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Washington, D. C. 20555-0001

Vogtle Electric Generating Plant Units 1 and 2
Fukushima Near-Term Task Force Recommendation 2.1: Seismic
Seismic Probabilistic Risk Assessment

References:

1. NRC Letter, *Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3 of the NTTF Review of Insights from the Fukushima Dai-ichi Accident,* dated March 12, 2012.
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." ML12333A170.
3. Letter to the NRC, Vogtle Electric Generating Plant Units 1 and 2, Seismic Hazard and Screening Report for CEUS Sites, dated March 31, 2014. ML14092A019.
4. NRC Letter, Vogtle Electric Generating Plant Units 1 and 2, Staff Assessment of Seismic Hazard Reevaluations Pertaining to Recommendation 2.1, dated April 20, 2015. ML15054A296.
5. NRC Letter, Final Determination of Licensee Seismic Probabilistic Risk Assessments, dated October 27, 2015. ML15194A015.

Ladies and Gentlemen:

On March 12, 2012, the Nuclear Regulatory Commission (NRC) issued a request for information pursuant to 10 CFR 50.54(f) associated with the recommendations of the Fukushima Near-Term Task Force (NTTF) (Reference 1). Enclosure 1 of Reference 1 requested each licensee to reevaluate the seismic hazards at their sites using present-day NRC requirements and guidance, and to identify actions taken or planned to address plant-specific vulnerabilities associated with the updated seismic hazards.

Reference 2 contains industry guidance developed by EPRI that provide the screening, prioritization and implementation details for the resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic. The SPID (Reference 2) was used to compare the reevaluated seismic hazard to the design basis hazard. The Vogtle Electric Generating Plant (VEGP), Units 1 and 2 reevaluated seismic hazard (Reference 3) concluded that the ground motion response spectrum (GMRS) exceeded the design basis seismic response spectrum in the 1 to 10 Hz range, and therefore a seismic probabilistic risk assessment was required.

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Reference 4 contains the NRC Staff Assessment of the VEGP Units 1 and 2 seismic hazard submittal which concluded that the reevaluated seismic hazard prepared for VEGP is suitable for other activities associated with the NRC Near-Term Task Force Recommendation 2.1: Seismic.

Reference 5 contains the NRC letter "Final Determination of Licensee Seismic Probabilistic Risk Assessments." In that letter (Table 1a - Recommendation 2.1 Seismic – Information Requests) the NRC instructed VEGP Units 1 and 2 to submit an SPRA by March 31, 2017.

Enclosure 1 of this letter contains the VEGP Units 1 and 2, Seismic Probabilistic Risk Assessment (SPRA) Summary Report which provides the information requested in Enclosure 1, Item (8) B. of the 10 CFR 50.54(f) letter.

In accordance with PWROG-14001, "PRA Model for the Generation III Westinghouse Shutdown Seal," a PWR Owners' Group project (PA-RMSC-1423) was initiated to evaluate the Generation III Westinghouse Reactor Coolant Pump (RCP) SHIELD® Passive Thermal Shutdown Seal (Generation III SDS). Coupling the current Emergency Operating Procedures with the results of the project, cold leg temperatures which could adversely impact the operation of the Generation III SDS would not be reached at VEGP.

This letter contains no NRC commitments. If you have any questions, please contact John Giddens at 205.992.7924.

Mr. J. J. Hutto states he is the Regulatory Affairs Director for Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and, to the best of his knowledge and belief, the facts set forth in this letter are true.

Respectfully submitted,

JJH

J. J. Hutto
Regulatory Affairs Director

JJH/JMG/GLS

Sworn to and subscribed before me this 21 *day of* March*, 2017.*

Laura L. Crumpton
Notary Public

My commission expires: 10 · 8 · 2017

Enclosure: Vogtle Electric Generating Plant – Units 1 and 2 Seismic Probabilistic Risk Assessment Summary Report

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Vogtle Electric Generating Plant – Units 1 and 2
Fukushima Near-Term Task Force Recommendation 2.1: Seismic
Seismic Probabilistic Risk Assessment

Enclosure

Vogtle Electric Generating Plant – Units 1 and 2
Seismic Probabilistic Risk Assessment
Summary Report

VOGTLE ELECTRIC GENERATING PLANT UNITS 1 AND 2 SEISMIC PROBABILISTIC RISK ASSESSMENT IN RESPONSE TO 10 CFR 50.54(f) LETTER WITH REGARD TO NTTF 2.1 SEISMIC

VERSION 0

MARCH 2017
SUMMARY REPORT

VEGP UNITS 1 AND 2 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 PRA (SPRA) has been developed to perform the seismic risk assessment for Plant Vogtle Units 1 and 2. The SPRA shows that the point estimate seismic Core Damage Frequency (SCDF) is 2.8×10^{-6}/yr and the seismic Large Early Release Frequency (SLERF) is 3.3×10^{-7}/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 the reduction of seismic risk. These sensitivity studies demonstrated that the model results were robust to the modeling and assumptions used. No seismic hazard vulnerabilities were identified, and no plant actions have been taken or are planned given the insights from the seismic risk assessment.

1.0 Purpose and Objective

Following the accident at the Fukushima Dai-ichi nuclear power plant resulting from the March 11, 2011, Great Tohoku Earthquake and subsequent tsunami, the 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 10 CFR 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 letter (commonly referred to as 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 Vogtle Electric Generating Plant (VEGP) Units 1 and 2 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 PRA (SPRA) has been developed to perform the seismic risk assessment for VEGP Units 1 and 2 in response to the 50.54(f) letter, specifically item (8) in Enclosure 1 of the 50.54(f) letter.

This report describes the seismic PRA developed for VEGP Units 1 and 2 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 [2]. 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 VEGP Units 1 and 2, 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.

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 VEGP Units 1 and 2 seismic PRA. For clarification, throughout the remainder of this report, there are some references to VEGP. While the site will eventually be a four-unit site, for this report, VEGP means Vogtle Electric Generating Plant Units 1 and 2.

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, Fussell-Vesely 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. SSC fragility values with reference to the method of seismic qualification, the dominant failure mode(s), and the source of information
   iii. Seismic fragility parameters
   iv. Important findings from plant walkdowns and any corrective actions taken
   v. Process used in the seismic plant response analysis and quantification, including the specific adaptations made in the internal events PRA model to produce the seismic PRA model and their motivation
   vi. Assumptions about containment performance
(3) Description of the process used to ensure that the SPRA is technically adequate, including the dates and findings of any peer reviews
(4) Identified plant-specific vulnerabilities and actions that are planned or taken

Note that 50.54(f) letter Enclosure 1 paragraphs 1 through 6, regarding the seismic hazard evaluation reporting, also apply, but have been satisfied through the previously submitted VEGP Seismic Hazard Submittal [3]. Further, 50.54(f) letter Enclosure 1 paragraph 9 requests information on the Spent Fuel Pool. This information is being submitted separately.

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 VEGP SPRA has been developed and documented in accordance with the SPID. The main elements of the SPRA performed for VEGP in response to the 50.54(f) Seismic letter correspond to those described in Section 6.1.1 of the SPID, i.e.:

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

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

The VEGP SPRA and associated documentation has been peer reviewed against the ASME/ANS PRA Standard [4] in accordance with the process defined in NEI 12-13 [5], as documented in the VEGP SPRA Peer Review Report. The VEGP 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 provides information related to the VEGP seismic hazard analysis.

Section 4 provides information related to the determination of seismic fragilities for VEGP SSCs included in the seismic plant response.

Section 5 provides information regarding the plant seismic response model (seismic accident sequence model) and the quantification of results.

Section 6 summarizes the results and conclusions of the SPRA, including identified plant seismic issues and actions taken or planned.

Section 7 provides references.

Section 8 provides a list of acronyms used.

Appendix A provides an assessment of SPRA Technical Adequacy for Response to NTTF 2.1 Seismic 50.54(f) Letter, including a summary of VEGP 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
2	Summary of the methodologies used to estimate the SCDF and SLERF	Sections 3, 4, 5
2i	Methodologies used to quantify the seismic fragilities of SSCs, together with key assumptions	Section 4
2ii	SSC fragility values with reference to the method of seismic qualification, the dominant failure mode(s), and the source of information	Tables 5.4-4 and 5.5-2 provide fragilities (Am and beta), failure mode information, and method of seismic quantification of fragilities for the top risk significant SSCs based on the Fussell-Vesely (F-V) risk importance measure. Seismic qualification reference is not provided as it is not relevant to development of SPRA.
2iii	Seismic fragility parameters	Tables 5.4-4 and 5.5-2 provide fragilities (Am and beta) information for the top risk significant SSCs based on the Fussell-Vesely (FV) risk importance measure.
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, 4.4, 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	App. 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 addresses this topic. No vulnerabilities were identified or actions planned as a result of the SPRA.

Table 2-2 Cross-Reference for Additional SPID [2] 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 model quantification sensitivities.
The level of detail needed in the submittal should be sufficient to make necessary regulatory decisions as a part of NTTF Phase 2 activities.	Entirety of the 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 VEGP Seismic Hazard and Plant Response

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

VEGP is a dual unit Westinghouse 4-loop pressurized water reactor plant located approximately 15 miles east-northeast of Waynesboro, Georgia and 26 miles southeast of Augusta, Georgia, adjacent to the Savannah River. The regional and site (local) geology is described in additional detail in the VEGP NTTF 2.1 Seismic Hazard submittal [3]. VEGP is a soil site with 88 feet of backfill on top of an in-situ strata identified as Blue Bluff Marl (BBM). The following Seismic Category I structures are founded directly on the BBM: the Auxiliary Building (AB), Nuclear Service Cooling Water (NSCW) towers, and instrumentation cavity of the Containment. The remaining Seismic Category I structures are founded on backfill. The soil profile was developed using the original Vogtle Units 1 and 2 borehole data supplemented with the latest borehole data taken for the Vogtle Units 3 and 4 new construction and the Dry Cask Storage facility. Additional site description and composite profile development are described in the VEGP NTTF 2.1 Seismic Hazard submittal [3].

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. Detailed information regarding the VEGP 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 VEGP SPRA.

3.1.1 Seismic Hazard Analysis Methodology

For the VEGP SPRA, the following method was used.

As reported in the VEGP NTTF 2.1 Seismic Hazard submittal [3], the control point (power block) hazard curves were used to develop uniform hazard response spectra (UHRS) and the ground motion response spectrum (GMRS). The UHRS were calculated using log-log interpolation to determine the spectral acceleration at each spectral frequency for the 10^{-4} and 10^{-5} per year hazard levels. The GMRS was calculated from the 10^{-4} and 10^{-5} UHRS at each spectral frequency. The control point elevation is defined in the VEGP NTTF 2.1 Seismic Hazard submittal [3] as being at plant grade at an elevation of 220 feet mean sea level (MSL), consistent with the Plant Vogtle Units 1 and 2 FSAR. Table 2.4-1 and Figure

2.4-1 in the VEGP NTTF 2.1 Seismic Hazard submittal [3] provide the mean UHRS for 10^{-4} and 10^{-5} and GMRS accelerations for a range of spectral frequencies.

The Reference Earthquake used in developing the building response, and subsequently in the fragility evaluation corresponds to the 10^{-4} UHRS at plant grade. The 10^{-4} horizontal UHRS at plant grade has a PGA of 0.436g.

UHRS were developed at specific horizons at the 10^{-4} and 10^{-5} hazard levels, along with the corresponding strain-compatible properties. This information was used in developing input motion for soil-structure interaction (SSI) analysis. Two types of motion were developed, outcrop UHRS and truncated soil column response (TSCR).

Similar to the site response analysis described in the VEGP NTTF 2.1 Seismic Hazard submittal [3], the rock high frequency (HF) and low frequency (LF) spectra at 10^{-4} and 10^{-5} hazard levels were applied at bedrock and are propagated through two sets of 60 simulated profiles. The 5% damping outcrop acceleration response spectra (ARS) at specific horizons were computed. The log-mean (median) results including strains, shear-wave velocity, and damping, are calculated, along with the corresponding log-standard deviations. At each hazard level, the arithmetic mean HF and LF ARS for the two soil columns are arithmetically averaged resulting in the uniform hazard response spectra at the considered horizon.

To calculate TSCR at the considered horizon, the soil layers above that horizon were truncated and site response analysis was repeated using the iterated strain-compatible properties, resulting from the site response analysis runs using the full soil column, and without further iterations.

The site response analyses [18, 39] and the fragility notebook [16] provide the horizontal 1E-4 UHRS at the elevations as shown in Figure 3.1-1. Plant grade, El 220 ft, is identified as 0 ft outcrop; the other two horizons are identified as depth below grade.

