9) Rule-of-the-Box: In cases where components are housed within or otherwise integral to a larger component, a limiting fragility is sometimes developed to represent all of the elements associated with the larger component. For example, electrical breakers may be considered rule-of-the-box with the overall cabinet itself. This is a reasonable approach and not suspected to be a significant source of conservatism.

10) HRA: The seismic HRA uses a conservative screening process in accordance with EPRI 3002008093 [8]. In particular the assignment of ground motion levels to each HRA bin likely introduces some conservatism, consistent with a screening process.

11) SMALL vs. LARGE: Penetrations were screened by the internal events PRA on the basis of line size (<2 inch) and were reviewed for the SPRA. All but the following three 1 inch penetrations were able to be similarly screened by the SPRA: 407A, 407B, and 420. Failure of each of these penetrations would require the relatively fragile instrument air system (including offsite power) to remain available and failure of two trains of actuation. Failure of the inboard

and outboard valves on at least two of these penetrations, in combination with all 3/8 inch and 3/4 inch penetrations, or all four of the 1 inch penetrations, would be required to create a release path exceeding 2 inch equivalent diameter. In addition, the valves associated with these penetrations are spread across three different elevations, which significantly reduces the likelihood of their correlated failure. Based on these considerations, failures of these penetrations in the open position have been screened from the SPRA.

12) Accident Sequence Models, Success Criteria, and System Models: The SPRA uses the internal events accident sequence models, success criteria, and system models associated with seismic-induced initiating events. It is assumed that the earthquake does not change the accident progressions sufficient to warrant structural modification of the underlying event trees. The initiating event review performed for the SPRA included an examination of correlated failures and applicability of the internal events event trees.

13) Quantification: The seismic PRA quantification method itself is a recognized limitation. The VCSNS model discretizes the hazard into 12 intervals, calculates SSC failure probabilities at each interval, and CCDP/CLERP at each interval using the plant response model.

    In addition, the minimal cutset upper bound approximation used by CAFTA to calculate CCDP/CLERP from cutsets can be over-conservative where the conditional probabilities exceed 0.1, as is the case for many dominant seismic sequences. To minimize, but not eliminate, this conservatism, ACUBE is applied to the top 10,000 CDF and LERF cutsets.

5.4 SCDF Results

This section presents the base SCDF results, a list of SSCs that are significant contributors, including risk importance measures, and a discussion of significant cutsets. Section 5.7 provides a discussion of sensitivity studies.

The total point estimate seismic core damage frequency is 4.00E-05 per reactor year. Table 5.4-1 provides the earthquake occurrence frequency, conditional core damage probability, SCDF, and percent SCDF contribution for each ground motion interval. Note that the reported SCDF includes a 0.891 plant availability factor. The largest individual contribution to SCDF is from interval GC, which represents ground motion between 0.350 and 0.475 g.

Table 5.4-1 Contribution to SCDF by Acceleration Interval

Hazard Interval	Frequency (/yr)	CCDP	SCDF (/yr)	% of Total SCDF	Cumulative SCDF (/yr)
GA (0.100-0.225g)	3.93E-04	2.61E-05	9.13E-09	0.0%	9.13E-09
GB (0.225-0.350g)	7.47E-05	5.99E-02	3.99E-06	10.0%	4.00E-06
GC (0.350-0.475g)	2.42E-05	5.51E-01	1.19E-05	29.7%	1.59E-05
GD (0.475-0.600g)	1.13E-05	8.92E-01	8.97E-06	22.4%	2.49E-05
GE (0.600-0.725g)	5.82E-06	9.77E-01	5.07E-06	12.7%	2.99E-05
GF (0.725-0.850g)	3.50E-06	9.96E-01	3.11E-06	7.8%	3.30E-05
GG (0.850-0.975g)	2.14E-06	1.00E+00	1.91E-06	4.8%	3.50E-05
GH (0.975-1.100g)	1.44E-06	1.00E+00	1.28E-06	3.2%	3.62E-05
GI (1.100-1.225g)	1.01E-06	1.00E+00	9.04E-07	2.3%	3.71E-05
GJ (1.225-1.350g)	7.23E-07	1.00E+00	6.44E-07	1.6%	3.78E-05
GK (1.350-1.475g)	5.19E-07	1.00E+00	4.63E-07	1.2%	3.82E-05
GL (>1.475g)	2.02E-06	1.00E+00	1.80E-06	4.5%	4.00E-05

Table 5.4-2 identifies the top 15 SCDF cutset groups. Note that the VCS seismic cutset reviews are performed by first identifying the unique cutsets (independent of ground motion interval). Then a combined frequency for each unique cutset is totaled across all ground motion intervals, and the unique cutsets are ordered by decreasing combined frequency. This approach allows better prioritizing the review of significant cutsets. The CDF values reported are calculated using the minimal cutset upper bound approximation, which for this model is significantly conservative and do not reflect the improved accuracy of ACUBE; the Table 5.4-2 CDF values are therefore provided only for the purpose of relative ranking.

Table 5.4-2 Summary of Top 15 SCDF Cutsets

ID	CDF	Cutset	Description
1	3.18E-05	1HR_1 | RCP-4 | RELAY_0.11AC | RELAY_0.11BD | SBO-FLAG | SF-LSP | XHR_5 | OAEFC	Seismic-induced loss of offsite power by SF-LSP. RELAY_0.11AC fails EDG-A and RELAY_0.11BD fails EDG-B. Offsite power is not recovered (assumed for all PGA intervals). Operator fails to continue EFW following battery depletion (OAEFC).

Table 5.4-2 Summary of Top 15 SCDF Cutsets

ID	CDF	Cutset	Description
2	2.31E-05	NOSBO-FLAG | RELAY_0.11AC | RELAY_0.11BD | SF-LSP | SF-VSLOCA	Seismic-induced loss of offsite power with concurrent leakage in the VSLOCA range, along with failure of HPI due to loss of power, modeled as a small LOCA. RELAY_0.11AC fails EDG-A and RELAY_0.11BD fails EDG-B. Offsite power is not recovered (assumed for all PGA intervals). SLOCA mitigating equipment fails due to loss of power. Note that while station blackout occurs in this cutset, the "NOSBO-FLAG" indicates that the cutset is associated with another event tree, in this case VSLOCA.
3	2.09E-05	SF-FLD-FS374A | SF-LSP | SF-VSLOCA	Seismic-induced loss of offsite power with concurrent leakage in the VSLOCA range, along with failure of HPI due to flood impact to charging pumps A/B/C, modeled as a small LOCA. Note that while SLO-7 is the minimal sequence (small LOCA with failure of charging), the flood also fails several aspects of ECCS, including reactor makeup pumps (RWST refill), RWST level transmitters ILT00990/1/2/3/4, and RHR pumps A/B.
4	2.09E-05	SF-FLD-FS412A | SF-LSP | SF-VSLOCA	Similar to Cutset Group #3. This cutset involves a different FS flood source, which has similar targets to that in Cutset Group #3.
5	2.09E-05	SF-FLD-FS436A | SF-LSP | SF-VSLOCA	Similar to Cutset Group #3. This cutset involves a different FS flood source, which has similar targets to that in Cutset Group #3.
6	2.09E-05	SF-FLD-FS463A | SF-LSP | SF-VSLOCA	Similar to Cutset Group #3. This cutset involves a different FS flood source, which has similar targets to that in Cutset Group #3.
7	1.51E-05	1HR_1 | RCP-4 | RELAY_0.11AC | SBO-FLAG | SF-H2F-TB05 | SF-LSP | XHR_5 | SF-IGNITION-H2 | OAEFC	Seismic-induced loss of offsite power by SF-LSP. RELAY_0.11AC fails EDG-A. Seismic-induced fire fails EDG-B by support system impact. Offsite power is not recovered (assumed for all PGA intervals). Operator fails to continue EFW following battery depletion (OAEFC).
8	1.51E-05	1HR_1 | RCP-4 | RELAY_0.11AC | SBO-FLAG | SF-H2F-TB0102 | SF-LSP | XHR_5 | SF-IGNITION-H2 | OAEFC	Similar to Cutset Group #7, except that this cutset involves a different fire source, which has similar targets to that in Cutset Group #7.
9	1.25E-05	SF-H2F-AB0104 | SF-LSP | SF-VSLOCA | SF-IGNITION-H2	Seismic-induced loss of offsite power with concurrent leakage in the VSLOCA range, along with failure of HPI due to fire impact, modeled as a small LOCA. Note that while SLO-7 (small LOCA with HPI failure) is the minimal sequence, the fire also fails several aspects of ECCS, including pumps, valves, and RWST level transmitters.
10	1.14E-05	NOSBO-FLAG | RELAY_0.11AC | SF-H2F-TB05 | SF-LSP | SF-VSLOCA | SF-IGNITION-H2	Similar to Cutset Group #2, except that in this cutset, EDG-B is failed by fire (SF-H2F-TB05) impact to support systems including service water, room cooling, and fuel oil transfer.
11	1.14E-05	NOSBO-FLAG | RELAY_0.11AC | SF-H2F-TB0102 | SF-LSP | SF-VSLOCA | SF-IGNITION-H2	Similar to Cutset Group #2, except that in this cutset, EDG-B is failed by fire (SF-H2F-TB05) impact to support systems including service water, room cooling, and fuel oil transfer.

Table 5.4-2 Summary of Top 15 SCDF Cutsets

ID	CDF	Cutset	Description
12	1.01E-05	NOSBO-FLAG | RELAY_0.11BD | RELAY_0.28A | SF-LSP | SF-VSLOCA	Seismic-induced loss of offsite power with concurrent leakage in the VSLOCA range, along with failure of HPI, modeled as a small LOCA. SLO-7 is the minimal sequence (small LOCA with failure of HPI). HPI fails due to loss of power to Train 'B' (RELAY_11BD impact to EDG-B with LOOP) and spurious isolation of VCT causing loss of charging pump suction via RELAY_0.28A chatter.
13	1.01E-05	NOSBO-FLAG | RELAY_0.11AC | RELAY_0.28B | SF-LSP | SF-VSLOCA	Similar to Cutset Group #12 except that opposite trains are impacted (RELAY_0.11AC fails EDG-A and RELAY_0.28B fails LCV-115E).
14	9.26E-06	1HR_1 | RCP-4 | RELAY_0.11AC | RELAY_0.32B | SBO-FLAG | SF-LSP | XHR_5 | OAEFC | FL-SWB-FAIL	Seismic-induced loss of offsite power by SF-LSP. RELAY_0.11AC fails EDG-A. RELAY_0.32B fails EDG-B by loss of service water support. Offsite power is not recovered (assumed for all PGA intervals). Operator fails to continue EFW following battery depletion (OAEFC).
15	9.26E-06	1HR_1 | RCP-4 | RELAY_0.11BD | RELAY_0.32A | SBO-FLAG | SF-LSP | XHR_5 | OAEFC	Seismic-induced loss of offsite power by SF-LSP. RELAY_0.32A fails EDG-A by loss of service water support. RELAY_0.11BD fails EDG-A. Offsite power is not recovered (assumed for all PGA intervals). Operator fails to continue EFW following battery depletion (OAEFC).