The VEGP Seismic Hazard Submittal [3] used Approach 3 as defined in NUREG/CR-6728 [34] to incorporate site amplification factors with the site rock hazard to calculate seismic hazard curves at the seven oscillator frequencies (0.5, 1, 2.5, 5, 10, 25 and PGA (100) Hz) at the ground surface. This seismic hazard approach resulted in the reported uniform hazard response spectra (UHRS) and the ground motion response spectrum (GMRS) at the ground surface (EL 220 ft) [3].

To provide UHRS at the ground surface, as well as at other foundation elevations, for the purpose of subsequent soil-structure interaction (SSI) analysis, Approach 2A [34] was used in order to readily obtain strain compatable soil profiles. In the case of the Vogtle site, the difference between the UHRS and GMRS calculated by the two approaches was determined to be insignificant.

The methodology for obtaining the vertical response spectra is discussed in Section 3.1.4.

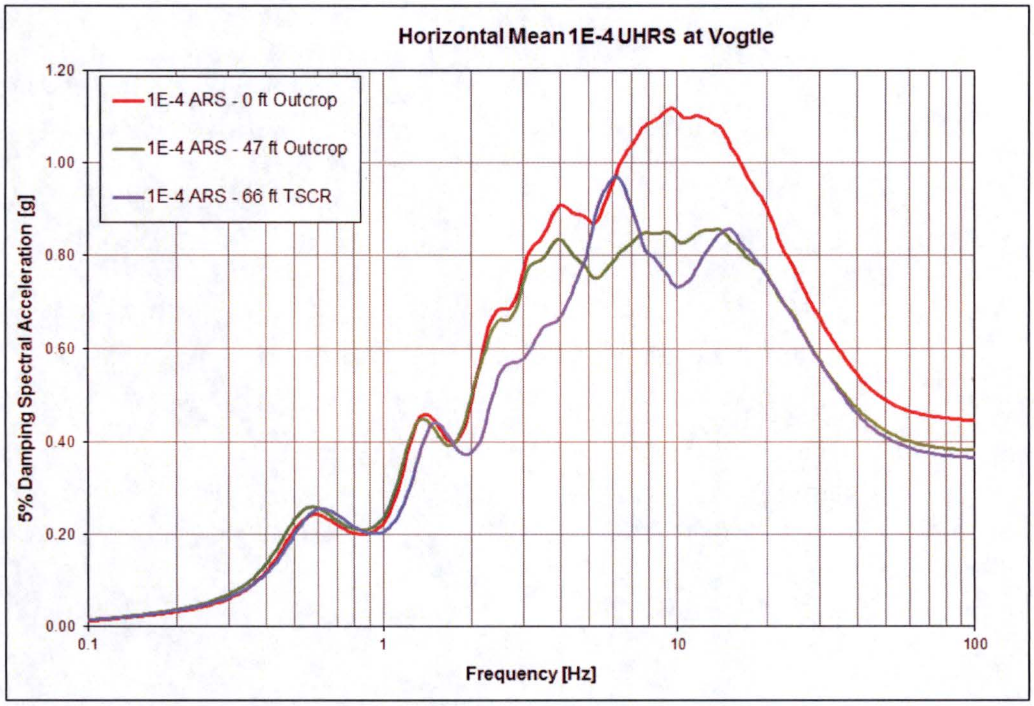

Figure 3.1-1 Horizontal Mean 1E-4 UHRS at Vogtle Site

3.1.2 Seismic Hazard Analysis Technical Adequacy

The VEGP SPRA hazard methodology and analysis associated with the horizontal response spectra at the control point were submitted to the NRC as part of the VEGP Seismic Hazard Submittal [3], and found to be technically acceptable by NRC for application to the VEGP SPRA [29].

The VEGP hazard analysis was also subjected to an independent peer review against the pertinent requirements in the PRA Standard [4]. 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, is described in Appendix A.

3.1.3 Seismic Hazard Analysis Results and Insights

Table 3.1-1 provides the final seismic hazard results used as input to the VEGP 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.

Uncertainties in the PSHA result from uncertainties in input models and parameters. These have been investigated for the VEGP SPRA [17]. As expected, background sources were found to have a large contribution to the 10 Hz spectral acceleration (SA) hazard,

Table 3.1-1 VEGP Mean and Fractile Exceedance Frequencies

	Exceedance Frequencies (/yr)			
PGA (g)	**0.16**	**0.5**	**MEAN**	**0.84**
0.1	7.23E-04	1.49E-03	1.93E-03	3.09E-03
0.15	3.19E-04	7.45E-04	1.11E-03	1.87E-03
0.3	6.09E-05	1.49E-04	2.87E-04	4.50E-04
0.5	1.36E-05	3.52E-05	6.78E-05	9.51E-05
0.75	2.76E-06	8.35E-06	1.49E-05	2.13E-05
1	6.36E-07	2.16E-06	3.72E-06	5.58E-06
1.5	3.84E-08	1.42E-07	2.50E-07	3.84E-07
2	4.00E-09	1.00E-08	1.60E-08	2.50E-08
3	1.53E-10	2.19E-10	2.23E-10	4.43E-10

and a repeated-large-magnitude-earthquake (RLME) source (Charleston) has a large contribution to the 1 Hz SA hazard. Note that the high frequency hazard is typically dominated by closer, moderate sized (background) earthquakes, and larger distant RLME events tend to be more important to low frequency hazard. For this reason, sensitivities to background sources were investigated for 10 Hz SA, and sensitivities to RLME sources were investigated for 1 Hz SA [17]. Sensitivity to site amplification model was also investigated [18].

The main contributors to hazard uncertainty are the ground motion prediction equations (GMPEs) used for hazard calculations, and the characteristic magnitude of the Charleston RLME source. The GMPEs contribute to uncertainty at both high and low spectral frequencies, at spectral amplitudes corresponding to mean annual frequencies of 10^{-4} and 10^{-5}. The characteristic magnitude of the Charleston source contributes to uncertainty primarily for low spectral frequencies, because the Charleston source has a lower contribution to hazard at high frequencies.

A review was performed [17] of the earthquake catalog used by EPRI for the 2012 hazard study [35]. It was determined that from January 1, 2009 through February 29, 2016, four earthquakes of magnitude M2.9 or greater were recorded within 320 km of the site. Because this is considerably lower than the frequency that would be expected from the mean annual rates of seismicity modeled for seismic sources by the EPRI 2012 study [35], it was concluded that the EPRI 2012 study [35] rates do not under-predict the seismicity observed during the period subsequent to that study. Furthermore, given the relatively short period of additional time covered by the updated catalog compared to the total period of time covered by the EPRI 2012 catalog [35], extending the catalog and re-computing new seismicity rates would result in only a very slight decrease in the activity rate in the study region. It was therefore concluded that the EPRI 2012 [35] seismicity parameters are adequate for evaluation of the seismic hazard at VEGP.

In the SPRA plant model, described in Section 5, the hazard data in Table 3.1-1 was discretized into 14 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	Interval Upper Bound	Representative Magnitude PGA (g)	Interval Mean Frequency
%G01	0.1	0.15	0.12	8.20E-04
%G02	0.15	0.3	0.21	8.23E-04
%G03	0.3	0.4	0.35	1.52E-04
%G04	0.4	0.5	0.45	6.71E-05
%G05	0.5	0.6	0.55	3.18E-05
%G06	0.6	0.7	0.65	1.61E-05
%G07	0.7	0.8	0.75	8.64E-06
%G08	0.8	0.9	0.85	4.79E-06
%G09	0.9	1	0.95	2.71E-06
%G10	1	1.1	1.05	1.55E-06
%G11	1.1	1.2	1.15	9.00E-07
%G12	1.2	1.5	1.34	1.02E-06
%G13	1.5	2	1.73	2.34E-07
%G14	2		2.2	1.60E-08

3.1.4 Horizontal and Vertical Response Spectra

This section provides the control point horizontal and vertical response spectra.

The 1E-4, 1E-5 and 1E-6 UHRS, along with the GMRS, at the control point are plotted in Figure 3.1-2. The development of the control point response spectra is described in detail in the VEGP NTTF 2.1 Seismic Hazard submittal [3].

The vertical response spectra were developed based on the corresponding horizontal response spectra, by scaling with an appropriate V/H function. The development of the V/H function is documented in the site response analysis [18] and the Vogtle Electric Generating Plant Units 3 and 4 Early Site Permit application (ESP) [36]. The acceptance of the Vogtle 3 and 4 ESP V/H function is provided in the NRC SER, NUREG 1923 [37].

Table 3.1-3 summarizes the horizontal and vertical response spectra at the control point. Figure 3.1-3 is a plot of the V/H function. Figure 3.1-4 provides a plot of the horizontal and vertical mean 1E-4 UHRS at the control point.

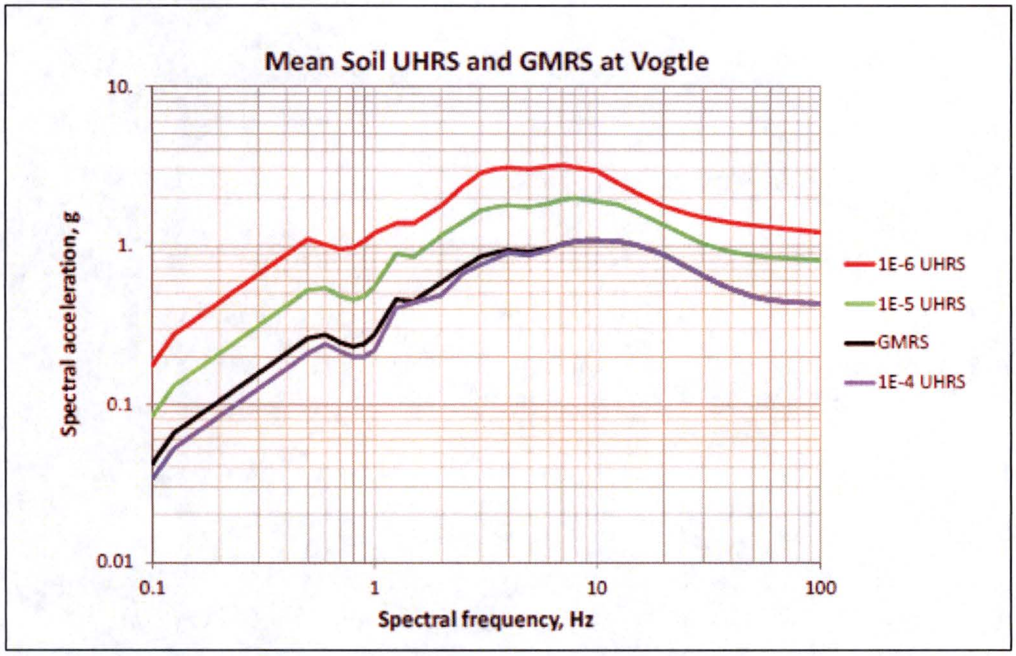

Figure 3.1-2 1E-4, 1E-5, AND 1E-6 UHRS AND GMRS AT THE CONTROL POINT

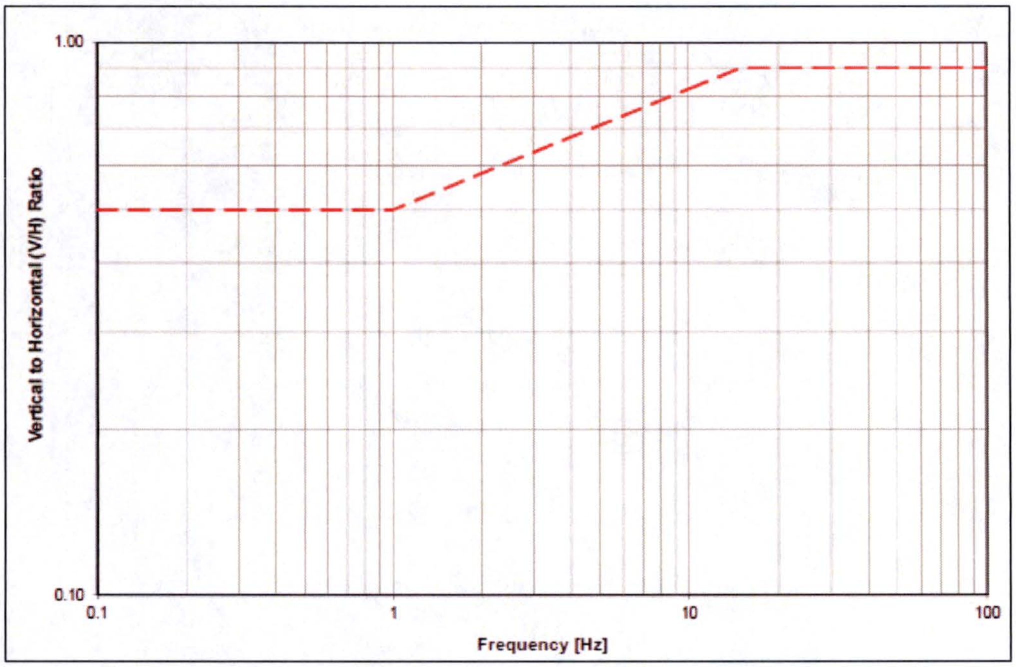

Figure 3.1-3 VERTICAL TO HORIZONTAL (V/H) RATIO FUNCTION

Table 3.1-3 Horizontal and Vertical Response Spectra at the Control Point

Frequency (Hz)	Horizontal 1E-4 UHRS (g)	Horizontal 1E-5 UHRS (g)	Horizontal GMRS (g)	V/H Ratio	Vertical 1E-4 UHRS (g)	Vertical 1E-5 UHRS (g)	Vertical GMRS (g)
100	4.36E-01	8.14E-01	4.36E-01	0.9	3.92E-01	7.33E-01	3.92E-01
90	4.38E-01	8.20E-01	4.38E-01	0.9	3.94E-01	7.38E-01	3.94E-01
80	4.41E-01	8.27E-01	4.41E-01	0.9	3.97E-01	7.44E-01	3.97E-01
70	4.47E-01	8.36E-01	4.47E-01	0.9	4.02E-01	7.52E-01	4.02E-01
60	4.58E-01	8.49E-01	4.58E-01	0.9	4.12E-01	7.64E-01	4.12E-01
50	4.80E-01	8.71E-01	4.80E-01	0.9	4.32E-01	7.84E-01	4.32E-01
40	5.34E-01	9.18E-01	5.34E-01	0.9	4.81E-01	8.26E-01	4.81E-01
35	5.83E-01	9.64E-01	5.83E-01	0.9	5.25E-01	8.68E-01	5.25E-01
30	6.51E-01	1.04E+00	6.51E-01	0.9	5.86E-01	9.36E-01	5.86E-01
25	7.48E-01	1.17E+00	7.48E-01	0.9	6.73E-01	1.05E+00	6.73E-01
20	8.83E-01	1.36E+00	8.83E-01	0.9	7.95E-01	1.22E+00	7.95E-01
15	1.02E+00	1.65E+00	1.02E+00	0.9	9.18E-01	1.49E+00	9.18E-01
12.5	1.07E+00	1.82E+00	1.07E+00	0.865	9.26E-01	1.57E+00	9.26E-01
10	1.09E+00	1.91E+00	1.09E+00	0.824	8.98E-01	1.57E+00	8.98E-01
9	1.09E+00	1.95E+00	1.09E+00	0.806	8.79E-01	1.57E+00	8.79E-01
8	1.07E+00	2.00E+00	1.07E+00	0.785	8.40E-01	1.57E+00	8.40E-01
7	1.02E+00	1.95E+00	1.03E+00	0.763	7.78E-01	1.49E+00	7.86E-01
6	9.36E-01	1.84E+00	9.64E-01	0.738	6.91E-01	1.36E+00	7.11E-01
5	8.76E-01	1.77E+00	9.21E-01	0.709	6.21E-01	1.25E+00	6.53E-01
4	9.03E-01	1.80E+00	9.39E-01	0.676	6.10E-01	1.22E+00	6.35E-01
3.5	8.33E-01	1.76E+00	9.09E-01	0.656	5.46E-01	1.15E+00	5.96E-01
3	7.62E-01	1.67E+00	8.55E-01	0.635	4.84E-01	1.06E+00	5.43E-01
2.5	6.69E-01	1.42E+00	7.31E-01	0.61	4.08E-01	8.66E-01	4.46E-01
2	4.87E-01	1.16E+00	5.87E-01	0.581	2.83E-01	6.74E-01	3.41E-01
1.5	4.39E-01	8.55E-01	4.49E-01	0.546	2.40E-01	4.67E-01	2.45E-01
1.25	4.06E-01	8.99E-01	4.60E-01	0.525	2.13E-01	4.72E-01	2.42E-01
1	2.21E-01	5.53E-01	2.76E-01	0.5	1.11E-01	2.77E-01	1.38E-01
0.9	2.00E-01	4.81E-01	2.42E-01	0.5	1.00E-01	2.41E-01	1.21E-01
0.8	2.00E-01	4.60E-01	2.33E-01	0.5	1.00E-01	2.30E-01	1.17E-01
0.7	2.17E-01	4.83E-01	2.47E-01	0.5	1.09E-01	2.42E-01	1.24E-01
0.6	2.42E-01	5.41E-01	2.76E-01	0.5	1.21E-01	2.71E-01	1.38E-01
0.5	2.10E-01	5.26E-01	2.62E-01	0.5	1.05E-01	2.63E-01	1.31E-01
0.4	1.68E-01	4.20E-01	2.10E-01	0.5	8.40E-02	2.10E-01	1.05E-01
0.35	1.47E-01	3.68E-01	1.84E-01	0.5	7.35E-02	1.84E-01	9.20E-02
0.3	1.26E-01	3.15E-01	1.57E-01	0.5	6.30E-02	1.58E-01	7.85E-02
0.25	1.05E-01	2.63E-01	1.31E-01	0.5	5.25E-02	1.32E-01	6.55E-02
0.2	8.40E-02	2.10E-01	1.05E-01	0.5	4.20E-02	1.05E-01	5.25E-02
0.15	6.30E-02	1.58E-01	7.87E-02	0.5	3.15E-02	7.90E-02	3.94E-02
0.125	5.25E-02	1.31E-01	6.56E-02	0.5	2.63E-02	6.55E-02	3.28E-02
0.1	3.36E-02	8.41E-02	4.20E-02	0.5	1.68E-02	4.21E-02	2.10E-02

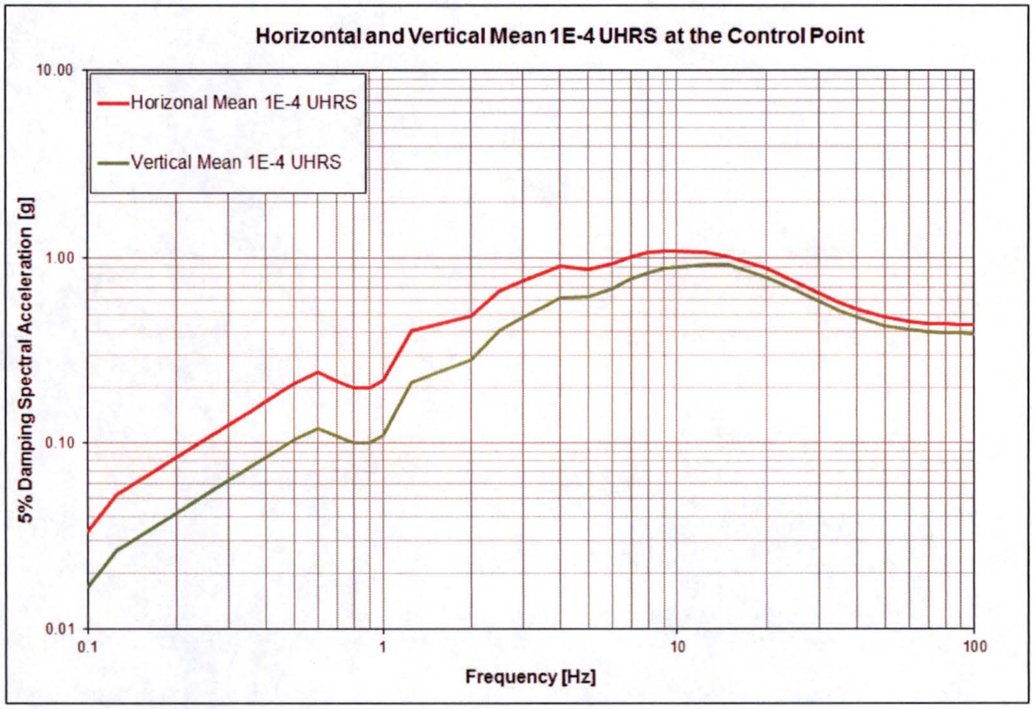

Figure 3.1-4 Horizontal and Vertical Mean 1E-4 UHRS at the Control Point

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

4.1 Seismic Equipment List

For the VEGP SPRA, a seismic equipment list (SEL) was developed that includes those SSCs that are important to achieving safe shutdown following a seismic event, and to mitigating radioactivity release if core damage occurs, and that are included in the SPRA model. The methodology used to develop the SEL is generally consistent with the guidance provided in the EPRI Seismic PRA Implementation Guide [10].

4.1.1 SEL Development

The following is a summary of items considered in developing the SEL [8].

The first step in developing the SEL was to determine the potential initiating events that could occur as a result of a seismic event. Initiating events considered could occur either directly as a result of the earthquake or due to random or consequential events that occur subsequent to the earthquake. The process of identification of potential initiating events used the internal events PRA for guidance.

Based on the internal events PRA and review of other potential seismic initiators, the primary seismic initiators identified were loss of offsite power (LOSP), loss of coolant accidents (LOCAs), reactor pressure vessel failure, secondary line break, and station blackout (SBO). The scope of the SPRA is power operation, therefore low power and shutdown states were not considered.

The safety functions that would be required to respond to the initiating events identified above were determined based on EPRI NP 6041-SL [7] and NUREG 1407 [15]. These safety functions are:

- Reactivity control
- Reactor coolant system pressure control
- Reactor coolant system inventory control
- Decay heat removal
- Containment isolation and integrity

The frontline systems used to meet the five safety functions were identified from the VEGP internal events PRA. In addition to the frontline systems, the required support systems were identified. However, unlike the internal events PRA, only systems that do not require offsite power were selected. Because the offsite power grid, switchyard insulators, and large transformers have relatively low seismic capacity, they cannot be relied on to provide power after a major earthquake. Only systems that can be supported by the onsite emergency AC power sources are considered. For Vogtle 1&2, the IPEEE Seismic Margin Analysis (SMA) safe shutdown equipment list (SSEL) was used as the initial list of equipment that would be used to mitigate seismic events. The SSEL already considered the five safety functions listed above, and contains useful information such as the equipment building and elevation, the normal and desired position for active components, the official mark number, the equipment drawing reference, and the equipment category.

Enhancements to the SSEL were made for the following reasons:

- Systems and equipment have been revised since the SMA SSEL was developed.
- The scope of the SMA was limited to consideration of two success paths, while the SPRA considers the broader accident sequence paths and associated systems and equipment.
- Additional seismic initiators must be considered, such as larger LOCAs, since the SMA only considered loss of offsite power and small LOCA.

The enhancements were identified by using system P&IDs and electrical diagrams to ensure that all necessary components are on the SEL.

The following types of equipment were added to the SEL:

- Components required to maintain pressure boundary integrity of the modeled systems.
- Active valves (and other components) that may have been screened from the SSEL or internal events PRA model but which could be transferred to an undesired state due to seismic-induced relay chatter.
- Reactor coolant system components, including: Reactor pressure vessel (and supports); Reactor internals; Control rods; Steam generators; Reactor coolant pumps (for RCS integrity, since they would not have power); Pressurizer; and Main RCS piping.
- Distribution systems (i.e., piping, HVAC ducting, and cable trays), treated as single distributed system entries in the SEL.
- Electrical panels, cabinets, and instrument racks need to provide emergency power and control for components on the SEL, including main control room benchboards and reactor protection system (RPS) cabinets.
- Equipment or instrumentation that would be required per the plant emergency procedures after an earthquake.

In addition, the plant areas in which operators would need to perform seismic response actions were reviewed for accessibility and evaluated for potential impact.

Components required for maintaining containment integrity were included in the SEL. These include SSCs related to containment isolation (such as containment isolation signals and valves) and containment pressure suppression and heat removal (such as the containment fan cooler units and containment sprays).

The structures associated with the SEL equipment are the following:

- Auxiliary Feedwater (AFW) Pump House
- Auxiliary Building
- Containment (Reactor Building)
- Control Building
- Diesel Generator Building
- Fuel Handling Building
- Nuclear Service Water Cooling Towers

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 and backdraft dampers
- Manual valves and dampers, including fire dampers
- Small spring-operated relief valves
- Small passive in-line filters that are supported only by the piping or ducting
- Heat tracing

In addition, instrumentation that is not required for mitigation of the seismic accident sequence (generally local instrumentation that is not part of a plant procedure that would be implemented during a seismic event) was not included in the SEL.

Equipment that is captured through "rule-of-the-box" considerations, e.g., equipment contained on a skid or in a cabinet, that can be subsumed into the major skid equipment or into the cabinet, was also not explicitly included on the SEL. For such equipment, the seismic fragilities for the containing equipment consider all of the equipment in the "box."

As a check on the SEL, the list of basic events in the internal events PRA was reviewed to identify additional systems and equipment that should be included in the SPRA, and the SEL. Systems and components that rely on offsite power were excluded.