Table 5.4-3 summarizes the fragilities with a CDF Fussell-Vesely (approximate) importance of greater than 2%. Refer to the SPRA quantification notebook for importance measures for the balance of fragilities. The Fussell-Vesely for each fragility was approximated by multiplying the median capacity by a factor of five, then recalculating the failure probability at each ground motion interval, imposing those failure probabilities onto the cutsets, recalculating the CDF by ACUBE, and finally calculating the percent reduction of total CDF that the improved capacity affords.

Offsite power (SF-LSP) has the greatest importance, which is due to its low assumed capacity, its assumed non-recoverability, and the significant plant impact that losing offsite power creates. Relay_0.11AC and Relay_0.11BD are important because they each fail a diesel generator, which is required to mitigate loss of offsite power. Very small LOCA following a non-LOCA initiating event is also an important contributor due to its low capacity, and the conservative modeling that core damage occurs if high pressure injection fails. The balance of fragilities have relatively low importance.

Note that random (non-seismic) failures generally do not contribute significantly to the seismic CDF, with the following exceptions which have estimated Fussell-Vesely values greater than 2%: the emergency diesel generators (XEG1A and

XEG1B) and the turbine-driven auxiliary feedwater pump (XPP8). The random failure contribution of theses SSCs is related to the importance of station blackout to the seismic risk profile.

Table 5.4-3 SCDF Importance Measures Ranked by Fussell-Vesely

Fragility	Description	F-V	A_m (g)	β_r	β_u	Failure Mode	Method
SF-LSP	Offsite Power	53.9%	0.3	0.3	0.45	Yard-Centered Loss of Offsite Power	Generic (Conservative Estimate)
Relay_0.11AC	Relay Fragility Group	18.7%	0.31	0.24	0.38	Chatter	CDFM
SF-VSLOCA	Very Small LOCA Fragility	14.5%	0.474	0.24	0.32	Structural failure of small impulse lines leading to RCS pressure boundary breach	Design Basis Scaling (SSE/UHRS)
Relay_0.11BD	Relay Fragility Group	10.5%	0.31	0.24	0.38	Chatter	CDFM
SF-XTFF-YD03	Seismic-Induced Fire on Bin 27/28/29 Yard Transformers	2.0%	0.43	0.24	0.49	Casing rupture leading to oil spill, followed by conditional ignition probability	Generic (Conservative Estimate)

Table 5.4-4 identifies the relative contribution of each initiator to total CDF for those initiators comprising the top 95% of CDF, or individually contributing greater than 1% to total CDF.

Table 5.4-4 Relative Contribution of Each Initiator to Total SCDF

Initiator	Description	Proportion
SLO	Small LOCA	42.27%
SBO	Loss of Offsite Power and Progression to SBO	22.05%
CSLO	Consequential Small LOCA (RCP)	8.18%
LCCW	Loss of CCW	6.38%
SSB	Secondary Side Break	4.88%
TRANS	General Transient	4.86%
MLO	Medium LOCA	3.67%
LLO	Large LOCA	2.91%
SI-CD	Direct Core Damage	1.95%
LSW	Loss of Service Water	1.61%
ATWS	Anticipated Transient Without SCRAM	1.24%

5.5 SLERF Results

This section presents the base SLERF results, a list of SSCs that are significant contributors, including risk importance measures, and a discussion of significant cutsets. Section 5.7 provides a discussion of sensitivity studies.

The total point estimate seismic large early release frequency is 3.65E-06 per reactor year. Table 5.5-1 provides the earthquake occurrence frequency, conditional core damage probability, SLERF, and percent SLERF contribution for each ground motion interval. Note that the reported SLERF includes a 0.891 plant availability factor. The largest individual contribution to SLERF is from interval GL, which represents ground motion exceeding 1.475 g.

Table 5.5-1 Contribution to SLERF by Acceleration Interval

Hazard Interval	Frequency (/yr)	CLERP	SLERF (/yr)	% of Total SLERF	Cumulative SLERF (/yr)
GA (0.100-0.225g)	3.93E-04	0.00E+00	0.00E+00	0.0%	0.00E+00
GB (0.225-0.350g)	7.47E-05	1.70E-04	1.13E-08	0.3%	1.13E-08
GC (0.350-0.475g)	2.42E-05	1.99E-03	4.30E-08	1.2%	5.43E-08
GD (0.475-0.600g)	1.13E-05	7.67E-03	7.71E-08	2.1%	1.31E-07
GE (0.600-0.725g)	5.82E-06	2.20E-02	1.14E-07	3.1%	2.46E-07
GF (0.725-0.850g)	3.50E-06	5.78E-02	1.80E-07	4.9%	4.26E-07
GG (0.850-0.975g)	2.14E-06	1.23E-01	2.35E-07	6.4%	6.60E-07
GH (0.975-1.100g)	1.44E-06	2.21E-01	2.83E-07	7.7%	9.44E-07
GI (1.100-1.225g)	1.01E-06	3.48E-01	3.15E-07	8.6%	1.26E-06
GJ (1.225-1.350g)	7.23E-07	4.88E-01	3.14E-07	8.6%	1.57E-06
GK (1.350-1.475g)	5.19E-07	6.17E-01	2.86E-07	7.8%	1.86E-06
GL (>1.475g)	2.02E-06	1.00E+00	1.80E-06	49.1%	3.65E-06

Table 5.5-2 identifies the top 15 SLERF cutset groups. Note that the VCS seismic cutset reviews are performed by first identifying the unique cutsets (independent of ground motion interval). Then a combined frequency for each unique cutset is totaled across all ground motion intervals, and the unique cutsets are ordered by decreasing combined frequency. This approach allows better prioritizing the review of significant cutsets. The LERF values reported are calculated using the minimal cutset upper bound approximation, which for this model is significantly conservative and do not reflect the improved accuracy of ACUBE; the Table 5.5-2 LERF values are therefore provided only for the purpose of relative ranking.

Table 5.5-2 Summary of Top SLERF Cutsets

ID	LERF	Cutset	Description
1	9.23E-07	1HR_1 | ISOLERF | RCP-4 | RELAY_0.11AC | RELAY_0.11BD | SBO-FLAG | SF-AB_SURG | SF-LSP | SF-XVT2662B | XHR_5 | OAEFC	Seismic-induced loss of offsite power by SF-LSP. RELAY_0.11AC fails EDG-A and RELAY_0.11BD fails EDG-B. Offsite power is not recovered (assumed for all PGA intervals). Operator fails to continue EFW following battery depletion (OAEFC). Containment isolation fails by failure to isolate instrument air containment penetration. XVT-2662A fails by SF-AB_SURG and XVT-2662B fails by SF-XVT2662B.
2	9.2E-07	ISOLERF | SF-AB_SURG | SF-LSP | SF-VSLOCA | SF-XVT2662B	Seismic-induced loss of offsite power with concurrent leakage in the VSLOCA range, along with failure of HPI due to SF-AB_SURG, modeled as a small LOCA. While SLO-7 (small LOCA with failure of HPI) is the minimal sequence, additional small LOCA mitigating failures occur by SF-AB_SURG impact to ECCS equipment. Containment isolation fails by failure to isolate instrument air containment penetration. XVT-2662A fails by SF-AB_SURG and XVT-2662B fails by SF-XVT2662B.
3	8.99E-07	SF-SCREENBLDG	Seismic-induced collapse of SC-I buildings (auxiliary, control, intermediate, reactor, and service water pump house buildings) leading directly to core damage and large early release.
4	7.87E-07	ISOLERF | RELAY_0.32A | SF-AB_SURG | SF-XVT2662B | RCPTHE	Loss of CCW initiating event. RELAY_0.32A fails CCW train A. SF-AB_SURG impact fails miniflow valve XVB-9503B, which fails CCW train B. Normal seal injection fails by SF-AB_SURG impact to flowpath valves. Alternate seal injection assumed to fail because it is not SC-I. Operator fails to trip the RCPs (RCPTHE). At this point there is loss of CCW, no seal injection, and RCP failure to trip leading to seal LOCA with no CCW, corresponding to the LCCW-2 core damage sequence.
5	7.85E-07	1HR_1 | ISOLERF | RCP-4 | RELAY_0.11BD | RELAY_0.32A | SBO-FLAG | SF-AB_SURG | SF-LSP | SF-XVT2662B | XHR_5 | OAEFC	Seismic-induced loss of offsite power by SF-LSP. RELAY_0.32A fails EDG-A and RELAY_0.11BD fails EDG-B. Offsite power is not recovered (assumed for all PGA intervals). Operator fails to continue EFW following battery depletion (OAEFC). Containment isolation fails by failure to isolate instrument air containment penetration. XVT-2662A fails by SF-AB_SURG and XVT-2662B fails by SF-XVT2662B.
6	7.85E-07	1HR_1 | ISOLERF | RCP-4 | RELAY_0.11AC | RELAY_0.32B | SBO-FLAG | SF-AB_SURG | SF-LSP | SF-XVT2662B | XHR_5 | OAEFC | FL-SWB-FAIL	Similar to Cutset Group #5, except that RELAY_0.11AC fails EDG-A and RELAY_0.32B fails EDG-B.

Table 5.5-2 Summary of Top SLERF Cutsets

ID	LERF	Cutset	Description
7	7.84E-07	ISOLERF | RELAY_0.32A | SF-AB_SURG | SF-VSLOCA | SF-XVT2662B	Seismic-induced loss of CCW concurrent leakage in the VSLOCA range, along with failure of HPI due to SF-AB_SURG, modeled as a small LOCA. RELAY_0.32A fails CCW train A and SF-AB_SURG fails CCW train B. Small LOCA mitigating failures by SF-AB_SURG impact to ECCS equipment. Containment isolation fails by failure to isolate instrument air containment penetration. XVT-2662A fails by SF-AB_SURG and XVT-2662B fails by SF-XVT2662B.
8	7.13E-07	ISOLERF | RELAY_0.37C | SF-AB_SURG | SF-XVT2662B | RCPTHE | FL-SWA-FAIL	Similar to Cutset Group #4 except that RELAY_0.37C fails CCW train A via SW train A support.
9	7.12E-07	1HR_1 | ISOLERF | RCP-4 | RELAY_0.11BD | RELAY_0.37C | SBO-FLAG | SF-AB_SURG | SF-LSP | SF-XVT2662B | XHR_5 | OAEFC | FL-SWA-FAIL	Similar to Cutset Group #5 except that RELAY_0.37C fails EDG-A via SW train A support.
10	7.11E-07	ISOLERF | RELAY_0.37C | SF-AB_SURG | SF-VSLOCA | SF-XVT2662B | FL-SWA-FAIL	Similar to Cutset Group #7 except that RELAY_0.37C fails CCW train A by SW train A support.
11	6.61E-07	1HR_1 | ISOLERF | RCP-4 | RELAY_0.40 | SBO-FLAG | SF-AB_SURG | SF-LSP | SF-XVT2662B | XHR_5 | OAEFC	Similar to Cutset Group #5 except that RELAY_0.40 fails both EDG-A and EDG-B by impact to room ventilation.
12	6.6E-07	ISOLERF | SF-AB_SURG | SF-IB_SURG | SF-XVT2662B	Loss of CCW initiating event. SF-AB_SURG fails CCW train A. SF-IB_SURG fails CCW train 'B'. EFW fails (LCC-1) by SF-IB_SURG broad impacts to EFW valves and pumps. Containment isolation fails by failure to isolate instrument air containment penetration. XVT-2662A fails by SF-AB_SURG and XVT-2662B fails by SF-XVT2662B.

Table 5.5-2 Summary of Top SLERF Cutsets

ID	LERF	Cutset	Description
13	6.55E-07	ISOLERF | RELAY_0.52 | SF-XVT2662B | CBPMI--XPP1BHE | RCPTHE | FL-MSO-14 | FL-SWA-FAIL	Loss of CCW initiating event. RELAY_0.52 fails CCW train A (by SW train A support). Operators fail to start CCW train B by CBPMI—XPP1BHE. Normal seal injection fails by RELAY_0.52 impact to valves leading to RWST draindown (MSO-14). Alternate seal injection assumed to fail because it is not SC-I. Operator fails to trip the RCPs (RCPTHE). At this point there is loss of CCW, no seal injection, and RCP failure to trip leading to seal LOCA with no CCW, corresponding to core damage sequence LCCW-2. Containment isolation fails by failure to isolate instrument air containment penetration. XVT-2662A fails by RELAY_0.52 and XVT-2662B fails by SF-XVT2662B.
14	6.53E-07	1HR_1 | ISOLERF | RCP-4 | RELAY_0.11BD | RELAY_0.52 | SBO-FLAG | SF-LSP | SF-XVT2662B | XHR_5 | OAEFC	Seismic-induced loss of offsite power by SF-LSP. RELAY_0.52 fails EDG-A by SW support and RELAY_0.11BD fails EDG-B. Offsite power is not recovered (assumed for all PGA intervals). Operator fails to continue EFW following battery depletion (OAEFC). Containment isolation fails by failure to isolate instrument air containment penetration. XVT-2662A fails by RELAY_0.52 and XVT-2662B fails by SF-XVT2662B.
15	6.52E-07	ISOLERF | RELAY_0.52 | SF-VSLOCA | SF-XVT2662B | CBPMI--XPP1BHE | FL-MSO-14 | FL-SWA-FAIL	Loss of CCW initiating event with concurrent leakage in the VSLOCA range. RELAY_0.52 fails CCW train A by SW train A support. Operators fail to start CCW train B by CBPMI—XPP1BHE. RELAY_0.52 impacts fail HPI. RELAY_0.52 broad impacts fail ECCS and SLOCA mitigation (for example by MSO-14 RWST draindown). Containment isolation fails by failure to isolate instrument air containment penetration. XVT-2662A fails by RELAY_0.52 and XVT-2662B fails by SF-XVT2662B.

Table 5.5-3 summarizes the fragilities with a LERF Fussell-Vesely (approximate) importance of greater than 2%. Refer to the SPRA quantification notebook for importance measures for the balance of fragilities. The Fussell-Vesely for each fragility was approximated by multiplying the median capacity by a factor of five, then recalculating the failure probability at each ground motion interval, imposing those failure probabilities onto the cutsets, recalculating the LERF by ACUBE, and finally calculating the percent reduction of total CDF that the improved capacity affords.

The LERF fragility importance profile is much flatter than for CDF. Fragilities most important to LERF are those representing SSCs whose failure affects containment isolation. This can include fragilities for specific containment isolation valves (for example SF-XVT2662B and SF-XVG06067), failure of automatic containment isolation caused by loss of 120 VAC (for example by SF-APN590X, SF-XIT590X, and SF-XBA1), and relay chatter impacts (for example Relay_0.52).

Note that random (non-seismic) failures generally do not contribute significantly to the seismic LERF, with the following exceptions which have estimated Fussell-Vesely values greater than 2%: the emergency diesel generators (XEG1A and XEG1B) and the reactor building alternate purge system being in operation. The random failure contribution of the diesel generators is related to the importance of station blackout to the seismic risk profile, and the random failure contribution of reactor building purge being in operation is related to containment isolation failure.

Table 5.5-3 SLERF Importance Measures Ranked by Fussell-Vesely

Fragility	Description	F-V	A_m (g)	β_r	β_u	Failure Mode	Method
SF-XVT2662B	RB IA SUCTION HDR ISOLATION VALVE	13.70%	1.8	0.14	0.36	Functional	SOV
SF-XBA1	DC DISTRIBUTION BUS BATTERY	8.20%	2.38	0.24	0.38	Functional	CDFM
Relay_0.52	Relay Fragility Group	5.70%	1.48	0.24	0.38	Chatter	CDFM
SF-APN590X	120 VOLT VITAL AC DISTR PANEL	5.70%	1.52	0.19	0.52	Functional	SOV
SF-AB_SURG	AB IN-STRUCTURE SURROGATE	4.90%	1.21	0.24	0.25	Functional	Surrogate
SF-XVG06067	ALTERNATE PURGE EXHAUST ISOLATION VALVE	4.40%	1.5	0.24	0.38	Functional	CDFM, Experience-based capacity
SF-XIT590X	120 VOLT VITAL AC 10 KVA UPS	4.10%	1.57	0.19	0.48	Functional	SOV
SF-DPN1HX	BATTERY MAIN DISTRIBUTION PANEL 1HA and 1HB	3.80%	2.89	0.2	0.42	Functional	SOV
SF-XVD06242B	AOVs Located at Low Elevation (412) on Fuel Building	3.00%	2.28	0.24	0.38	Functional	CDFM, Experience-based capacity
Relay_0.14D	Relay Fragility Group	2.70%	0.43	0.24	0.38	Chatter	Screening Capacity
SF-IB_SURG	IB IN-STRUCTURE SURROGATE	2.50%	1.21	0.24	0.25	Functional	Surrogate
SF-DPN1HA1_B1	DC DISTRIBUTION PANELS 1HA1 AND 1HB1	2.20%	3.6	0.19	0.49	Functional	SOV
SF-LSP	OFFSITE POWER	2.20%	0.3	0.3	0.45	Yard-Centered Loss of Offsite Power	Generic (Conservative Estimate)

5.6 SPRA Quantification Uncertainty Analysis

Parametric uncertainty in the SPRA results originates from seismic hazard curve uncertainty, the SSC fragility uncertainties, and basic event failure parameter uncertainties from the internal events PRA. Parametric uncertainty quantification was performed using the UNCERT code.

Table 5.6-1 documents the SCDF and SLERF uncertainty quantification results, using Latin Hypercube with 10,000 samples and ACUBE applied to the top 1,000 cutsets. Note that the plant availability factor is applied to the UNCERT results post-quantification.

Table 5.6-1 Uncertainty Quantification Results

Metric	Point Estimate	Mean	5^{th}	Median	95^{th}	Standard Deviation	Skewness
SCDF	4.00E-05	4.79E-05	6.61E-06	3.04E-05	1.43E-04	5.91E-05	5.612
SLERF	3.65E-06	5.21E-06	5.21E-07	2.95E-06	1.68E-05	7.92E-06	10.535

The error factor (defined as [95th / Median], or [Median / 5th]) for SCDF and SLERF is approximately five (5) and six (6) respectively, which is generally high compared to internal events error factors. This is caused by relatively high fragility uncertainty, indicated by parameters β_u and β_r, as well as hazard uncertainty. The relatively high error factor (compared to internal evens) resulting from the seismic uncertainty quantification is expected and consistent with industry operating experience.

Figure 5.6-1 and Figure 5.6-2 depict the probability density function and cumulative distribution function for SCDF and SLERF.

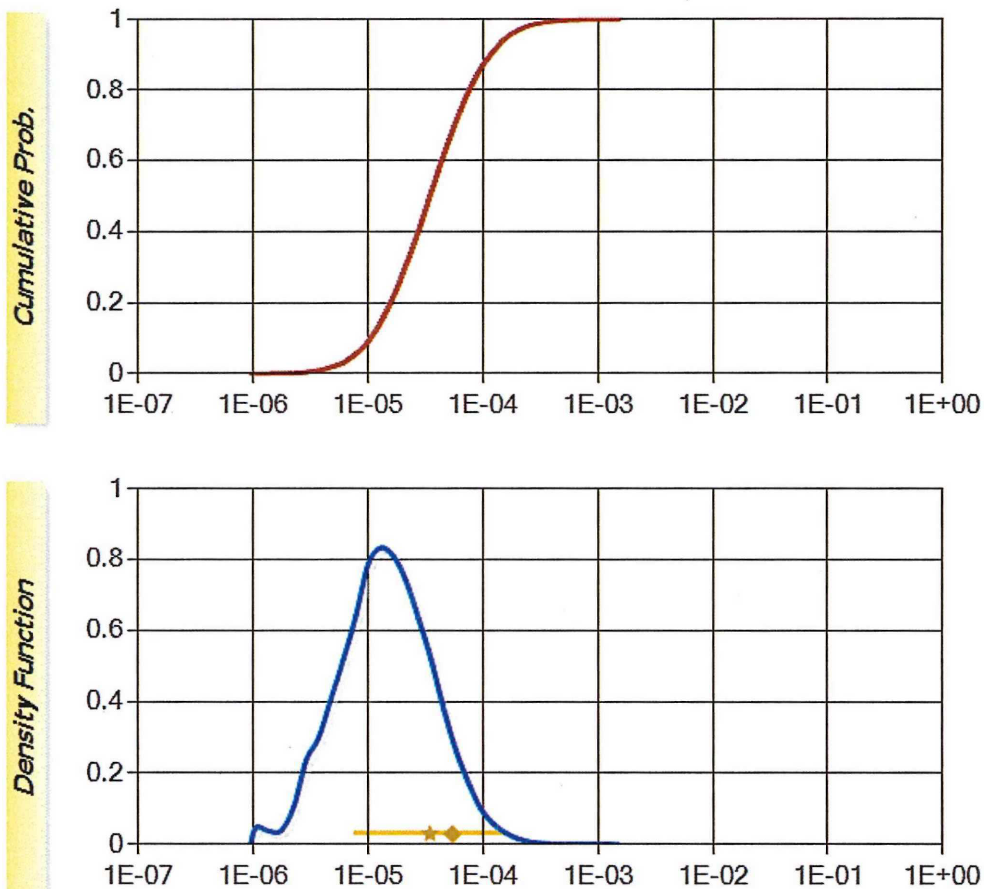

Figure 5.6-1 Seismic CDF Uncertainty (10,000 sample LH and ACUBE on 1,000 cutsets)