The resulting SEL for each unit includes approximately 950 components for each unit.

4.1.2 Relay Chatter/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 VEGP SPRA, in accordance with SPID [2] Section 6.4.2 and ASME/ANS PRA Standard [4] Section 5-2.2. The evaluation resulted in most relay chatter scenarios screened from further evaluation for reasons such as no impact to component function, other components in the circuit that would prevent undesired impact, a self-correcting condition in which a signal would restore the proper position, or seismic qualification as part of the equipment containing the relay.

The unscreened relays in each unit were considered in the SPRA fragility and evaluated for inclusion in the model. The relays that were ultimately included in the SPRA model are listed in table 4.1-1.

A systematic evaluation of spurious trips of breakers was also performed for low and medium voltage switchgear. The functionality of breakers was evaluated either through the EPRI NP 6041-SL [7] value or through test response spectra evaluation. The major types of breakers at the plant are vacuum, air and molded case circuit breakers. Molded case circuit breakers inherently have high seismic capacity. The switchgear which houses air and vacuum type breakers are evaluated through EPRI NP 6041-SL [7] proxy evaluation. The seismic capacities used in this evaluation are the same or lower than EPRI generic equipment ruggedness spectra (GERS) given for low and medium voltage switchgears in EPRI NP-5223-SL [25].

This evaluation meets the intent of the high frequency screening requirement in Section 3.4.1 of the SPID [2]. Section 4.4.2 of this report provides discussion on seismic fragility evaluation of the critical relays.

Table 4.1-1 Summary of Unscreened Relays from Each Unit Included in SPRA Model

Relay	Function	Disposition
AFW AOV Trip Relay	Trips the AFW Turbine Trip and throttle valve	Modeled in fault tree for seismic failure with separation of variables (SOV) fragility and operator recovery
Emergency Diesel Generator (EDG) Engine Protective Relays	Trips the EDG	Modeled in fault tree for seismic failure with SOV fragility

4.2 Walkdown Approach

This section provides a summary of the methodology and scope of the seismic walkdowns performed for the SPRA [8]. Walkdowns were performed by personnel with appropriate qualifications as defined in the SPID [2] Table 6-5 and the associated requirements in the PRA Standard [4]. The seismic review team (SRT) was comprised of several seismic engineering experts with extensive experience in fragility assessment. Walkdowns of those SSCs included on the seismic equipment list were performed, as part of the development of the SEL, to assess the as-installed condition of these SSCs for use in determining their seismic capacity and performing initial screening, to Identify potential II/I spatial interactions and look for potential seismic-induced fire/flood interactions. The fragilities walkdowns were performed in accordance with the criteria provided in EPRI NP 6041-SL [7] and SQUG guidance [30].

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

The seismic fragility walkdowns were conducted on a mixture of both Unit 1 and Unit 2 equipment. The fragility walkdowns included the evaluation of seismic interactions, including the effects of seismic-induced fires and flooding. The SRT was comprised of several seismic engineering experts with extensive experience in fragility assessment.

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. The fire detection equipment was found to be ruggedly mounted and no concerns with seismically induced inadvertent initiation were identified.

The major concern for seismic-induced fires is from flammable liquids and gases. Thus, the walkdown focused on these sources and their proximity to components on the SEL. Potential fires in the turbine building (and yard areas), hydrogen cylinders outside the MSIV area, transformers in SEL buildings, and lube oil for the diesel generators 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 was reviewed. Particular emphasis was placed upon threaded or jointed piping. Flood sources, including the fire-protection system, the turbine building, large tanks, were evaluated. While most of the SEL components were robust, it was identified that anchorage failure could lead to subsequent flooding scenarios. Identified scenarios that were included in the model are the following:

- Seismic failure of the essential service water (ESW) chillers causing a NSCW flood in the control building;

- Seismic failure of the auxiliary component cooling water (ACCW) heat exchangers causing a NSCW flood in the auxiliary building;
- Seismic failure of the aux cooling units or containment cooling units causing a NSCW flood in the containment.

4.2.1 Significant Walkdown Results and Insights

Consistent with the guidance from NP 6041-SL [7], no significant findings were noted during the VEGP seismic walkdowns.

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 flood scenarios were incorporated into the VEGP 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 VEGP 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 PRA Standard [4].

The peer review assessment, and subsequent disposition of peer review findings, is described in Appendix A, and establishes that the VEGP 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.

4.3.1 Fixed-base Analyses

Since VEGP is a deep soil site, fixed-base analyses were not applicable.

4.3.2 Soil Structure Interaction (SSI) Analyses

VEGP is a deep soil site where all safety-related structures are founded on or embedded in the soil where significant soil-structure-interaction (SSI) effects are expected [16]. The effects of uncertainty in SSI are dominated by the soil response. Uncertainty in the VEGP seismic analysis was accounted for by evaluating the SSI model for three soil columns, best estimate (BE), upper bound (UB), and lower bound (LB) (best estimate, upper bound, and lower bound) in accordance with NRC Standard Review Plan (SRP) Section 3.7.2 [26]. These three soil columns account for the measured variation in site-specific soil properties and account for most of the uncertainty in seismic response of the VEGP structures.

The ground motion input was developed using the site-specific BE, UB, and LB shear wave velocity profiles, and strain-compatible damping ratio profiles. The reference level earthquake ground motion corresponds to the 1E-4 UHRS at plant grade, which has a horizontal PGA of 0.436g. The SASSI2010 analysis code was used to perform the SSI analysis and the depth of soil considered was at least three times the maximum foundation dimension below the foundation (ASCE 4-98 [22]). Cutoff frequency for the SSI analysis was chosen to be 30 Hz, as all input motion spectra show that the energy content decreases above 20 Hz and completely fades at frequencies above 30 Hz. The SSI models were sufficiently refined to transmit frequencies up to 30 Hz through the soil-foundation interface.

Preliminary SSI analyses with the assumption of uncracked section for concrete elements were performed using three soil conditions (BE, UB, LB). In accordance with ASCE 4-98 [22], response spectrum or time history analysis was performed to confirm locations of cracked concrete. If the predicted stresses exceeded ASCE 4-98 limits, then the highly stressed regions were re-analyzed with reduced stiffness to simulate concrete cracking.

The SSI analysis for the surface founded structures utilized the SASSI Direct Method. The SSI analysis for the deeply embedded structures relied on the SASSI Modified Subtraction Method (MSM) or Extended Subtraction Method 2 (ESM). The size of the model did not permit the use of the SASSI direct method for the embedded structures. In addition, sensitivity studies were performed to confirm the accuracy of the MSM and ESM results.

4.3.3 Structure Response Models

This section summarizes the Seismic Structure Response and Soil Structure Interaction Analysis methodology used, discusses significant / limiting seismic structure response and structure fragility results for the SSCs modeled in the SPRA, discusses important assumptions and important sources of uncertainty, and describes any particular fragility-related insights identified.

The seismic structure response analysis considers the impact of seismic events on the response of site structures containing systems and components important to achieving a safe shutdown. VEGP is characterized as a deep soil site based on the site-specific best-estimate 1E-4 UHRS shear wave velocity profile, which does not exceed 3,000 fps for a depth of 1,000 feet below grade elevation, see Figure 4.3-1 [16].

The in-structure response spectra (ISRS) for structures considered in the seismic PRA were developed using time-history analysis. Both horizontal and vertical ISRS were computed from time-history motions at various floors or other important locations. In-structure response was generated by applying five sets of input motions, each set was applied to simulate fault normal conditions and then applied to simulate fault parallel conditions. At representative node locations, various damping acceleration response spectra (ARS) in the three orthogonal directions are calculated for each of the three directions of the input ground motion. Selection of the locations at which the responses were calculated was based on the equipment location within the building. The ARS are calculated at 301 frequency points equally distributed on the logarithmic scale at the range of frequency from 0.1 Hz to 100 Hz. The responses obtained for the three directions of the input ground motion are combined using the square root sum of the squares (SRSS) method as follows:

$$ARS_x = \sqrt{ARS_{xx}^2 + ARS_{yx}^2 + ARS_{zx}^2}$$

$$ARS_y = \sqrt{ARS_{xy}^2 + ARS_{yy}^2 + ARS_{zy}^2}$$

$$ARS_z = \sqrt{ARS_{xz}^2 + ARS_{yz}^2 + ARS_{zz}^2}$$

where ARS(m)(n) are the SASSI ARS results for the response in "n" direction due to earthquake in "m" direction.

Median and 84th percentile seismic demands (ISRS) are computed using the following procedure:

- Step 1: For each soil case and time history set, the ARS from the three earthquake components are combined using the SRSS method as explained above.
- Step 2: The SRSS combined demands for each of the time history sets corresponding to a single soil case are averaged to get the median demands for that particular soil case.
- Step 3: The median demands for the three soil cases are averaged to get the final median seismic demands. The median demands for the three soil cases are enveloped to get the final 84th percentile seismic demands.

ISRS with highly amplified narrow frequency content was clipped for comparison to broad-banded test response spectra; typical of most NPP components. The guidance in EPRI TR-103959 [21] was performed for the peak clipping process.

The seismic models were based on recent NRC guidance (SRP [26]) and industry codes and standards (ASCE 4-98 [22], and ASCE 43-05 [23]). Building models were developed and median centered response analyses including soil-structure-interaction effects were performed to determine seismic response of SSCs for the 1E-4 uniform hazard response spectra input motion.

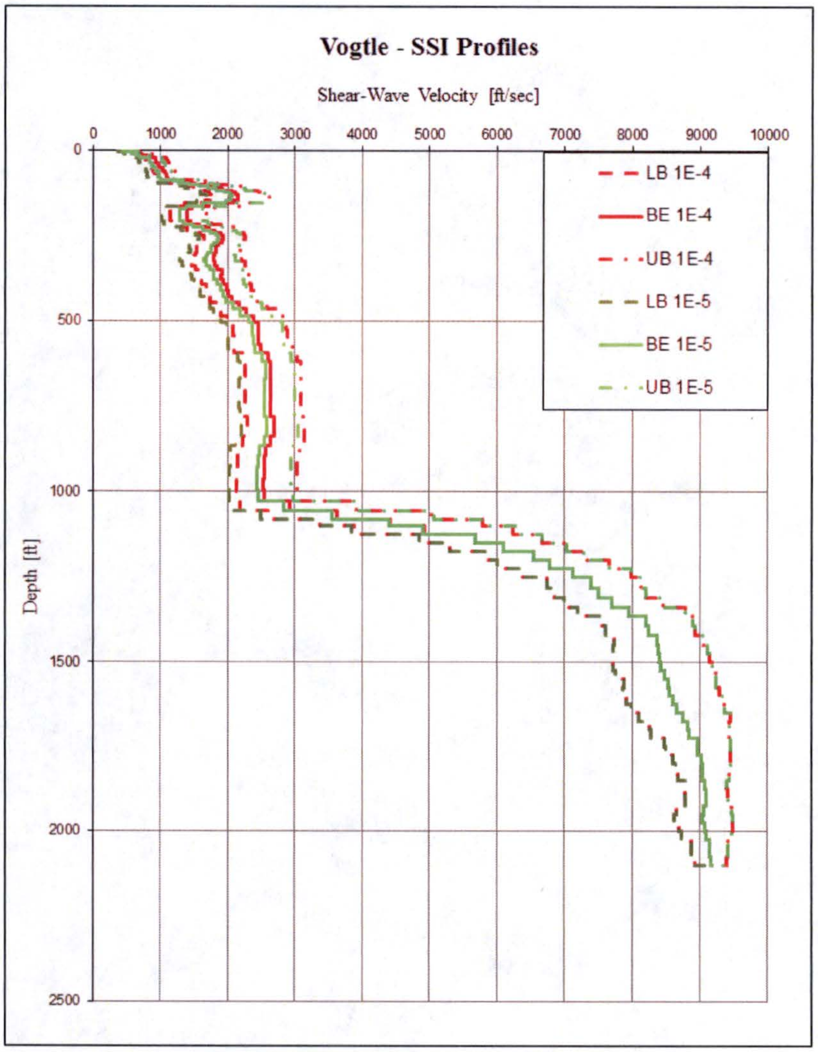

Figure 4.3-1 Vogtle SSI Profile

Simple structures, such as tanks, fuel handling building, and auxiliary feedwater pump house were modeled with lumped-mass stick models (LMSM). More complex structures, such as the Control Building, Auxiliary Building, Diesel Generator Building, Containment Building, and NSCW tower were all modeled with three-dimensional finite element models (FEMs). These detailed building models were sufficiently refined to capture building torsion, out-of-plane floor response, and in-plane floor diaphragm stiffness. Consideration of secondary system masses (point loads and assumed live load distribution) was performed in accordance with ASCE 4-98 [22]. The compressive strength for concrete material was building-specific and ranged from 4,000 psi to 6,000 psi. The yield strength for steel structures was 36,000 psi. FEM analysis model verification was performed by comparing fixed-base fundamental frequencies with those of the design basis LMSM. Static analyses were also performed in which 1g acceleration forces were independently imposed in each of the orthogonal directions and the results of these analyses were reviewed to confirm that the model reasonably represents the fixed-base structure behavior.

Consideration of concrete cracking and structural damping was performed in accordance with ASCE 4-98 [22] and ASCE 43-05 [23]. The effects of concrete cracking and reduced stiffness were addressed by checking for stresses that exceeded code limits and then reducing the stiffness of those elements. In addition, material damping in cracked regions was increased from 4% to 7%.

Reviewing transfer functions is an important task when performing SSI analysis. The results of the SSI analysis were validated by carefully reviewing the behavior of transfer functions in all directions and soil cases. The functions were reviewed to confirm low frequency response (i.e., transfer function values should approach 1.0 at low frequency) and to confirm the reasonableness of amplification with increased building elevation. Soil column modes, predicted by the SASSI analysis, were verified by comparing against soil column frequency relationships in ASCE 4-98 [22].

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 VEGP SPRA

Structure	Foundation Condition	Type of Model	Analysis Method	Comments/Other Information
Containment Building	Soil	FEM	Deterministic SSI	LB, BE, UB cases, 5 sets of time histories (T-H) used
Auxiliary Building	Soil	FEM	Deterministic SSI	LB, BE, UB cases, 5 sets of T-H used
Control Building	Soil	FEM	Deterministic SSI	LB, BE, UB cases, 5 sets of T-H used
Fuel Handling Building	Soil	LMSM	Deterministic SSI	LB, BE, UB cases, 5 sets of T-H used
Diesel Generator Building	Soil	FEM	Deterministic SSI	LB, BE, UB cases, 5 sets of T-H used
Auxiliary Feedwater Pumphouse	Soil	LMSM	Deterministic SSI	LB, BE, UB cases, 5 sets of T-H used
NSCW Tower	Soil	FEM	Deterministic SSI	LB, BE, UB cases, 5 sets of T-H used
Condensate Storage Tanks	Soil	LMSM	Deterministic SSI	LB, BE, UB cases, 5 sets of T-H used
Refueling Water Storage Tank	Soil	LMSM	Deterministic SSI	LB, BE, UB cases, 5 sets of T-H used
Reactor Make-up Water Storage Tank	Soil	LMSM	Deterministic SSI	LB, BE, UB cases, 5 sets of T-H used

4.3.4 Seismic Structure Response Analysis Technical Adequacy

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

The peer review assessment, and subsequent disposition of peer review findings, is described in Appendix A, and establishes that the VEGP 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 VEGP anchors the probability of SSC failures to the horizontal PGA of 0.436g, which corresponds to the 1E-4 UHRS at plant grade. The fragilities of the SSCs that participate in the SPRA accident sequences, i.e., those included on the seismic equipment 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, presents a tabulation of the fragilities (with appropriate parameters i.e., Am, 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 based on the rough estimation (rugged, high, medium, low) of the walkdowns. Many of the components on the walkdown list were screened out from explicit seismic modeling in the quantification based on their rugged seismic capacity. For example, virtually all of the instrumentation was assessed qualitatively to have seismically rugged anchorage, such that no quantitative evaluation was needed. The initial logic model included the seismic failures of the low and medium capacity equipment, including most of the electrical switchgear, cabinets, and panels, which were initially judged to have medium capacity.

As the fragility analyses proceeded, a screening criterion was established based on potential contribution to SCDF. It was determined that a component (or correlated group of components) with median capacity of 2.5g (assuming βc of 0.3) could contribute about 2E-07/yr to SCDF if its failure led directly to core damage, which is conservative for many components. With respect to the final SCDF, the maximum contribution for screening at a seismic capacity of 2.5g is 7%. Since this contribution is significant, the screening value was adjusted until the maximum contribution was 2% of the final SCDF, with the result being a 3g screening level. The equipment with seismic capacities between 2.5g and 3g were evaluated to determine if there was the potential for significant improvement in realistic fragilities through additional refinements in fragility calculations. Equipment in this group that could not refined beyond 3g were included in the model or screened out from the model on a system response basis. Exceptions to this screening were made for components that were known to be important (such as the diesel generators), or potentially had significant SLERF impacts. Another exception was made for the structures housing SEL equipment. These were all included in the seismic logic model, except for the AFW pumphouse. The seismic fragility analysis demonstrated that the AFW pumphouse had very high seismic capacity (> 5g), and could be screened from the logic model.

4.4.2 SSC Fragility Analysis Methodology

Seismic fragility evaluations were performed for VEGP SSCs contributing to core damage and large early release. The SSC fragility analysis was performed in accordance with Section 6.4.3 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 recommends Seismic Fragility Applications Guide Update (EPRI 1019200 [19]), Seismic Fragility Application Guide (EPRI 1002988 [20]), Methodology for Developing Seismic Fragilities (EPRI TR-103959 [21]), and A Methodology for Assessment of Nuclear Plant Seismic Margin (EPRI NP 6041-SL [7]). The VEGP fragility analysis is based on these documents, among other industry codes and standards.

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

- Plant-specific design information.
- Use of conservative generic fragilities (e.g., EPRI proxy methods).
- The hybrid method outlined in the Seismic Fragility Application Guide (EPRI 1002988 [20]) 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 [21]).

Critical failure modes were identified and seismic fragility calculations were performed to estimate 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.

A detailed evaluation was performed for VEGP Units 1 and 2 in order to determine how similar the two units are. The evaluation concluded that the two units are sufficiently identical and Unit 1 results can be applicable to Unit 2.

Structures

The VEGP Seismic Category I structures evaluated are:

- Containment Building
- Auxiliary Building
- Control Building
- Fuel Handling Building
- NSCW Tower and Valve House
- Diesel Generator Building
- Auxiliary Feedwater Pumphouse
- Condensate Storage Tanks
- Refueling Water Storage Tank, and
- Reactor Make-up Water Storage Tank.

Structural demands (member forces and acceleration response) required for fragility analysis were derived from seismic models based on recent NRC guidance (SRP [26]) and industry codes and standards (ASCE 4-98 [22], and ASCE 43-05 [23]). These seismic models were detailed, three-dimensional, finite element models or LMSM (see Table 4.3-1) based on plant-specific information and used the 1E-4 uniform hazard as input motion. As VEGP is a deep soil site, these models accounted for the effects of soil-structure-interaction. The effects of building stability, such as sliding and overturning, were evaluated as well as the potential for differential displacement between buildings. These effects, along with earthquake-induced settlement and liquefaction were evaluated and shown not to be significant factors in the fragility evaluation.

For cylindrical shell structures, such as tanks and containment, two failure modes (tangential shear failure and flexural failure) were evaluated in accordance with EPRI NP 6041-SL [7], EPRI 103959 [21], and ACI 349 [24]. For shear wall structures, three failure modes (diagonal shear cracking, flexure and shear friction) were evaluated in accordance with EPRI NP 6041-SL [7], EPRI 103959 [21] and ACI 349 [24]. Inelastic energy absorption, which accounts for additional capacity due to ductile design detailing, was considered in accordance with EPRI NP 6041-SL R1 [7] and ASCE 43-05 [23].

Components

The VEGP component fragilities were derived using a multi-step approach. The EPRI Proxy Method described in EPRI 1019200 [19] was used to develop and assign fragilities to the components as the first step, which included some conservative simplifying assumptions. The EPRI Proxy Method uses the capacity based on EPRI NP 6041-SL [7] and plant-specific demands. 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 EPRI Proxy Method fragilities were utilized for mechanical and electrical components, the EPRI NP 6041-SL [7] equipment caveats were confirmed to be satisfied.

Realistic component failure modes included anchorage, 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 in-structure response spectra (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 critical relays were performed, and made use of GERS [25, 32, 33, 40, 41] for seismic capacities. It was confirmed that the relay vintage and model numbers were consistent with each GERS equipment class. The GERS capacities used are lower than, or the same as, the capacities of these relays in the high frequency range [31, 42]. VEGP is a deep soil site and due to the associated soil-structure interaction effects the predominant seismic demand occurs in the low frequency range. 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 scaling of the existing safety analysis results, in accordance with SPID [2] guidance.

Correlation

Correlation of components (or common cause failure) was considered in accordance with the ASME/ANS PRA Standard [4]. For the VEGP 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 some cases, detailed model results were used to develop location-specific fragilities. These results were used to refine fragility estimates for similar components located on the same floor.

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 4-kv 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 4-kv switchgear failures are not correlated with the 480-v motor control center failures.

However, there were a few exceptions to this general correlation rule. Because detailed finite element models of the structures were developed, the seismic demand at different nodes of the buildings could be determined. Since the seismic fragility of a component is a function of the component seismic capacity and the seismic demand at the component location, similar components at different locations could have different demands, and thus different fragilities. If the difference between fragilities was small, then the components were correlated using the lower fragility value. However, if there was a significant difference in fragilities, then the higher capacity was used to assign a higher correlated fragility to both components, but the lower capacity component was also assigned a unique seismic capacity that only failed that component. Thus, the lower capacity component could fail by itself, but was guaranteed to fail if the higher capacity component was failed. Detailed individual fragilities were not always calculated for every component, so in some cases this more detailed correlation modeling could not be performed, and the general correlation rule was followed.

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 5.4-4 (for SCDF) and Table 5.5-2 (for SLERF). Detailed (separation of variables, SOV) calculations have been performed for the highest risk significant SSCs, as well as for selected other components.

4.4.4 SSC Fragility Analysis Technical Adequacy

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

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

5.0 Plant Seismic Logic Model

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

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

5.1 Development of the SPRA Plant Seismic Logic Model

The VEGP seismic response model was developed by starting with the VEGP internal events at power PRA model of record as of August 31, 2015, and adapting the model in accordance with guidance in the SPID [2] and PRA Standard [4], including adding seismic fragility-related basic events to the appropriate portions of the internal events PRA, eliminating some parts of the internal events model that do not apply or that were screened-out, and adjusting the internal events PRA model human reliability analysis to account for response during and following a seismic event. The model is developed using the EPRI CAFTA software suite. This model does not credit non-permanently installed FLEX equipment, but does include low leakage reactor coolant pump (RCP) seals. Both random and seismic-induced failures of modeled SSCs are included. The modifications to develop the SCDF fault tree are summarized in Table 5.1-1.

For the VEGP SPRA, the following discussion addresses the methods used to develop the seismic plant response model.

Initiating Events and Accident Sequences

The seismic hazard was modeled using 14 discrete hazard intervals (or bins) based on increasing peak ground acceleration. The seismic hazard bins are as listed in Table 3.1-2. Each bin is treated as a seismic initiator and the SCDF (and SLERF) results are summed over all the bins to obtain the total SCDF (and SLERF). Bin-specific SSC fragilities are used in the accident sequences for each bin.

The SPRA models each seismic event (i.e., each bin) as possibly leading to transients and LOCAs (small, medium, large, and excess LOCA (e.g., reactor pressure vessel failure)), with and without onsite AC power, and with response reflecting impact of the seismic event on mitigating systems.

Table 5.1-1 Summary of Modifications to Internal Events CDF Fault Tree to Create Seismic CDF Fault Tree

Modification
Added seismic-induced large loss of coolant accident (LLOCA), medium loss of coolant accident (MLOCA), small loss of coolant accident (SLOCA), LOSP to their respective internal initiator gate logic; Revised the seismic LOSP gate to include the high capacity seismic initiating events (LLOCA, MLOCA, SLOCA).
Revised SBO initiator FT to include the seismic SBO initiator fault tree; Revised SBO fault tree (FT) to include all failures of 4KV emergency buses rather than only long term bus failures.
Removed recovery of AC power; Removed the ability to use Plant Wilson after seismic event; Removed the ability to crosstie Unit 2 EDGs with Unit 1.
Added direct seismic core damage FT for seismic failures of buildings, excessive LOCAs, and loss of instrumentation and control.
Added logic to incorporate consequential SLOCA given a seismic LOSP with inadvertent safety injection signal.
Added two new seismic SBO sequences to reflect core damage with a 21gpm seal LOCA.
Added seismic initiators that could result in an anticipated transient without trip (ATWT) to the existing ATWT sequences; Revised ATWT logic to fail recovery of ATWT by driving CRDs or manually tripping RX in the case of seismic failures of CRD or RV internals; Added seismic SBO ATWT (assumed core damage with no mitigation).
Added seismic failure of the electrical aux board in main control room (MCR) to existing failures.
Added operator actions to start EDG, close breakers, start equipment for sequencer failure.
Added seismic failure of the NSCW piping to the Control Building ESF chillers causing large flooding on Control Building 260' and propagating to core damage.
Added seismic failure of the ACCW heat exchangers as initiator for a flood event, with small and large flood scenarios, with and without seismic LOCAs, with associated Operator actions.
Added seismic failure of the ACU and CCUs as flood initiators inside containment, with and without seismic LOCA scenarios, with associated Operator actions.
Added seismic failure of the CCUs as a flood initiator inside containment, with and without seismic LOCA scenarios. Operator actions were also added.