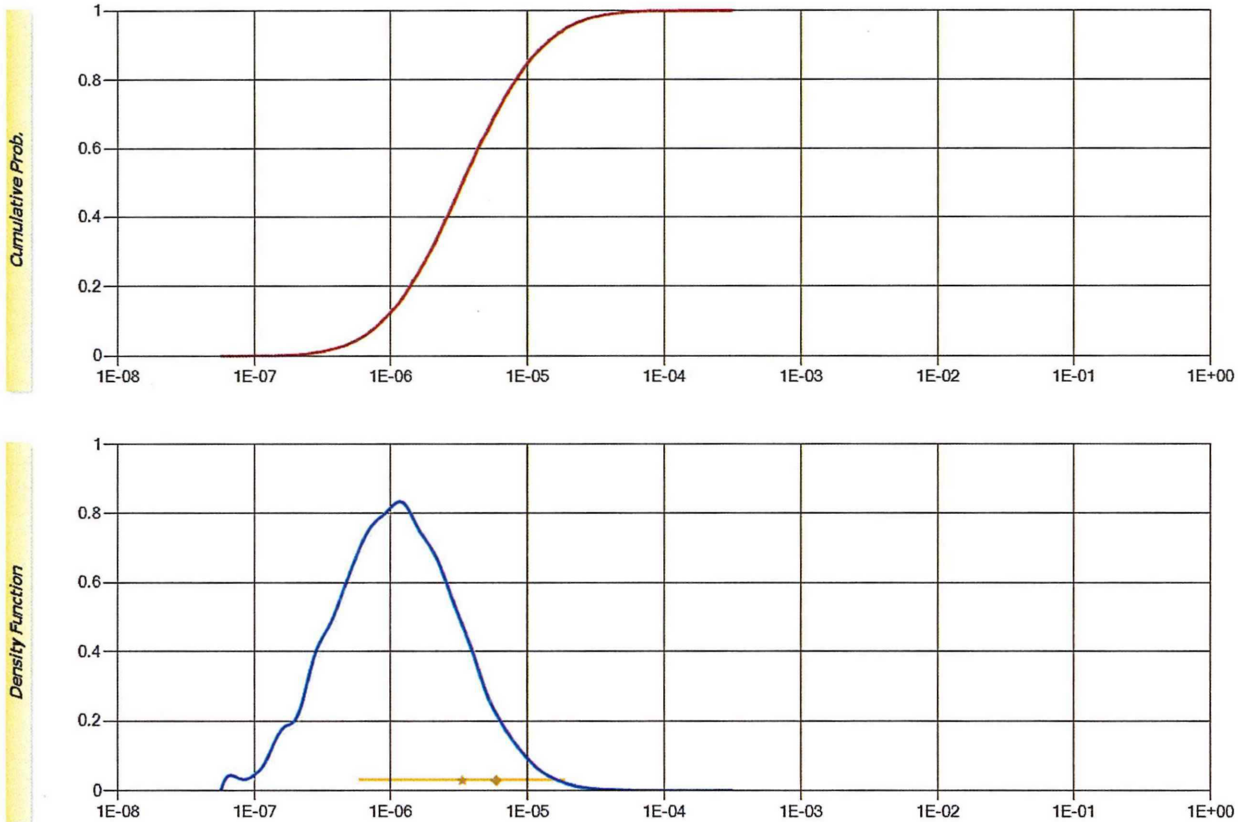

Figure 5.6-2 Seismic LERF Uncertainty (10,000 sample LH and ACUBE on 1,000 cutsets)

Model uncertainty is introduced when assumptions are made by the SPRA model and inputs to represent plant response, when there may be alternative approaches to particular aspects of the modeling, or when there is no consensus approach for a particular issue. For the VCSNS SPRA, the important model uncertainties are addressed through the sensitivity studies to determine the potential impact on SCDF or SLERF.

Completeness uncertainty relates to potential risk contributors that are not in the model. The scope of the VCSNS SPRA is for at-power operation and does not include risk contributors from low power-shutdown operation, or for spent fuel pool risk. In addition, there may be potential issues related to factors that are not included, such as the impact of aging on equipment reliability and fragility. Other potential issues include impacts of plant organizational performance on risk, and unknown omitted phenomena and failure mechanisms. By their nature, the impacts on risk of these types of uncertainties are not known.

5.7 SPRA Quantification Sensitivity Analysis

The VCSNS SPRA includes quantitative sensitivity studies of the following elements to assess model stability and sources of epistemic uncertainty:

- Truncation Limits for Model Convergence
- ACUBE Post-Processing
- Hazard Interval Definition
- Fragility Sensitivity Study of SF-APN590X Correlation
- Fragility Sensitivity Study of SF-XIT590X Correlation
- Model Sensitivity Study of Seismic HRA (HEPs Set to One)
- Model Sensitivity Study of Seismic HRA (HEPs Set to Nominal)
- Model Sensitivity Study of Containment Failure by Multiple Small Penetrations

5.7.1 Truncation Limits for Model Convergence

The VCSNS SPRA truncation limits for CDF and LERF are established in accordance with the ASME/ANS PRA Standard [4]. Specifically, convergence is considered sufficient when successive reductions in the truncation limit of one decade result in decreasing changes in CDF (LERF) and the final change is less than five (5) percent.

A truncation study was performed on the peer reviewed VCSNS SPRA to demonstrate convergence of total seismic CDF and LERF across all ground motion intervals. ACUBE was applied to the top 10,000 cutsets of each ground motion interval, or all of the cutsets for ground motion intervals with fewer than 10,000 cutsets. Convergence for CDF was demonstrated with a truncation value of 1.00E-08/yr for all ground motion intervals. Convergence was demonstrated for LERF with a truncation value of 1.00E-09/yr for all ground motion intervals except the final (GL) interval, which had to be kept at 1.00E-07/yr to quantify within memory limitations. The same truncation limits were used in the Revision 1 model, which included resolution to several F&Os, with the exception that the final two LERF intervals (GK and GL), which used 5.00E-09/yr and 5.00E-08/yr to allow the quantification to complete without exceeding memory limitations.

5.7.2 ACUBE Post-Processing

Due to the rapidly increasing memory requirements of ACUBE with larger numbers of cutsets, it is not always possible to evaluate an entire cutset file with ACUBE. An evaluation was performed using the peer reviewed model, where the model was iteratively re-quantified using increasing numbers of cutsets to which ACUBE was applied, and examining the percent change in results and computational burden. Based on this evaluation, the application of ACUBE to 10,000 cutsets was selected for the final SPRA quantification.

5.7.3 Hazard Interval Definition

The VCSNS SPRA discretizes the seismic hazard, starting at the 0.1 g operating basis earthquake, into 12 intervals of uniform width (0.125 g), with the exception of the final interval that accumulates all peak ground acceleration above 1.475 g. Using the peer reviewed model, convolution of the mean hazard with the point estimate plant level fragilities for core damage and large early release confirmed that the CDF and LERF estimates are converged at the selected 12-interval hazard discretization.

5.7.4 Fragility Sensitivity Study on SF-APN590X Correlation

The fragility group SF-APN590X includes three vital 120 volt alternating current (VAC) distribution panels and was consistently a dominant LERF contributor throughout model development, due to the importance of 120 VAC to automatic containment isolation. A sensitivity study was performed using the peer reviewed SPRA, which treated seismic failure of each individual panel as independent, rather than fully correlated. The sensitivity case resulted in a 10.2% reduction in LERF, and a 1.0% reduction in CDF. While the base treatment (assuming full correlation) was retained in the model due to similarity of design and location between each panel, the study characterized the relative importance of panels within the SF-APN590X group.

Note that resolution of F&Os in the Revision 1 SPRA included improved HRA on operator failure to manually isolate containment, which reduced the importance of the 120 VAC distribution panels.

5.7.5 Fragility Sensitivity Study on SF-XIT590X Correlation

Similar to the vital 120 VAC distribution panels, the fragility group for the vital 120 VAC inverters (SF-XIT590X) was consistently a dominant LERF contributor throughout model development. A sensitivity study was performed using the peer reviewed SPRA, which treated seismic failure of each individual inverter as independent, rather than fully correlated. The sensitivity case resulted in a 7.2% reduction in LERF, and a 0.6% reduction in CDF. While the base treatment (assuming full correlation) was retained in the model due to similarity of design and location between each inverter, the study characterized the relative importance of inverters within the SF-XIT590X group.

Note that resolution of F&Os in the Revision 1 SPRA included improved HRA on operator failure to manually isolate containment, which reduced the importance of the 120 VAC inverters.

5.7.6 Model Sensitivity Study on Seismic HRA (HEPs Set to One)

A sensitivity study was performed using the peer reviewed SPRA, which set all seismic HEPs to 1.0 (guaranteed failure). This increased seismic CDF by 34.2% and LERF by 1.8%. The low LERF impact suggests that most HEPs important to LERF are already 1.0, or near 1.0, in the base model. The modest CDF increase suggests that some of the more important operator actions are not set to one in the base model.

5.7.7 Model Sensitivity Study on Seismic HRA (HEPs Set to Nominal)

A sensitivity study was performed using the peer reviewed SPRA, which set all seismic HEPs to their internal events values. The intent of this study was to characterize the lower bound seismic CDF and LERF that could be achieved by detailed HRA on all seismic HFEs.

The sensitivity case reduced seismic CDF by 52.3% and LERF by 36.8%. This illustrates that although performing additional detailed seismic HRA could benefit both CDF and LERF, achieving significant reductions is unlikely, especially given the importance of higher ground motion intervals where human action reliability is significantly reduced.

5.7.8 Model Sensitivity Study on Containment Isolation Failure

The SPRA includes a qualitative screening of correlated failures of multiple small containment penetrations from the LERF model. To support this screening, a sensitivity study was performed using the peer reviewed SPRA, which assumes guaranteed failure of containment isolation if both fragilities for offsite power and instrument air are successful. The sensitivity case increased LERF 11.0% above the base case. Given that this is a bounding estimate, which does not credit fragilities associated with the small containment penetrations and assumes their correlated failure, the qualitative screening of the subject small containment penetrations from the base model is considered acceptable.

5.7.9 Summary of Sensitivity Study Results

Table 5.7-1 summarizes the quantitative sensitivity study results discussed in preceding sections.

Table 5.7-1 Summary of Quantitative Sensitivity Study Results

Study	ΔCDF	ΔLERF
Breaking SF-APN590X Correlation	-1.0%	-10.2%
Breaking SF-XIT590X Correlation	-0.6%	-7.2%
All Seismic HEPs to 1.0	+34.2%	+1.8%
All Seismic HEPs to Nominal	-52.3%	-36.8%
Assuming Failure of Small Containment Penetrations	-	+11.0%

5.8 SPRA Quantification Technical Adequacy

The VCSNS SPRA quantification methodology and analysis were subjected to an independent peer review against the pertinent requirements in the ASME/ANS PRA Standard [4].

The peer review assessment, and subsequent disposition of peer review findings, is described in Appendix A, and establishes that the VCSNS SPRA quantification is suitable for this SPRA application.

6.0 Conclusions

A seismic PRA has been performed for Virgil C. Summer Nuclear Station in accordance with the guidance in the SPID [2]. The SPRA shows that the point estimate seismic CDF is 4.00E-05/yr and the seismic LERF is 3.65E-06/yr. Uncertainty, importance, and sensitivity analyses were performed. 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.