Certain structural failures are modeled as leading directly to core damage given the potential for multiple system impacts or distributed system failures. These include seismic failure of:

- Containment
- Auxiliary building
- Control building
- NSCW cooling towers and basins

In addition, the following failures of instrumentation and control were included in the model, and assumed to lead directly to core damage:

- Seismic failure of the main control board
- Seismic failure of 125vdc control power panels

Although not required by the SPID [2], the potential for seismically-induced internal fires and internal floods was evaluated based on walkdown observations and several internal flooding scenarios were developed for inclusion in the SPRA model. These are:

- Seismic failure of the ESF chillers causing a NSCW flood in the control building;
- Seismic failure of the ACCW heat exchangers causing a NSCW flood in the auxiliary building;
- Seismic failure of the aux cooling units or containment cooling units causing a NSCW flood in the containment.

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 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 for all redundant components in the model that met these correlation criteria. For correlated groups where there was a significant difference in fragilities, then the higher capacity was used to assign a higher correlated fragility to both components, but the lower capacity component was also assigned a unique seismic capacity that only failed that component. Thus, the lower capacity component could fail by itself, but was guaranteed to fail if the higher capacity component was failed.

Modeling of Human Actions

Human error probabilities (HEP) for operator actions in the SPRA model are developed using the same methodology as in the internal events PRA. The EPRI Human Reliability Analysis (HRA) Calculator software was used to develop and document the HEPs for the internal events actions and for the limited set of seismic response operator actions. HEPs were then adjusted as a function of seismic magnitude using a performance shaping factor approach consistent with the EPRI seismic HRA methodology [9].

Several seismic specific human actions were also identified in response to seismically induced flooding. The seismic response operator actions are listed in Table 5.1-2.

In the peer reviewed model, several operator actions were found to be risk-significant. However, this was based on conservative fragility estimates in the peer reviewed model. Subsequently, in addressing peer review findings, many of the fragilities were refined to reduce conservatism. These changes in fragility estimation resulted in the seismic operator actions no longer being risk-significant.

Table 5.1-2 Seismic Response Operator Actions

Basic Event	Description	HEP
S-OA-BKR-LOCAL	Failure of Operator action to locally reclose breaker after seismic event	2.2E-03
S-OA-DG-START	Operator fails to start and load DG if sequencer fails	6.2E-03
S-OA-TDAFW-RELAY	Operator fails to reset the TDAFW trip throttle valve after relay chatter due to a seismic event	1.1E-03
S-OA-ISOL-CIV	Operator fails to manually isolate containment isolation valves with loss of I&C	1.6E-02
S-OA-ACCW-1-45	Operator fails to isolate flood and recover NSCW in 45 minutes (1 hr available)	1.7E-01
S-OA-ACCW-1-90	Operator fails to isolate flood and recover NSCW in 90 minutes (1 hr available)	1.0E+00
S-OA-ACCW-2-45	Operator fails to isolate flood and recover NSCW in 45 minutes (2 hr available)	4.6E-02
S-OA-ACCW-2-90	Operator fails to isolate flood and recover NSCW in 90 minutes (2 hr available)	8.5E-02
S-OA-ACCW-4-45	Operator fails to isolate flood and recover NSCW in 45 minutes (4 hr available)	4.1E-02
S-OA-ACCW-4-90	Operator fails to isolate flood and recover NSCW in 90 minutes (4 hr available)	4.1E-02
S-OA-CU-ISOL-15	Operator fails to isolate NSCW to CCU in <15min	1.2E-02

A complete dependency analysis was performed on all human actions (including both seismic-specific actions and actions included in the internal events model on which the SPRA is based) required for a response to a seismic event. The results of this dependency demonstrated that those combinations of actions that were identified as dependent were not risk significant (less than 0.5% contribution to SCDF).

SLERF Model

The additional seismic initiating events, and their associated accident sequences, added to the core damage model were also added to the seismic LERF model. Each new seismic core damage sequence was mapped to the appropriate SLERF groups based on the mapping in the internal events level 2 PRA. Most core damage sequences went to several SLERF groups depending on failures in the Level 2 event trees from the internal events PRA. Some of the new sequences, such as failure of the containment or steam generators, were directly mapped to SLERF. Others, such as a SBO with 21gpm seal LOCA, were mapped based on similar core damage sequence mapping, using the level 2 event trees in the internal events PRA. The modifications to develop the seismic LERF fault tree are summarized in Table 5.1-3.

Table 5.1-3 Modifications to Internal Events LERF Fault Tree to Create Seismic LERF Fault Tree

Modification
Added LERF gate inputs for sequences that were added to SCDF logic model: • structural failures, • I&C failures, • 21gpm seal LOCA after SBO, • SBO with ATWT, • seismic reactor vessel rupture, • Steam Generator failure, • ACCW floods, • ACU and CCU floods

Additional SPRA model and quantification assumptions, including treatment of loss of offsite power and seismic induced reactor coolant system leakage, are listed in Section 5.3.2.

5.2 SPRA Plant Seismic Logic Model Technical Adequacy

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

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

5.3 Seismic Risk Quantification

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

5.3.1 SPRA Quantification Methodology

For the VEGP SPRA, the following approach was used to quantify the seismic plant response model and determine seismic CDF and LERF.

The EPRI FRANX software code was used to discretize the seismic hazard into the 14 seismic initiators, and quantify to produce cutsets and estimate the mean SCDF. The EPRI ACUBE code was then utilized to estimate the CDF/LERF more accurately by calculating the exact probability on the entire set of SCDF/SLERF cutsets. ACUBE does not use the rare events approximation as is utilized in CAFTA's min cut upper bound estimation calculation and so ACUBE provides a more accurate solution. Additional details can be found in the following sections, along with descriptions of sensitivity studies, uncertainty estimations and a more complete description on the insights from top contributors to SCDF/SLERF.

5.3.2 SPRA Model and Quantification Assumptions

The following assumptions are important to the seismic PRA model development and quantification:

1. Offsite power cannot be recovered within the 24-hour mission time.
2. Diesel generators and the 4KV emergency switchgear cannot be shared between units.
3. Plant Wilson has a low seismic capacity, and is not available during a seismic event.
4. The potential impacts of aging on equipment are not included in the SPRA. However, the potential impacts identified in the walkdowns, such as corrosion or concrete cracking, are included when judged to be significant.
5. The potential for a seismic-induced SGTR is assessed to be very low, based on a detailed assessment, and failure of the steam generators with potential LERF is dominated by failure of the steam generator (SG) supports.
6. In a seismic LOSP with failure of the CRDM or RV internals, it was assumed that the control rods would not insert, and an ATWT would occur. The internal events PRA did not question ATWT for LOSP sequences. In a LOSP (0.3g), the control rods would be released immediately, so they may insert before the failure of the higher capacity failures of the CRDM (2.2g) or RV internals (>4g).
7. It was conservatively assumed that the normal reactor trip signals and reactor trip breakers must work, even after a LOSP. In part, this was to account for potentially delayed LOSP.
8. The seismic capacity for a small-small LOCA is evaluated to be very high, based on walkdown observations, and is not included in the baseline SPRA.
9. The seismic capacity for small LOCA was estimated using the fragility for the seismic capacity reactor coolant pump, which is also used for the medium, large and excess LOCA fragility. This is conservative since it essentially over estimates the effect of failure of the pump.
10. For the NSCW flooding scenarios for failure of the ACCW heat exchangers, it was assumed that the NSCW pumps would have to be stopped in order for the operators to close the manual isolation valves next to the ACCW HX's. These are large valves near the expected rupture flange, and are the only valves that can be used to isolate the ACCW HX's from the NSCW pumps.
11. The NSCW flooding scenario for failure of the ESF chillers in the control building conservatively assumes that the flooding causes loss of instrumentation and control, since the flood would propagate to the control building basement, flooding the electrical rooms.
12. Seismic failure of the auxiliary building is conservatively assumed to result in core damage and large early release. The calculated seismic capacity is based on failure of the entire first story of the building. The story failure would fail the containment penetration areas. The penetrations are actually connected to the containment, and the auxiliary building does not have walls around the containment, just sealant where walls touch. Therefore, these penetrations would be failed when

the auxiliary building collapses. Although an interior wall has lower capacity, which could potentially fail the A train RHR pump and A train pipe chase, the failure of the interior wall would not lead to core damage.

13. A detailed functional fragility analysis could not be performed for the TDAFW pump based on the available information, so the detailed anchorage fragility was used for this pump. This fragility was similar to the controlling fragilities for other large pumps, such as the RHR and SI pumps.
14. The two motor-driven AFW pumps (MDAFWP) were assumed to have a correlated failure based on similar equipment in the same location. The turbine-driven AFW pump (TDAFWP) is located in the transverse direction, and has an entirely different driver, with anchorage differences. Therefore the MDAFWP failure was not correlated with the TDAFWP for the baseline CDF analysis.

5.4 SCDF Results

This section presents the base SCDF results, a list of the SSCs that are significant contributors, including risk importance measures, and a discussion of significant sequences/cutsets and their relative SCDF contributions. A discussion of sensitivity studies is provided in Section 5.7.

The VEGP SCDF is 2.8×10^{-6}/yr. Table 5.4-1 presents the 14 most dominant cutsets that each contribute at least 1% or more to seismic core damage frequency. Note that these cutsets have been combined across all the hazard bin intervals. Therefore, they are a summation of all cutsets with the same failures but with different seismic initiators.

The dominant 14 cutsets represent approximately 45% contribution to SCDF. Most of the remaining significant cutsets are variations of the top 14 where all the same sequences are represented with varying failures of components leading to the loss of the same function. This is discussed in the following descriptions of the dominant cutsets. Note that the percentage contributions represent the sum of contributions over all the seismic hazard bins for the particular cutset.

The most dominant cutset, representing about 16% of the SCDF, is correlated seismic failure of all four of the 125 VDC 1E Distribution Panels which leads to the failure of vital instrumentation and control. Given this failure, the operators would not have any indication of RCS level, temperature or pressure, so failure of these panels is assumed to lead to failure of automatic and manual response, leading directly to core damage. A sensitivity study has been performed where the median seismic capacity of these panels has been increased to determine if additional insights are masked due to this modeling assumption. Additional failures, such as the Main Control Board (1ACBD-MCB), represent an additional 5% contribution to overall SCDF.

The next most dominant cutset (#2) involves anticipated transient without trip (ATWT) sequences where offsite power is lost due to seismic failure, the control rods fail to drop due to seismically-induced mechanical displacement issues, and the operator fails to perform emergency boration in time to reduce reactor power. The sequence represents an inability to control reactivity; leading to core damage. This cutset is approximately 7% of SCDF and mostly comes from the higher PGA intervals (%G10 and higher). In these high PGA scenarios, the operator action to borate is modeled as guaranteed to fail. Nine of the top 14 cutsets (3, 5, 6, & 9-14) are ATWT sequences where the CRDM seismically fails following LOSP, with a 3rd failure that includes one of the following:

- Seismic failure of vital inverter (1ACIV-120-AB220-LC-B) – 4.7%
- Seismic failure of diesel generator building fans (1DGFN-FAN) – 2.1%
- Seismic failure of diesel generator lube oil (1DGHE-LUBEOIL) – 1.8%
- Seismic failure of motor drive AFW pumps (1AFPM-MDP) – 1.3%
- Seismic failure of Diesel Generator components – 4.8%

The fourth most dominant cutset includes the seismic support failure of all the RCPs. The failure of all the pumps is assumed to lead to a LOCA beyond mitigation capability (excessive LOCA) and core damage. The contribution from this cutset is approximately 4%.

The seventh and eighth most dominant cutsets involves the seismic failure of the control building ESF Chillers causing an NSCW flood in the control building, and the seismic failure of the main control board, respectively. These failures lead to the failure of vital instrumentation and control. The operators will not have any indication on level, temperature or pressure, so failure of these components is assumed to lead directly to core damage. However, the contribution from these cutsets is only 1.5% and 1.3%, respectively.

Additional cutsets involving smaller LOCAs are not represented in the top 14 cutsets but contribute approximately 15% to SCDF. Examples include a seismically induced LOCA with failure of the following components:

- Seismic failure of Diesel Generator components
- Vital AC Inverters
- Vital DC Buses

Table 5.4-1 Dominant SCDF Cutsets *

#	%	CDF	Input 1	Input 2	Input 3	Input 4	Input 5
1	16.2%	4.65E-07	S_1DCBS-PN-CB180-1E	SEQ_DAMAGE			
2	7.1%	2.02E-07	PLL	S_CRDM-ATWT	S_SEISMIC-LOSP	SEQ_ATWT-GT40-7	OA-OBR-------H
3	4.7%	1.35E-07	PLL	S_1ACIV-120-AB220-LC-B	S_CRDM-ATWT	S_SEISMIC-LOSP	SEQ_ATWT-GT40-13
4	4.0%	1.13E-07	S_SEISMIC_RCP_ALL	SEQ_DAMAGE			
5	2.1%	6.05E-08	PLL	S_1DGFN-FAN	S_CRDM-ATWT	S_SEISMIC-LOSP	SEQ_ATWT-GT40-13
6	1.8%	5.19E-08	S_1DGHE-LUBEOIL	S_CRDM-ATWT	S_SEISMIC-LOSP	SEQ_SEIS-SBO-ATWT	
7	1.5%	4.17E-08	S_CB-CHLR-NSCW-FLOOD	SEQ_DAMAGE			
8	1.3%	3.69E-08	S_1ACBD-MCB	SEQ_DAMAGE			
9	1.3%	3.64E-08	PLL	S_1AFPM-MDP	S_CRDM-ATWT	S_SEISMIC-LOSP	SEQ_ATWT-GT40-13
10	1.0%	2.83E-08	PLL	S_1DGDM-VENT-1-3	S_CRDM-ATWT	S_SEISMIC-LOSP	SEQ_ATWT-GT40-13
11	1.0%	2.83E-08	PLL	S_1DGHE-LUBEOIL	S_CRDM-ATWT	S_SEISMIC-LOSP	SEQ_ATWT-GT40-13
12	1.0%	2.78E-08	S_CRDM-ATWT	S_DG-BLDG	S_SEISMIC-LOSP	SEQ_SEIS-SBO-ATWT	
13	1.0%	2.70E-08	PLL	S_1DG	S_CRDM-ATWT	S_SEISMIC-LOSP	SEQ_ATWT-GT40-13
14	1.0%	2.68E-08	S_1SWFN-NSCW-FANS	S_CRDM-ATWT	S_SEISMIC-LOSP	SEQ_SEIS-SBO-ATWT	

* Frequencies are point estimates that do not reflect quantification refinement using ACUBE; however they are valid for relative sequence evaluation; each cutset also includes a plant availability factor basic event (0.9, not shown) to reflect the fraction of time at power.

Descriptions of each basic event listed in the cutsets are provided in Table 5.4-1a.

Table 5.4-1a Basic Event Description Table

EVENT	DESCRIPTION
OA-OBR-------H	Seismically induced failure of Operator Fails to perform emergency boration
S_1ACBD-MCB	Seismically induced failure of Main Control Board
S_1ACIV-120-AB220-LC-B	Seismically induced failure of Vital AC Inverter 1BD1I12
S_1AFPM-MDP	Seismically induced failure of BOTH AFW MDP
S_1DCBS-PN-CB180-1E	Seismically induced failure of 125 VDC 1E Distribution Panel - CB180
S_1DG	Seismically induced failure of Both Diesel Generators
S_1DGDM-VENT-1-3	Seismically induced failure of DG Vent Damper For Fans 1&3
S_1DGFN-FAN	Seismically induced failure of DG BLDG ESF Supply Fan
S_1DGHE-LUBEOIL	Seismically induced failure of DG Lube Oil HX
S_1SWFN-NSCW-FANS	Seismically induced failure of NSCW Tower Fans
S_CB-CHLR-NSCW-FLOOD	Seismic Failure Of CB ESF Chillers Cause NSCW Flood On CB 260
S_CRDM-ATWT	Seismically induced failure of ATWT Due To CRDM Fail To Drop
S_DG-BLDG	Seismically induced failure of Diesel Buildings
S_SEISMIC_RCP_ALL	Seismic Failure Of All RCPS
S_SEISMIC-LOSP	Seismic Loss Of Offsite Power
SEQ_ATWT-GT40-13	Seismically induced failure Sequence Label
SEQ_ATWT-GT40-7	Seismically induced failure Sequence Label
SEQ_DAMAGE	Seismically induced failure Sequence Label
SEQ_SEIS-SBO-ATWT	Seismically induced failure Sequence Label
PLL	Fraction of time above power level specified in ATWS model

Table 5.4-2 shows the contribution from seismically induced initiators that accounts for 90% of total SCDF. These initiators are described earlier in Section 5.1 and were developed specifically for the SPRA.

Table 5.4-2 Contribution of Seismic Initiators to SCDF

Seismic Initiator	CDF	%
Seismic LOSP (without other seismic initiators)	9.18E-07	33
Seismic Failure of Instrumentation and Control (direct core damage)	5.93E-07	22
Seismically Induced ATWT	4.41E-07	16
Seismically Induced LOCA (all sizes)	3.35E-07	19

Table 5.4-3 presents the percentage of the core damage frequency that derives from each interval in the seismic hazard curve. Also shown are the conditional core damage probability (CCDP) for the bin, the percent of total SCDF, and the cumulative SCDF. As can be seen, the majority of the contribution comes from seismic events with PGA greater than 0.8g. Bin %G12 contributes the most to the overall SCDF with 22%. From 0.8g to 2g, the contribution totals about 85%, where below 0.8g the contribution is only about 15%.

Importance analyses were performed for both SCDF and SLERF, using the ACUBE code. From the ACUBE output, Fussell-Vesely (FV) values were determined for each basic event (BE) in the model. Since seismically induced failures require a unique BE for each seismic interval, the FV values for seismic failures for each interval were summed together for each seismic fragility group.

Table 5.4-4 provides the important SCDF contributors, sorted by FV, for SSCs with SCDF FV ≥ 0.02. Because the VEGP SCDF is very low, it was judged that this is a sufficiently low importance target for consideration of important contributors. This table also indicates the seismic fragility parameters for the significant contributors, including median capacity (Am), uncertainty parameters (βr and βu), failure mode, and method of fragility calculation.

The FV listing shows the top individual contributors to SCDF as seismically induced LOSP, due to the low median seismic capacity assumed for offsite power failure following a seismic event. The fragility for LOSP is a generic value and considered reasonably representative for VEGP.

The next highest contributor is seismically induced correlated failure of all 125 VDC 1E Distribution Panels, discussed earlier as a contributor to the dominant seismic cutsets. Other important contributors are seismically induced failure of the control rod drive mechanism (CRDM) resulting in failure to drop the control rods; seismically induced dislocation failure of the RCPs leading to an excessive LOCA; and seismic failures of RCS piping leading to various sizes of LOCA.

The remaining significant components all have relatively low FV contributions.

Table 5.4-3 Contribution to SCDF by Acceleration Interval

Seismic IE Bin	Frequency	CCDP	SCDF	%	Cumulative SCDF
%G01- (0.1g to <0.15g)	8.20E-04	1.08E-06	8.86E-10	0%	8.86E-10
%G02- (0.15g to <0.3g)	8.23E-04	1.28E-05	1.05E-08	0%	1.14E-08
%G03- (0.3g to <0.4g)	1.52E-04	9.15E-05	1.39E-08	0%	2.53E-08
%G04- (0.4g to <0.5g)	6.71E-05	6.22E-04	4.17E-08	1%	6.70E-08
%G05- (0.5g to <0.6g)	3.18E-05	3.00E-03	9.54E-08	3%	1.62E-07
%G06- (0.6g to <0.7g)	1.61E-05	1.05E-02	1.69E-07	6%	3.31E-07
%G07- (0.7g to <0.8g)	8.64E-06	3.00E-02	2.59E-07	9%	5.90E-07
%G08- (0.8g to <0.9g)	4.79E-06	6.78E-02	3.25E-07	12%	9.15E-07
%G09- (0.9g to <1g)	2.71E-06	1.33E-01	3.60E-07	13%	1.28E-06
%G10- (1g to <1.1g)	1.55E-06	2.31E-01	3.58E-07	13%	1.63E-06
%G11- (1.1g to <1.2g)	9.00E-07	3.52E-01	3.17E-07	11%	1.95E-06
%G12- (1.2g to <1.5g)	1.02E-06	6.01E-01	6.12E-07	22%	2.56E-06
%G13- (1.5g to <2g)	2.34E-07	8.97E-01	2.10E-07	8%	2.77E-06
%G14- (>2g)	1.60E-08	9.59E-01	1.53E-08	1%	2.79E-06

Table 5.4-4 SCDF Importance Measures and Fragility Parameters Ranked by FV

Component/ Fragility Group	Description and Failure Mode	FV	Am (g)	βr	βu	Failure Mode	Fragility Method
S_SEISMIC-LOSP	SEISMIC LOSS OF OFFSITE POWER	0.33	0.30	0.40	0.27	Generic	Generic
S_1DCBS-PN-CB180-1E	SEISMIC FAILURE OF 125 VDC 1E DISTR. PANEL - CB180	0.21	2.07	0.25	0.37	Functional	SOV
S_CRDM-ATWT	SEISMIC FAILURE OF ATWT DUE TO CRDM FAIL TO DROP	0.16	2.42	0.26	0.45	Anchorage	SOV
S_SEISMIC_RCP_ALL	SEISMIC FAILURE OF ALL RCPS	0.07	2.54	0.30	0.33	Anchorage	SOV
S_SEISMIC-LLOCA	SEISMIC INDUCED LLOCA	0.04	2.54	0.30	0.33	Anchorage	SOV (1)
S_SEISMIC-MLOCA	SEISMIC INDUCED MLOCA	0.04	2.54	0.30	0.33	Anchorage	SOV (1)
S_SEISMIC-SLOCA	SEISMIC INDUCED SLOCA	0.04	2.54	0.30	0.33	Anchorage	SOV (1)
S_1ACBD-MCB	SEISMIC FAILURE OF MAIN CONTROL BOARD	0.04	2.86	0.32	0.33	Functional	SOV
SDS-F	RCP SHUTDOWN SEAL FAILS TO ACTIVATE AND SEAL FOR 24 HRS.	0.04	N/A	N/A	N/A	Random	N/A
S_1AFPM-MDP	SEISMIC FAILURE OF BOTH AFW MDP	0.03	1.42	0.32	0.32	Functional	EPRI NP 6041-SL
RCPSL-21GPM	RCP SEAL LEAK 21 GPM/PUMP AFTER 13 MIN. TOTAL LOSS OF SEAL COOLING	0.03	N/A	N/A	N/A	Random	N/A
S_RX-TRIP-BKRS-SEIS	SEISMIC FAILURE OF REACTOR TRIP BREAKERS	0.02	3.00	0.32	0.32	Functional	EPRI NP 6041-SL
S_1FC-CCU-FLD	SEISMIC FAILURE OF MULTIPLE CCU WITH NSCW FLD	0.02	1.92	0.28	0.28	Anchorage	SOV
S_RV-INT-ATWT	SEISMIC INDUCED FAILURE OF RX VESSEL INTERNALS	0.02	3.08	0.31	0.34	Anchorage	SOV
S_1DCBS-SGR-CB180	SEISMIC FAILURE OF 125 VDC SWITCHGEAR CB180	0.02	1.98	0.32	0.32	Functional	EPRI NP 6041-SL

Note 1: Based on seismic capacity of RCP coolant pump supports. All other NSSS equipment that lead to a LOCA had a greater capacity.

5.5 SLERF Results

This section presents the seismic large early release frequency (SLERF) results, a list of the SSCs that are significant contributors, including risk importance measures, and a discussion of significant sequences/cutsets and their relative SLERF contributions.

Seismic LERF is defined consistent with the internal events model. A 12 hour time period after event initiation is assumed to allow for evacuation. This time period is considered to be valid for Vogtle seismic events, particularly due to the very low population density in the area. Other characteristics, such as bypass and scrubbing, are the same for seismic as for internal events. The logic for the internal events LERF model is very straightforward, with sequences from the SCDF model ANDed with the appropriate fault tree that models failures leading to bypass of containment.

The VEGP SLERF is 3.3×10^{-7}/yr. Table 5.5-1 lists the dominant SLERF cutsets aggregated across all the seismic hazard bins. The most dominant SLERF cutset, representing about 8% of the SLERF, is failure of the steam generator piping connections, leading to a LOCA beyond mitigation capability (excessive LOCA) and modeled as a direct bypass due to the assumption of damage to the containment penetrations when the SGs are displaced. The majority of this SLERF contribution comes from the seismic bins %G09 and higher.