The SPRA as described in this submittal reflects the as-built/as-operated Virgil C. Summer Nuclear Station as of the SPRA freeze date, July 20, 2017. An assessment is included in Appendix A of the impact on the results of plant changes not included in the model. 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

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8.0 Acronyms

ACUBE	Advanced Cutset Upper Bound Estimator
AFE	Annual Frequencies of Exceedance
ANS	American Nuclear Society
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
BE	Best Estimate
CCDP	Conditional Core Damage Probability
CCW	Component Cooling Water
CDF	Core Damage Frequency
CDFM	Conservative Deterministic Failure Margin
CEUS	Central-Eastern United States
CEUS-SSC	Central-Eastern United States Seismic Source Characterization
CST	Condensate Storage Tank
ECCS	Emergency Core Cooling System
EPRI	Electric Power Research Institute
ESEP	Expedited Seismic Evaluation Program
ETSZ	Eastern Tennessee Seismic Zone
F&O	Fact and Observation
F-V	Fussel-Vesely
FEM	Finite Element Model
FIRS	Foundation Input Response Spectra
FRANX	Fire Risk Analysis and XINIT
FROI	Frequency Range of Interest
GERS	Generic Equipment Ruggedness Spectra
GMM	Ground-Motion Model
GMRS	Ground Motion Response Spectra
IPEEE	Individual Plant Examination for External Events
ISRS	In-Structure Response Spectra
HCLPF	High Confidence of a Low Probability of Failure
HEP	Human Error Probability
HFE	Human Failure Event
HRA	Human Reliability Analysis
HLR	High Level Requirement
ISRS	In-Structure Response Spectra
LB	Lower Bound
LERF	Large Early Release Frequency
LH	Latin Hypercube
LMSM	Lumped-Mass Stick Model
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power

MAFEs	Mean Annual Frequencies of Exceedances
MC	Monte-Carlo
MSIV	Main Steam Isolation Valve
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NTTF	Near Term Task Force
OBE	Operating Basis Earthquake
PGA	Peak Ground Acceleration
PRA	Probabilistic Risk Assessment
PSHA	Probabilistic Seismic Hazard Analysis
PWR	Pressurized Water Reactor
RAW	Risk Achievement Worth
RCS	Reactor Coolant System
RLME	Repeated Large-Magnitude Earthquake
RMWST	Reactor Makeup Water Storage Tank
RRW	Risk Reduction Worth
RWST	Refueling Water Storage Tank
SA	Spectral Acceleration
SBO	Station Blackout
SCDF	Seismic Core Damage Frequency
SEL	Seismic Equipment List
SFR	Seismic Fragility Element Within ASME/ANS PRA Standard
SHA	Seismic Hazard Analysis Element Within ASME/ANS PRA Standard
SLERF	Seismic Large Early Release Frequency
SOV	Separation of Variables
SPID	Screening, Prioritization and Implementation Details
SPR	Seismic PRA Modeling Element Within ASME/ANS PRA Standard
SPR	Seismic Plant Response
SPRA	Seismic Probabilistic Risk Assessment
SR	Supporting Requirement
SRT	Seismic Review Team
SSCs	Structures, Systems and Components
SSE	Safe Shutdown Earthquake
SSI	Soil Structure Interaction
TH	Time History
UB	Upper Bound
UHRS	Uniform Hazard Response Spectra
V/H	Vertical-to-Horizontal
VCSNS	Virgil C. Summer Nuclear Station

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

This Appendix provides a summary of the peer review of the VCSNS SPRA, the peer review F&O closure reviews, and provides the bases for why the SPRA is technically adequate for the response to the NRC's request under 50.54(f) [1].

The VCSNS SPRA was subjected to an independent peer review against the pertinent requirements in Part 5 of Addendum B of the ASME/ANS PRA Standard [4] as detailed in Section A.1.

The information presented in this Appendix 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 Revision 2 [45] and the requirements in Section 1-6 of the ASME/ANS PRA Standard [4], and presents the significant results of the peer review.

A.1. Overview of Peer Review

The peer review assessments, and subsequent disposition of peer review findings, are summarized in this Appendix. The scope of the reviews encompassed the set of technical elements and supporting requirements (SRs) for the SHA (seismic hazard), SFR (seismic fragilities), and SPR (seismic PRA modeling) elements for seismic CDF and LERF. Two peer reviews address these elements. The SHA elements for VCSNS (Unit 1) were peer reviewed as part of the VCSNS Units 2 and 3 (AP1000®[1]) SPRA peer review performed in 2017 [46]. The AP1000 SPRA used a seismic hazard evaluation for Unit 1 that was judged to be applicable to Units 2 and 3, so the AP1000 peer review [46] explicitly addressed the Unit 1 hazard. The SFR and SPR elements were subject to a peer review in 2018 [6]. Combining the VCSNS SPRA peer review [6] and the AP1000 SPRA peer review [46] the full set of SRs identified in Tables 6-4 through 6-6 of the SPID [2] have been addressed.

The VCSNS SPRA peer review for the SFR and SPR elements during the week of April 9, 2018. A walkdown of the VCSNS plant was performed by a subset of the team on April 9, 2018.

The AP1000 SPRA peer review that covered the SHA elements was performed over five days during the week of July 10-14, 2017.

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A.2. Summary of the Peer Review Process

The April 2018 peer review [6] was performed against the requirements in Part 5 (Seismic) of Addenda B of the ASME/ANS PRA Standard [4], using the peer review process defined in NEI 12-13 [5]. The review was conducted over a four-day period, with a summary and exit meeting on the morning of the fifth day.

The July 2017 peer review [46] followed the NEI-12-13 [5] process for performing peer reviews of external hazard PRAs, and the ASME/ANS PRA Standard Addendum A [56], as well as ASME/ANS PRA Standard Addendum B [4], with NRC clarifications provided in Regulatory Guide 1.200 [45]. The review was conducted over a four-day period, with a summary and exit meeting on the morning of the fifth day.

The NEI 12-13 SPRA peer review process [5] 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 of 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 supporting requirements (SRs) provide a structure which, in combination with the peer reviewers' PRA experience, provides the basis for examining the various PRA technical elements. If a reviewer identifies a question or discrepancy, that leads to additional investigation until the issue is resolved or a Fact and Observation (F&O) is written describing the issue and its potential impacts, and suggesting possible resolution.

For each technical element, i.e., SHA, SFR, SPR, a team of two peer reviewers were assigned, one having lead responsibility for that area. For each SR reviewed, the responsible reviewers reached consensus regarding which of the Capability Categories defined in the Standard that 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 are prepared. There are three (3) types of F&Os defined in NEI 12-13 [5]: Findings - which identify issues that must be addressed in order 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.3. Peer Review Team Qualifications

The April 2018 VCSNS SPRA review was led by Mr. Kenneth Kiper of Westinghouse Electric Company. Team members included: Dr. Ram Srinivasan, an independent consultant; Mr. Eddie Guerra of RIZZO Associates; Dr. Se Kwon Jung of Duke Energy; Mr. Vince Andersen of JENSEN-HUGHES; and Mr. Bob Kirchner of JENSEN-HUGHES, representing Exelon Generation Company. The lead and reviewer qualifications have been reviewed by SCE&G and have been confirmed to be consistent with requirements in the ASME/ANS PRA Standard [4] and the guidelines of NEI-12-13 [5]. Consistent with the requirement in Section 1-6.2.2 of the ASME/ANS PRA Standard [4], the members of the peer review team were independent of the VCSNS SPRA. They were not involved in performing or directly supervising work on any PRA Element evaluated in the overall VCSNS SPRA.

Mr. Kenneth Kiper, the team lead, has over 35 years of experience at Westinghouse and, previously at Seabrook Station, in the nuclear safety area generally and PRA specifically for both existing and new nuclear power plants. He has lead a number of peer reviews, including reviews of internal events PRAs, internal flood PRAs, fire PRAs, high wind PRAs, and several seismic PRAs.

Dr. Ram Srinivasan was the lead for the review of the Seismic Fragility Analysis (SFR) technical element. Dr. Srinivasan has over 45 years of experience in the nuclear industry, principally in the design, analysis (static and dynamic, including seismic), and construction of nuclear power plant structures. He is actively involved in the Post-Fukushima Seismic Assessments (NRC NTTF 2.1 and 2.3) and is a member of the NEI Seismic Task Force and the ASME/ANS JCNRM, Part 5 Working Group (Seismic and other External Hazards PRA). He has participated on several previous SPRA peer reviews, either as reviewer or utility consultant.

Dr. Srinivasan was assisted in fragility review by Mr. Eddie Guerra and Dr. Se Kwon Jung. Mr. Guerra is the director of structural engineering at RIZZO Associates, and has seven years specializing in earthquake engineering, seismic PRA, structural dynamics, and steel and concrete structural design. Mr. Guerra has participated in a number of SPRA peer reviews as a utility consultant.

Dr. Se Kwon Jung is a lead PRA engineer at Duke Energy, responsible for external hazards PRA development for the Duke fleet including seismic PRA projects for Robinson and Oconee Nuclear Stations, both fragility development and systems analysis. He has over 15 years of experience as a civil/structural engineer in the nuclear power industry and has participated as a reviewer in several SPRA peer reviews.

Mr. Vince Andersen was the lead for the review of the Seismic System Response Analysis (SPR) technical element. Mr. Andersen is a consultant with JENSEN-HUGHES Inc. with over 30 years' experience in systems and reliability engineering and risk assessment. He is a member of the ASME/ANS PRA Standards committee and a member of the Part 5 Working Group, specifically for technical element SPR. He has participated in several previous SPRA peer reviews, either as reviewer or utility consultant.

Mr. Andersen was assisted in the SPR review by Mr. Robert Kirchner and Mr. Kenneth Kiper (the team lead). Mr. Kirchner is a consultant with JENSEN-HUGHES Inc. with 28 years' experience in reliability and risk assessment. He has been involved in all aspects of PRA development and application. He has participated in a number of previous SPRA peer reviews, as reviewer or utility consultant.

The July 2017 AP1000 SPRA review was led by Mr. Barry Sloane of JENSEN HUGHES. Team members that reviewed the SHA elements were: Dr. Annie Kammerer, an independent consultant; and Dr. Robert Youngs, from AMEC Foster Wheeler Environment and Infrastructure. The lead and reviewer qualifications have been confirmed to be consistent with requirements in the ASME/ANS PRA Standard [4] and the guidelines of NEI-12-13 [5]. None of the reviewers were involved in the development of the AP1000 V.C. Summer 2&3 SPRA and as such, the review team meets the necessary level of independence per NEI-12-13 [5] from the V.C. Summer 2&3 SPRA development.

Mr. Barry Sloane has 35 years of experience serving the commercial nuclear power industry, thirty-one years of which have been focused in risk assessment, risk management, reliability, and related areas. Mr. Sloane is a JENSEN HUGHES manager responsible for leading various PRA modeling and risk application development and implementation programs and has served as technical lead for the development of seismic PRAs. Mr. Sloane has extensive knowledge of PWR reactor systems, procedures, and operations. Mr. Sloane has led and participated in numerous PRA peer reviews and supports the NEI PRA Peer Review Task Force. Mr. Sloane is currently a member of the ASME/ANS Joint Committee on Nuclear Risk Management (JCNRM), serving as Chair of the JCNRM Subcommittee on Standards Development and Chair of the JCNRM Working Group on Interpretations.

Dr. Annie Kammerer is a seismic hazard and risk specialist with over 15 years of consulting and nuclear regulatory experience. Dr. Kammerer was the lead reviewer for 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 power plants and LNG facilities. Her experience also includes projects in the industrial, transportation, and critical buildings sectors. She is currently an independent consultant, as well as a visiting scholar at the Pacific Earthquake Engineering Research Center at the University of California, Berkeley. She has participated in numerous peer reviews and is a member of Working Group 5 of the ASME/ANS PRA Standards committees.

Dr. Robert Youngs was a supporting reviewer for the SHA technical element. Dr. Youngs has more than 40 years of consulting experience, with primary emphasis on hazard and decision analysis. He has pioneered approaches for incorporating earth sciences data, along with their associated uncertainties, into probabilistic hazard analyses. The focus of this work has been to develop quantitative evaluations of hazard by combining statistical data and expert judgment. He is the Manager of the Decision Analysis (DA) practice area at AMEC Foster Wheeler Environment and Infrastructure. Dr. Youngs was awarded the 2012 Jesuit Seismological Award by the Eastern Section of the Seismological Society of America for contributions to observational seismology. Dr. Youngs has considerable experience in assessing earthquake hazards in central and eastern North America (CENA) and has completed seismic hazard analyses for numerous nuclear power plant sites.

A.4. Summary of the Peer Review Conclusions

The review team's assessment of the SPRA elements is excerpted from the peer review reports [6] and [46] as follows. Where the review team identified issues, these are captured in peer review findings, for which the dispositions are summarized in the next section of this Appendix.

A.4.1. Seismic Hazard Analysis

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 NUREG-2115 (i.e., the “CEUSSSC” model) with the adjustments to the earthquake recurrence rates provided in EPRI (2014). 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. Because the V.C. Summer site is a hard rock site, the seismic hazard analysis does not need to incorporate the effects of local site response.

The Senior Seismic Hazard Analysis Committee (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 and NUREG-2117) 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 as a result of SSHAC Level 3 analyses. 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 method of conduct of the PSHA generally, as well as addressing several individual requirements related to data collection, data evaluation and model development, and quantification of uncertainties supporting HLRSHA-A to HLR-SHA-D.

As a first step to performing a PSHA, the Standard requires that 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 paleoseismic 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 V.C. Summer site, those portions of all distributed seismic sources in the CEUS-SSC model within 640 km of the site were included in the PSHA calculations. In addition, all Repeated Large Magnitude Earthquake (RLME) sources within 1000 km of the site 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 V.C. Summer 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 V.C. Summer site PSHA. Even though the PSHA conducted was performed specifically for the V.C. Summer 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 or not the existing model is appropriate for its continued use in the intended application. In the case of the V.C. Summer site, no evaluation of new seismic activity was made to assess the impact of recent earthquakes and no literature review was performed to determine if new sources had been identified, leading to a finding.

The PSHA was computed using the EPRI (2013) GMC model, which is the most recent well-developed characterization of earthquake ground motions for the CEUS. The EPRI (2013) model addresses both aleatory and epistemic uncertainties in ground motions. The EPRI (2013) GMC model does not account for differences in earthquake mechanism.

Additionally, the Standard addresses all sources that can potentially cause important vibratory ground motion at the V.C. Summer site. The CEUS-SSC model used to assess vibratory ground motions explicitly removes non-tectonic earthquakes, which is appropriate because the underlying causation is different from tectonic earthquakes and is non-stationary (i.e., it may change over relatively short time periods). However, human-induced seismicity (e.g., earthquakes from wastewater injection) can produce damaging ground motions in some cases. While induced seismicity should not be incorporated into the tectonic SSC model, a separate catalog of induced events can be compiled and evaluated via a screening process. A search for of potential sources of human activity that could produce future induced earthquakes was only performed within the 5-mile site area, leading to a finding. An evaluation of the nearby Monticello Reservoir (which has had two episodes of reservoir-induced seismicity) was not performed, leading to a second finding. Paths to resolution of both of the human-induced seismicity findings were identified during the review.

The PSHA results are provided over an appropriately wide range of spectral frequencies and annual frequencies of exceedances. Uncertainties in the control point 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 and mean hazard curves and uniform hazard response spectra, and 100 weighted hazard curves for peak ground acceleration.

The Standard requires that spectral shapes be based on a site-specific evaluation taking into account the contributions of deaggregated magnitude-distance results of the probabilistic seismic hazard analysis. 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.

Vertical to horizontal (V/H) ratios are used to calculate vertical response spectra. Site-specific V/H ratios were developed for the V.C. Summer site using accepted practice for nuclear applications.

Both the aleatory and epistemic uncertainties have been addressed in characterizing the seismic sources and ground motion models. The control point hazard calculations are based on the CEUS-SSC and EPRI GMC models. During the development of these models, some uncertainties in the seismic sources and ground motion prediction equations were included and appropriate sensitivity analyses were performed to demonstrate the sensitivity of the results to uncertainties in key model parameters. However, the sensitivity results were provided without any context or efforts towards identifying the most important uncertainties in the PSHA.

As discussed above, a search for human activities that may lead to induced earthquakes was carried out only with the 5-mile site area. However, induced earthquakes at distances greater than 5 km could have an effect on the site ground-shaking hazard, which led to a finding. A path forward to resolving this issue was identified during the onsite peer review.

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 PSHA is documented in a manner that generally facilitates PRA applications, upgrades, and peer review. However, a comprehensive summary report for seismic hazard analysis that includes and summarizes the development of probabilistic ground motions, evaluations of new information, site response, and evaluations related to human-induced and secondary hazard in one place would benefit future reviews. Documentation related F&Os are provided to improve the completeness and clarity of the documentation.

A.4.2. Seismic Fragility Analysis

The SFR assessment covered the three principal elements of the fragility analysis; namely, site-specific seismic response analysis, plant walkdown, and fragility analysis calculations. The reviews of the three elements are briefly summarized below.

Site-specific seismic response analysis of the various buildings housing the Seismic Equipment List items used the ground motions corresponding to the Uniform Hazard Response Spectra (UHRS) shape provided in the plant Probabilistic Seismic Hazard Analysis (PSHA) reports. The Reference Level Earthquake (RLE) corresponded to the UHRS shape with a mean annual exceedance frequency of 1E-5, which is anchored to 0.70g peak ground acceleration (PGA). This ground motion level was based on the interim results of the PRA quantifications and corresponds to the expected failure levels of most of the top contributors to seismic risk. The seismic input to the building response analysis is based on one set (3 components) of time history matched to the UHRS shape. The development of building models relies heavily in the modification of existing lumped-mass stick models. These modifications are clearly documented in the respective building model development report. For the Control and Intermediate Buildings, a new finite element building model was developed. Fixed-base dynamic analyses were performed for most of the buildings as they are founded on competent rock or concrete backfill. Soil-Structure Interaction analyses were performed for the Diesel Generator building and Service Water Pumphouse structures that are founded on soil. Dynamic analyses used median centered (best estimate) properties and considered variability in soil and structural properties. The results of the dynamic analyses were used in the subsequent fragility analysis of SSCs.

The seismic response analysis of buildings was supplemented by a number of sensitivity studies used to assess important modelling assumptions. These sensitivity studies included difference in response between single and multiple time histories; effect of spatial incoherency of ground motion in building response; and control building in-structure sensitivity study. These studies served to reinforce engineering judgments and assumptions implemented in the fragility analysis of SSCs.

A limited walkdown of the V.C. Summer Nuclear Plant was performed as part of this Peer Review. The walkdown review focused on the dominant risk contributors in seismic CDF and LERF, but also included some additional example SSCs to confirm findings or observations made by the Seismic Review Team. The walkdown included the following buildings:

- Control Building
- Auxiliary Building
- Intermediate Building
- Turbine Building
- Circulating Water Pump Structure
- Service Water Pump House

For the DRPI Cooling Unit Outlet Header Isolation Valve, the walkdown team reviewed past walkdown photos to verify sufficient shake space. From the peer review walkdown, the peer review team was able to confirm seismic capacity walkdown findings and observations made by the Seismic Review Team for selected items. For example, electrical cabinets and mechanical equipment were well supported without seismic interaction concerns. Similarly, distribution systems such as cable trays, raceways and fluid system piping runs were well supported vertically and laterally. It was noteworthy that cable trays in SC-I structures are supported by relatively stiff welded steel angle frames. Battery cells inside Battery Room were well supported by the rugged braced battery rack frame. The peer review team noted that metal shims were inserted between the battery rack and battery cells to eliminate the potential for pounding of the battery cell with the battery rack due to earthquake excitation. Hydrogen piping and much of Fire Service Water system piping were welded piping and well supported. No seismic interaction concerns were noted along the operator pathway from Auxiliary Building to TD EFW Pump Room.

Fragility parameters were calculated for the all the SEL items credited in the plant Seismic PRA Model (roughly 220). A capacity-based screening was performed using the guidelines of EPRI NP-6041. However, no items were screened out of the model. For those items exceeding the screening level, surrogate elements were introduced with fragility values corresponding to the screening level. The initial fragility calculations were categorized into two groups: one group consisted of about 20 types of components for which detailed analyses were performed, and the second group, consisting about 50 types of components, simplified analyses were performed based on scaling design basis or other available plant data. The initial analyses were all based on the EPRI Conservative Deterministic Failure Margin (CDFM) approach. Based on the results of initial seismic risk quantification, refined methods including the EPRI Separation of Variables (SoV) were used for a select group of components that were significant contributors to risk.

The fragility calculation considered credible failure modes, primarily functional and structural (anchorage) failure modes. Given the vintage of the plant (early 1980's), a broad band seismic ground motion (NUREG/CR-0098 shape) was used in the design of the plant. It was noticed that the structures and component anchorage have significant margin over the design basis, and thus tended not to be controlling. In addition, there were no masonry block walls in the main power block buildings. Thus, there were no significant Seismic Category II/I interactions which needed to be addressed. The fragility evaluation did address seismic induced fire and flood scenarios.

In summary, the fragility analysis generally meets the applicable requirements of element SFR in Part 5 the ASME/ANS PRA Standard.

A.4.3. Seismic Plant Response Analysis

The V.C. Summer SPRA systems model was developed starting from the internal events PRA and captures seismically-induced failures along with random failures, unavailability and operator errors. The SPRA was determined to adequately model seismically induced initiating events: the process was systematic to identify, screen, and model the events. The seismic initiating events are appropriately tied to the corresponding accident sequence models. Significant accident sequences involving safety related systems are included in the model. F&Os are provided regarding some seismic-induced initiating event states, as well as a suggestion to use an actual SIET model as opposed to a conceptual one.

The seismic PRA systems model has been reasonably adapted to incorporate aspects of seismic analysis that are different from corresponding aspects found in the at-power, internal-events PRA systems model. However, numerous open finding-level F&Os from the internal events PRA peer review remain unresolved. Section 4.1.4 addresses the impact on the SPRA model review of these open findings.

The approach to seismic correlation of SSCs is described. Many of the SEL SSCs represented in the SPRA model have been treated as correlated response groups at this stage of model development. The fragility correlation modeling approach used is binary; partial correlation is not used. Nuances for fragility correlation grouping of relay chatter fragilities are discussed in the analysis. These are typical SPRA modeling approaches.

The SPRA in effect does not screen out SSCs from the SPRA quantification process. The fragility complement (i.e., "success") probabilities are addressed through the use of the EPRI ACUBE post-processor software; this is a typical SPRA approach when using the EPRI CAFTA suite of codes.

The SPRA models the potential of a “very small LOCA” (VSLOCA) using the small LOCA fragility as well as treating the scenario as if it is a small LOCA. As a result, the VSLOCA model is effectively the SLOCA model and the VSLOCA scenarios are missing. The review team did not identify a reasonable basis for assuming the capacity leading to a VSLOCA could be modeled as a SLOCA capacity. This aspect of the SPRA was determined to be inconsistent with the requirement of SPR-B8.

The SPRA documents the assessment and modeling of seismically-induced internal fire and internal flooding scenarios. The level of analysis was assessed as meeting the applicable requirements but suggestion F&Os are provided regarding the modeling of some of these scenarios in the SPRA.

The SPRA is assessed as generally reflecting the as-built as-operated plant given the SPRA is built upon the Internal Events PRA and considering the extensive walkdowns. A number of F&Os are provided on conservatisms of note (e.g., FLEX not modeled, relay chatter recovery not modeled for some key chatter scenarios).

The list of SSCs included in the seismic equipment list (SEL) was compiled using a systematic process and is contained in both a database and documented in the Model Notebook includes all SSCs that participate in the accident sequences, i.e., the SEL reflects the PRA model. A single element, buried pipe, was identified as missing from the SEL.

The process used to quantify seismic CDF and seismic LERF appropriately integrates the seismic hazard, the seismic plant response model, and the seismic and non-seismic failures. The process is reasonably well documented and traceable, and the high level requirements for quantification in Part 2 of the Standard were judged to be satisfied. The modeling does not selectively decide to keep or remove fragilities from the quantification process; all those included in the SPRA logic model propagate during the quantification process. Parametric uncertainty analyses and sensitivity studies are performed and risk importance results are provided.

The SPRA documentation was assessed as meeting the SPR documentation requirements. The SPRA reports are designed in a modular fashion that includes an SPRA Modeling report that describes approaches and then SPRA Quantification report documents the compilation and quantification of the SPRA. These main documents are supported by numerous supported files and documents.

A.5. Revision of SPRA Model and Documentation

Following the peer review, the SPRA model and documentation were updated to address 32 F&Os. In addition, SCE&G generated closure documentation for each of these F&Os. Subsequently, the updated SPRA model and documentation were subjected to an independent assessment in August 2018 of the F&O closure. This assessment is described in Section A.6.

A.6. Finding Closure by Independent Assessment and Focused Scope Peer Review

An independent assessment of SCE&G's resolution of 32 F&Os was performed in August 2018 and is documented in PWROG-18050-P [58]. The process used for the independent assessment is outlined in Section X.1.3 (Close Out F&Os by Independent Assessment) of Appendix X to NEI 12-13 [5], which has been accepted by NRC, with two conditions (NRC Letter dated May 3, 2017 ([59]):

> (i) Use of New Methods: "A PRA method is new if it has not been reviewed by the NRC staff. There are two ways new methods are considered accepted by the NRC staff: (1) they have been explicitly accepted by the NRC (i.e., they have been reviewed, and the acceptance has been documented in a safety evaluation, frequently-asked-questions, or other publicly available organizational endorsement), or (2) they have been implicitly accepted by the NRC (i.e., there has been no documented denial) in multiple risk-informed licensing applications. The NRC's treatment of a new PRA method for closure of F&Os is described in the memorandum "U.S. Nuclear Regulatory Commission Staff Expectations for an Industry Facts and Observations Independent Assessment Process," dated May 1, 2017 (ADAMS Accession No. ML17121A271)."
>
> (ii) Use of Appendix X in Its Entirety: "In order for the NRC to consider the F&Os closed so that they need not be provided in submissions of future risk-informed licensing applications, the licensee should adhere to the guidance in Appendix X in its entirety. Following the Appendix X guidance will reinforce the NRC staff's confidence in the F&O closure process and potentially obviate the need for a more in-depth review."

The result of this independent assessment was intended to support future VCSNS license amendment request submittals, other regulatory interactions, risk-informed applications, and risk-informed decision-making. Finding resolutions reviewed and determined to have been adequately addressed through this independent assessment are considered "closed" and no longer relevant to the current PRA model, and thus need not be carried forward nor discussed in such future activities.

A.6.1. Selection of Independent Assessment Team Members

The independent assessment was led by Mr. Kenneth Kiper of Westinghouse Electric Company. Team members included: Dr. Annie Kammerer, and independent consultant; Dr. Ram Srinivasan, an independent consultant; Mr. Eddie Guerra of RIZZO Associates; Mr. Vince Andersen of JENSEN-HUGHES; and Mr. Bob Kirchner of JENSEN-HUGHES, representing Exelon Generation Company. The lead and reviewer qualifications have been reviewed by SCE&G and have been confirmed to be consistent with requirements in the ASME/ANS PRA Standard [4] and the guidelines of NEI-12-13 [5]. Consistent with the requirement in Section 1-6.2.2 of the ASME/ANS PRA Standard [4], the members of the independent assessment team were independent of the VCSNS SPRA. They were not involved in performing or directly supervising work on any PRA Element evaluated in the overall VCSNS SPRA.

See Section A.3 for the qualifications of Mr. Kenneth Kiper, Dr. Ram Srinivasan, Mr. Eddie Guerra, Mr. Vince Andersen, Mr. Bob Kirchner, and Dr. Annie Kammerer to support the independent assessment.

A.6.2. Pre-Review Activities

In preparation for the independent assessment associated with F&O close-out, SCE&G and the independent assessment team IAT performed the activities described in the following subsections.

A.6.3. Host Utility Preparation

SCE&G provided the complete and relevant review material to the independent assessment team in advance to allow the reviewers to prepare and conduct a more efficient technical review. As input to the review, SCE&G provided the following documentation:

- Exact wording of each original F&O within scope of the independent assessment;
- A summary description of how each F&O was dispositioned;
- SCE&G self-assessment of whether the F&O closure involved an upgrade or a maintenance activity, based on the definition of upgrade vs. maintenance documented in the PRA Standard [4];
- Documents that were revised to resolve the F&Os.

A.6.4. Offsite Review

All material generated in support to the F&O closure activities performed by SCE&G were provided to the independent assessment team two weeks before the onsite review and consensus session. The review team started the review and familiarization of the documentation.

A.6.5. Onsite Review and Consensus

During the onsite review and consensus session, the team achieved the following for each reviewed F&O:

- Consensus on the status of the F&O (i.e., CLOSED, OPEN or PARTIALLY CLOSED). This conclusion was reached through a review of the original basis and description of the F&O and on the technical work and documentation provided by SCE&G to resolve the issue identified in the F&O;
- Consensus on whether the activities performed to close the F&O are to be considered maintenance or upgrade, per the appropriate definition of the PRA Standard;
- If the F&O was associated with an SR that was originally judged as Not Met or Met at Capability Category CC I, upon confirming closure of the associated F&Os, the SR has been re-assessed to reach consensus on whether the intent of the SR is now Met or Met at capability category CC II or higher.

A.6.6. Treatment of "New Methods"

All of the changes to the VCSNS Unit 1 SPRA were classified as either PRA Maintenance or PRA Upgrade by the independent assessment. Therefore, no new methods were identified during the independent assessment.

A.6.7. Use of Remote Reviewers

The independent assessment team lead, Mr. Kenneth Kiper, as well as the lead SFR reviewer Mr. Eddie Guerra and the lead SPR reviewer Mr. Vince Anderson were present at the independent assessment location. The lead SHA reviewer Dr. Annie Kammerer, an SHA and SFR reviewer Dr. Ram Srinivasan, and an SPR reviewer Mr. Robert Kirchner participated remotely.

A.6.8. Status of Findings at End of Independent Assessment

The following bullets summarize the independent assessment conclusions for each high level requirement:

- **SHA:** 2 F&Os were closed and 1 F&O remained open. The 1 F&O that remained open is related to plant specific documentation of seismic activity between the CEUS ground motion model data and the time of the Peer Review with an assessment of the impact on the seismic hazard.

- **SFR:** 16 F&Os were closed, and 1 F&O remained open. The open F&O relates to use of an earlier methodology, which has since been improved upon, for assessing soil liquefaction.

- **SPR:** All 12 assessed F&Os were closed. Note that two SPR F&Os were not included in the Independent Assessment as delineated in Table A-5.

A.6.9. Final Independent Assessment Report

A final report was provided at the end of the independent assessment, which documented the review and its conclusions [58]. This report includes the following information:

- Descriptions of the F&O independent assessment process. See Section 2 of [58].
- Description of the scope of the independent assessment (i.e., identification and description of the findings being reviewed for closure). See Table 1-1 and Table 1-2 of [58].
- Identification of the SRs that the F&Os were written against, and the basis for the SR assessment from the peer review of record. See Table A-1 [58].
- A summary of the review team's decisions for each finding within the scope of the review, along with the rationale for determination of adequacy or inadequacy for closure of each finding in relation to the affected portions of the associated SR. If multiple SRs are referenced by a single finding, the affected portions of all associated SRs were addressed. See Section 3 and Table A-1 of [58].
- For each finding, assessment of whether the resolution was determined to be a PRA upgrade, maintenance update, or other, and the basis for that determination. See Table A-1 of [58].
- A summary of issues were identified by the independent assessment team that are directly related to the findings being closed. See Section 3.1, Table 3.2-3, and 3.3-3 of [58].

The report categorized each in-scope finding as "closed" or “open.” For each finding, the basis for the decision on closure was documented.

The final report included each of the independent assessment team members' resumes and summary of their experience as it applies to qualification guidelines of NEI guidance documents and the ASME/ANS PRA Standard. See Appendix B of [58].

A.6.10.Summary of Independent Assessment Team Conclusions

Of the 32 F&Os reviewed, the independent assessment team concurred that all except two (2) can be considered closed. As a result of the closure of the associated F&Os, two (2) SRs, originally judged as Not Met are now judged to be Met.

The independent assessment team recognized a significant amount of work invested in the resolution of the F&Os from the original peer review, including generation of new fragilities, additional sensitivity studies, improved documentation, and an additional walkdown for risk significant seismic-induced flood and fire sources. The independent assessment team concluded that, as a result of the closure of the associated F&Os, the V.C. Summer Unit 1 SPRA more realistically reflects the current seismic risk at the site.

Table A-1 identifies the SRs graded as not met or not Capability Category II, and the disposition for each, following the independent assessment. Table A-5 (at the end of this Appendix) presents summary of all open Finding F&Os and the disposition for each.

Table A-1 Summary of SRs Graded as Not Met or Capability Category I for Supporting Requirements Covered by the VCSNS SPRA Peer Review and Subsequent Independent Assessment of F&O Closure			
SR	**Assessed Capability Category**	**Associated Open Finding F&Os**	**Disposition to Achieve Met or Capability Category II**
SHA – Seismic Hazard Technical Element (from [46])			
SHA-H1	Not Met	H20-1	No expected technical impact to SPRA results; See Table A-2
SHA-I1	Not Met	24-7	No expected technical impact to SPRA results; See Table A-2
SFR – Seismic Fragility Technical Element (from [6])			
All SRs assessed as met by SPRA peer review [6] and subsequent independent assessment of F&O closure [58]			
SPR – Seismic System Analysis Technical Element (from [6])			
All SRs assessed as met by SPRA peer review [6] and subsequent independent assessment of F&O closure [58]			

A.6.11.Compliance of Independent Assessment with NRC Conditions

As indicated in Section A.6, the NRC's acceptance of the F&O closure process described in Appendix X to NEI 12-13 [5], as documented in [59], includes two conditions. The independent assessment of the finding closure for the VCSNS Unit 1 SPRA satisfies the conditions as follows:

(i) Use of New Methods: As indicated in Section A.6.6, new methods were not employed in the resolution of findings associated with the VCSNS Unit 1 SPRA. Therefore, this condition does not apply.

(ii) Use of Appendix X in its Entirety: The finding closure process, as outlined in Section A.6, encompasses all of the elements of Appendix X.

Therefore, the application of the Appendix X process to the closure of the findings identified during the VCSNS SPRA peer review is in conformance with the NRC's requirements.

A.7. Summary of Technical Adequacy of the SPRA for the 50.54(f) Response

The set of supporting requirements from the ASME/ANS PRA Standard [4] that are identified in Tables 6-4 through 6-6 of the SPID [2] 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 VCSNS SPRA model meets the expectations for PRA scope and technical adequacy as presented in RG 1.200, Revision 2 [45] as clarified in the SPID [2]. 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 CDF and LERF (Section 5)
- Summary of adaptations made in the internal events PRA model to produce the seismic PRA model and bases for the adaptations (Section 5)

Detailed archival information for the SPRA consistent with the listing in Section 4.1 of RG 1.200 Rev. 2 [45] is available if required to facilitate the NRC staff's review of this submittal.

The VCSNS SPRA reflects the as-built and as-operated plant as of the cutoff date for the SPRA, July 20, 2017. The SPRA model does not credit portable or offsite FLEX capabilities for response to extended loss of offsite power or loss of ultimate heat sink response.

The peer review observations and conclusions noted in Section A.4, the F&O closure review discussion in Section A.6, and the discussion in Section A.8 demonstrate that the VCSNS SPRA is technically adequate in all aspects for this submittal. Subsequent to the SPRA peer review, the peer review findings have been appropriately dispositioned, and the SPRA model has been updated to reflect these dispositions and further refine several fragility values. The results presented in this submittal reflect the updated model as of July 2018.

A.8. Summary of SPRA Capability Relative to SPID [2] Tables 6-4 through 6-6

The Owners Group performed a full scope peer review of the VCSNS internal events PRA and internal flooding PRA that forms the basis for the SPRA to determine compliance with ASME/ANS PRA Standard ([4] and [56]) and Regulatory Guide 1.200 [45] in June 2016. This review documented findings for all supporting requirements (SRs) which failed to meet at least Capability Category II.

The PWR Owners Group peer review of the VCSNS SPRA was conducted in April 2018. For the SHA elements a peer review was conducted in July 2017. An independent assessment of F&O closure was conducted in August 2018. The results of these peer reviews are discussed above, including resolution of SRs not assessed by the peer review as meeting Capability Category II, and resolution of peer review findings pertinent to this submittal. The peer review teams expressed the opinion that the VCSNS SPRA model is of good quality and integrates the seismic hazard, the seismic fragilities, and the systems-analysis aspects appropriately to quantify core damage frequency and large early release frequency. The general conclusion of the peer reviews was that the VCSNS SPRA is judged to be suitable for use for risk-informed applications.

- Table A-2 Provides a summary of the disposition of SRs judged by the peer reviews to be not met, or not meeting Capability Category II.
- Table A-5 provides a summary of the disposition of the open SPRA peer review findings.
- Table A-2 provides an assessment of the expected impact on the results of the VCSNS SPRA of those SRs and peer review Findings that have not been fully addressed.

Table A-2 Summary of Impact of Not Met SRs and Open Peer Review Findings		
SR # or F&O #	**Summary of Issue Not Fully Resolved**	**Impact on SPRA Results**
SHA-H1	The F&O relates to plant specific documentation of seismic activity between the CEUS ground motion model data and the time of the Peer Review.	See discussion for F&O H20-1 in Table A-5.
SHA-I1	The F&O closure report says that SHA-I1 remains not met, even though all the associated F&Os have been closed, because the open F&O 24-7 on soil liquefaction could have been considered applicable to SHA-I1 if the SHA and SFR peer reviews were done concurrently.	See discussion for F&O 24-7 in Table A-5.

A.9. Identification of Key Assumptions and Uncertainties Relevant to the SPRA Results

The ASME/ANS PRA Standard [4] includes a number of requirements related to identification and evaluation of the impact of assumptions and sources of uncertainty on the PRA results. NUREG-1855 [47] and EPRI 1016737 [48] 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 VCSNS 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 as a way to address a particular modeling uncertainty because there is not a single definitive approach. Plant-specific assumptions made for each of the VCSNS 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 VCSNS SPRA is listed in Table A-3.

Table A-3 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 AP1000 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 VCSNS site.	The seismic hazard reasonably reflects sources of uncertainty.
Seismic Fragilities	The peer review team noted that the seismic response analysis of buildings was supplemented by a number of sensitivity studies used to assess important modelling assumptions. These sensitivity studies included difference in response between single and multiple time histories; effect of spatial incoherency of ground motion in building response; and control building in-structure sensitivity study. These studies served to reinforce engineering judgments and assumptions implemented in the fragility analysis of SSCs.	Several of the sensitivity studies described in Section 5.7 of this report evaluate the impact of changes to fragilities on the SPRA results as one means of assessing the impact of fragilities uncertainties on the SPRA results. No changes to the model were recommended based on these results.
Seismic PRA Model	The peer review team noted that parametric uncertainty analyses and sensitivity studies are performed and risk importance results are provided.	A characterization of the mean SCDF and SLERF is provided in Section 5.6 of this report. Several sources of model uncertainty are discussed in Section 5.7 along with sensitivities performed to evaluate the impact of possible changes to address these.

A.10. Identification of Plant Changes Not Reflected in the SPRA

The VCSNS SPRA reflects the plant as of the cutoff date for the SPRA, which was *July 20, 2017*. Table A-4 lists significant plant changes subsequent to this date and provides a qualitative assessment of the likely impact of those changes on the SPRA results and insights.

Table A-4 Summary of Significant Plant Changes Since SPRA Cutoff Date	
Description of Plant Change	**Impact on SPRA Results**
The Seismic PRA model is based on the July 2017 Internal Events model update. This is the most recent Internal Events model. There have been no significant plant changes since that time that have warranted an update to the Internal Events model.	The SPRA is representative of the current plant.

Table A-5 Summary of Open Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
SFR-D1	24-7	The liquefaction potential was not considered in identification of failure modes that can affect the Service Water system. (This F&O originated from SR SFR-D1)	Although the GEI Project 1411090 report screens out the liquefaction potential for the West Embankment and much of the areas in the Pumphouse and Intake Structure, it does not completely rule out the liquefaction potential for one area where the saprolite was left in place below the Pumphouse and Intake Structure. The GEI Project 1411090 Report states that this area is not expected to undergo liquefaction during a Magnitude 7.5, 0.35 g peak ground acceleration earthquake. This only provides a lower bound capacity for this area. However, 13C4188-RPT-001 concludes that soils are not expected to undergo liquefaction.	Screen consequences of the potential liquefaction-induced failure modes from further consideration or consider the effects of liquefaction-induced settlement or lateral spreading on Pumphouse and Intake Structure fragility calculations.	Resolution Summary The project geotechnical screening report (GEI, 2014) provided a liquefaction evaluation of foundation and embankment soils near the SWPH. The evaluation concluded that only the saprolite layer is potentially susceptible to liquefaction. The embankment select fill, residual soils, and decomposed rock are not susceptible to liquefaction. A triggering evaluation was performed for the saprolite layer and factor of safety of 1.27 was determined for the project reference earthquake loading (0.70g PGA at rock) using the Youd 2001 method. Based on the above, a screening-based HCLPF for SWPH Embankment liquefaction was set to 0.70g and included in the final fragility report. The failure mode is liquefaction-induced settlement of structures or lateral spreading of surrounding embankment with the potential to restrict flow to or from the service water system. Note the above HCLPF still exceeds the 0.67g HCLPF applied for the SW Pond Dam failure. Therefore, further refinement of the above embankment HCLPF to remove potential conservatism is of limited value since dam failure results in loss of function for the SW system. The above resolution is consistent with the Possible Resolution identified in the F&O details.
SHA-C4, SHA-H1	H20-1	The PSHA for the VC Summer site was performed using the existing seismic source model described in NUREG-2115. HLR SHA-H, as modified in Regulatory Guide 1.200, states 'ENSURE, in light of established current information, the study meets the requirements in HLR-SHA-A thru HLR-SHA-G.' The NUREG-2115 source model was completed in 2012, using an earthquake catalog for the time period ending in 2008. (This F&O originated from SR SHA-H1)	The response to Question AMK-03 indicates that no evaluation of new seismic source information or an evaluation of the post 2008 seismicity was performed for the VC Summer site.	The response to Question AMK-03 proposes to utilize the assessment of new information performed for the Vogtle site as a basis for evaluating the impact of new information on the seismic hazard model for the VC Summer site. The proposed resolution document this assessment using the arguments presented in the response to Question AMK-03.	The CEUS 2011 EPRI Seismic data was used for both the PSHA's used in the SPRA models. VCSNS considered this data frozen at the time of the model development. The GMRS developed for the NTTF 2.1 screening process used the CEUS 2011 EPRI Seismic data and the NRC considers this hazard acceptable for use in the SPRA. The PSHA will be updated as part of the routine model update process during the next scheduled SPRA model update.

Table A-5 Summary of Open Finding F&Os and Disposition Status

SR	F&O	Description	Basis	Suggested Resolution	Disposition
SPR-B1 SPR-E2 SPR-E6	19-10	The previous IE-PRA peer review generated more than 90 Finding-level F&Os that had not been resolved at the time of the SPRA peer review. These F&Os are documented in the Self Assessment Report for SPR (LTR-RAM-18-15), along with possible resolutions and the potential impact of the possible resolutions on the SPRA model. Because of the broad nature of the F&Os against the IE-PRA model and the lack of actual resolutions, it was not possible for the SPRA Peer Review Team to assess the collective impact of the open F&Os on the SPRA model. Several examples are offered: Findings written against SRs HR-F2, G4, G5, & G6 address timing input to HRA. Timing is one of the key inputs to adjusting HEPs for seismic impact. Findings written against SRs SC-B3, B4, & B5 address the basic success criteria in the IE-PRA. This can directly impact accident sequences in the SPRA model. Findings written against SY-A22, B6, & B12 address issues of modeling HVAC systems. The absence of these systems in the IE-PRA carries over directly to the SPRA. (This F&O originated from SR SPR-B1)	The IE-PRA model is used as the foundation for the SPRA in many elements. Section 5-2.3 of the PRA Standard includes the statement, "It is also assumed that the internal-events, at-power PRA is in general conformance with Part 2." Supporting requirements SPR-B1 (IE, AS, SC, SY, DA, HR), SPR-E2 (QU), and SPR-E6 (LE) specifically invoke the requirements of technical elements in Part 2. It is important to understand the impact on the SPRA model of the actual changes in the IE-PRA model made to resolve F&Os from the IE-PRA peer review.	When the IE-PRA peer review F&Os have been resolved, verify the impact of IE-PRA model changes on the SPRA, specifically for model elements AS, SC, SY, HR, QU, and LE.	This F&O was not included in the Finding Closure by Independent Assessment. Extensive discussion on the internal events PRA peer review was provided in letter RC-18-0091 "License Amendment Request – LAR 16-01490 License Amendment Request to Revise the National Fire Protection Association (NFPA) Standard 805 Program," submitted August 30, 2018. Specifically, Attachment 8 of Enclosure 1 contains the discussion. Resolution of the internal events model F&Os is not expected to result in significant changes to CDF or LERF.
SPR-C1	19-7	FLEX equipment is not modeled in the SPRA. Not taking credit for FLEX may result in an overly conservative model. In particular, FLEX equipment may be important to realistically address the safe-stable state for a seismic event (see related F&O 26-2). (This F&O originated from SR SPR-C1)	This SR requires justification for conservatisms in the SPRA model. The identified conservatisms may be more significant when other issues (e.g., safe-stable state issue) are addressed.	Provide a justification for the conservatisms that may result from excluding the FLEX equipment from the SPRA model.	This F&O was not included in the Finding Closure by Independent Assessment. Consistent with regulatory position, FLEX equipment is not included in the seismic PRA model. This was an intentional decision. Modeling of FLEX equipment may reduce the importance of loss of offsite power in future applications.