Cutsets 2 and 4 are representative of the dominant LERF sequence, and are seismically induced LOCA with seismically induced failure of vital inverters. Cutsets 3, 5, and 6 are similar, but involve failure of the DC switchgear instead of the inverters. This leads to the failure of engineered safety features actuation system (ESFAS) signal to automatically close air-operated containment isolation (CI) valves. The inverter failure also fails the operator's ability to close the air operated valves (AOVs) manually from the control room. These AOVs are designed to fail closed on loss of DC power or instrument air, but both could remain available for some time following the seismic event. Therefore, it is assumed that the AOVs will remain open if they fail to receive the isolation signal. This is a potentially conservative modeling assumption, as there are additional ways to verify those valves should be manually closed by the operator. There is also a backup power source, 480V AC power, which could potentially power the ESFAS panels to allow the signal to be sent. Neither of these compensatory measures were included in the model and a sensitivity has been performed to show that the current model is potentially conservative. Cutsets 8 and 9 are similar in impact to Cutsets 2 and 4, except involving failures of AC panels leading to the failures noted above. There are additional cutsets containing higher capacity components that fail the same functions leading to the same accident sequence not represented in the top contributing cutsets. Various combinations of the failure of the vital AC inverters, DC buses and diesel generator components all contribute in lesser amounts to add up to the 60% contribution.

Cutset 7 (3%) is failure of containment, modeled as tangential shear cracking failure, leading to direct bypass. This is dominated by failures in the bins that represent the highest seismic ground motion.

Cutsets 10 through 12 are similar to the scenario in cutsets 2 through 6, 8 and 9, except that for small LOCA, failures of the EDGs are also required.

The listed cutsets contribute approximately 53% of total SLERF. The remaining cutsets constitute small individual SLERF contributions.

SLERF importance measures were calculated in the same manner as for SCDF, and the results are listed in Table 5.5-2.

The FV listing for SLERF indicates that top contributors to SLERF are seismically induced LOCAs of all sizes. In addition, the seismic failure of the 125 VDC Switchgear, CB180 AC Inverter, 120 VAC Panel, battery charger and battery have a high FV values. These SSCs are modeled as leading to a loss of ESFAS signal to close containment isolation AOVs.

The next highest contributing SSCs are the seismically induced failure of all four Steam Generators and seismically induced failure of the containment at high seismic levels. The failure of either of these SSCs would lead to direct failure of containment (penetrations in the steam generator failure case) and cause a large early release.

Table 5.5-3 presents the percentage of the SLERF that derives from each interval in the seismic hazard curve. Also shown are: the hazard bin conditional large early release probability (CLERP), i.e., the probability that a large early release occurs given that a core damage event occurs; the percent of total SLERF; and the cumulative SLERF. As can be seen, 90% of the contribution comes from seismic events with PGA greater than 1.0g. The relatively low CLERP results even at high seismic magnitudes (e.g., %G12) is an indication of the robust containment capability.

Table 5.5-1 Dominant SLERF Cutsets *

#	%	CDF	Input 1	Input 2	Input 3	Input 4	Input 5	Input 6
1	8.21%	2.68E-08	S_SEISMIC-SG	SEQ_SEIS-LERF-DIR-1				
2	5.71%	1.86E-08	S_1ACIV-120-CB180	S_SEISMIC-LLOCA	SEQ_LERF-08	SEQ_LL-3		
3	5.26%	1.71E-08	S_1DCBS-SGR-CB180	S_SEISMIC-LLOCA	SEQ_LERF-08	SEQ_LL-3		
4	5.26%	1.71E-08	S_1ACIV-120-CB180	S_SEISMIC-MLOCA	SEQ_LERF-08	SEQ_ML-5		
5	4.82%	1.57E-08	S_1DCBS-SGR-CB180	S_SEISMIC-MLOCA	SEQ_LERF-08	SEQ_ML-5		
6	4.51%	1.47E-08	S_1DCBS-SGR-CB180	S_SEISMIC-SLOCA	SEQ_LERF-08	SEQ_SL-9		
7	3.34%	1.09E-08	S_CONTAINMENT	SEQ_SEIS-LERF-DIR-1				
8	2.97%	9.69E-09	S_1ACBS-120PN-CB180	S_SEISMIC-LLOCA	SEQ_LERF-08	SEQ_LL-3		
9	2.65%	8.65E-09	S_1ACBS-120PN-CB180	S_SEISMIC-MLOCA	SEQ_LERF-08	SEQ_ML-5		
10	1.96%	6.38E-09	S_1ACIV-120-CB180	S_1DGHE-LUBEOIL	S_SEISMIC-SLOCA	SEQ_LERF-08	SEQ_SL-8	
11	1.35%	4.42E-09	S_1ACBS-120PN-CB180	S_1DGHE-LUBEOIL	S_SEISMIC-SLOCA	SEQ_LERF-08	SEQ_SL-8	
12	1.25%	4.06E-09	S_1ACIV-120-CB180	S_1DG	S_SEISMIC-SLOCA	SEQ_LERF-08	SEQ_SL-8	
13	1.16%	3.80E-09	S_1ACIV-120-CB180	S_1CCHE-4	S_1FCMO-CCU-6FANS	S_SEISMIC-SLOCA	SEQ_LERF-08	SEQ_SL-2
14	1.15%	3.75E-09	S_1ACIV-120-CB180	S_1SWFN-NSCW-FANS	S_SEISMIC-SLOCA	SEQ_LERF-08	SEQ_SL-8	
15	1.12%	3.65E-09	S_1ACIV-120-AB220-LC-B	S_1ACIV-120-CB180	S_1DGDM-VENT-1-3	S_SEISMIC-SLOCA	SEQ_LERF-08	SEQ_SL-8
16	1.05%	3.42E-09	S_1ACIV-120-CB180	S_DG-BLDG	S_SEISMIC-SLOCA	SEQ_LERF-08	SEQ_SL-8	
17	1.05%	3.42E-09	S_1ACIV-120-CB180	S_1DGPN-ENG	S_SEISMIC-SLOCA	SEQ_LERF-08	SEQ_SL-8	

* Frequencies are point estimates that do not reflect quantification refinement using ACUBE; however they are valid for relative sequence evaluation; each cutset also includes a plant availability factor basic event (not shown) to reflect the fraction of time at power.

Descriptions of each basic event listed in the cutsets are provided in Table 5.5-1a.

Table 5.5-1a Basic Event Description Table

Event	Description
S_SEISMIC-SG	Seismic Failure Of Steam Generators
S_1ACIV-120-CB180	Seismically induced failure of AC Inverter CB180
S_1DCBS-SGR-CB180	Seismically induced failure of 125 VDC Switchgear CB180
S_CONTAINMENT	Seismically induced failure of Containment
S_1ACBS-120PN-CB180	Seismically induced failure of 120 VAC Panel CB 180
S_1ACIV-120-AB220-LC-B	Seismically induced failure of Vital AC Inverter 1BD1I12
S_SEISMIC-LLOCA	Seismic Induced LLOCA
S_SEISMIC-MLOCA	Seismic Induced MLOCA
S_SEISMIC-SLOCA	Seismic Induced SLOCA
S_1DGHE-LUBEOIL	Seismically induced failure of DG Lube Oil HX
S_1DG	Seismically induced failure of Both Diesel Generators
S_1CCHE-4	Seismically induced failure of CCW Heat Exchanger
S_1SWFN-NSCW-FANS	Seismically induced failure of NSCW Tower Fans
S_DG-BLDG	Seismically induced failure of Diesel Buildings
S_1DGPN-ENG	Seismically induced failure of DG Engine Control Panel
S_1FCMO-CCU-6FANS	Seismically induced failure of Containment Fan Cooler Units-6,3,4,1,5,8
S_1DGDM-VENT-1-3	Seismically induced failure of DG Vent Damper For Fans 1&3
SEQ_SEIS-LERF-DIR-1	Direct Bypass Sequence Tag
SEQ_LERF-08	Large Early Release Sequence Tag
SEQ_LL-3	Core Damage Sequence Tag
SEQ_ML-5	Core Damage Sequence Tag
SEQ_SL-9	Core Damage Sequence Tag
SEQ_SL-8	Core Damage Sequence Tag

Table 5.5-2 SLERF Importance Measures Ranked by FV

Final Equipment	Description	FV	Am	βr	βu	Mode	Method
S_SEISMIC-LLOCA	SEISMIC INDUCED LLOCA	0.21	2.54	0.3	0.33	Anchorage	SOV (1)
S_SEISMIC-MLOCA	SEISMIC INDUCED MLOCA	0.21	2.54	0.3	0.33	Anchorage	SOV (1)
S_SEISMIC-SLOCA	SEISMIC INDUCED SLOCA	0.20	2.54	0.3	0.33	Anchorage	SOV (1)
S_1DCBS-SGR-CB180	SEISMIC FAILURE OF 125 VDC SWITCHGEAR CB180	0.19	1.98	0.32	0.32	Functional	EPRI NP 6041-SL
S_1ACIV-120-CB180	SEISMIC FAILURE OF AC INVERTER CB180	0.19	1.98	0.32	0.32	Functional	EPRI NP 6041-SL
S_SEISMIC-SG	SEISMIC FAILURE OF SG	0.14	2.75	0.24	0.26	Anchorage	CDFM
S_1ACBS-120PN-CB180	SEISMIC FAILURE OF 120 VAC PANEL CB 180	0.12	2.16	0.31	0.31	Functional	SOV
S_CONTAINMENT	SEISMIC FAILURE OF CONTAINMENT	0.08	2.90	0.23	0.25	Structural	CDFM
S_1DCBC-CB180	SEISMIC FAILURE OF BATTERY CHARGER CB180	0.02	1.98	0.32	0.32	Functional	EPRI NP 6041-SL
S_1DCBY-CB180	SEISMIC FAILURE OF 125 VDC BATTERY CB180	0.02	2.26	0.25	0.35	Functional	SOV

Notes:

1: Based on seismic capacity of RCP coolant pump supports. All other NSSS equipment that lead to a LOCA had a greater capacity.

Table 5.5-3 Contribution to SLERF by Acceleration Interval

Seismic IE Bin	Frequency	CLERP	SLERF	%	Cumulative SLERF
%G01- (0.1g to <0.15g)	8.20E-04	Note 1	Note 1	0%	Note 1
%G02- (0.15g to <0.3g)	8.23E-04	Note 1	Note 1	0%	Note 1
%G03- (0.3g to <0.4g)	1.52E-04	3.15E-08	4.79E-12	<1%	4.79E-12
%G04- (0.4g to <0.5g)	6.71E-05	7.31E-07	4.90E-11	<1%	5.38E-11
%G05- (0.5g to <0.6g)	3.18E-05	9.00E-06	2.86E-10	<1%	3.40E-10
%G06- (0.6g to <0.7g)	1.61E-05	7.71E-05	1.24E-09	<1%	1.58E-09
%G07- (0.7g to <0.8g)	8.64E-06	4.60E-04	3.98E-09	1%	5.56E-09
%G08- (0.8g to <0.9g)	4.79E-06	1.96E-03	9.40E-09	3%	1.50E-08
%G09- (0.9g to <1g)	2.71E-06	6.45E-03	1.75E-08	5%	3.25E-08
%G10- (1g to <1.1g)	1.55E-06	1.72E-02	2.67E-08	8%	5.92E-08
%G11- (1.1g to <1.2g)	9.00E-07	3.79E-02	3.41E-08	10%	9.33E-08
%G12- (1.2g to <1.5g)	1.02E-06	1.18E-01	1.20E-07	37%	2.13E-07
%G13- (1.5g to <2g)	2.34E-07	4.27E-01	9.99E-08	31%	3.13E-07
%G14- (>2g)	1.60E-08	7.66E-01	1.23E-08	4%	3.25E-07

Note 1: Contribution is insignificant in this interval.

5.6 SPRA Quantification Uncertainty Analysis

Parameter uncertainty in seismic PRA results comes from seismic hazard curve uncertainty, the SSC fragility uncertainties, and uncertainties in the human interaction and random failure calculations. SPRA model parameter uncertainty was quantified using the EPRI UNCERT code. The results are provided in Table 5.6-1, and Figures 5.6-1 and 5.6-2, each of which shows the curves of cumulative probability and probability density function.

Table 5.6-1 Parameter Uncertainty Analysis Results

	Point Estimate Mean	Mean	5%	Median	95%	Standard Deviation	Skewness
Seismic CDF Mean	2.79E-06	3.57E-06	3.34E-07	1.92E-06	1.18E-05	5.69E-06	7.12
Seismic LERF Mean	3.25E-07	4.29E-07	3.17E-08	2.13E-07	1.50E-06	6.90E-07	5.97

The UNCERT runs were performed using the Monte Carlo method of sampling and a total of 20,000 samples. Both SCDF and SLERF runs solved 20,000 cutsets using ACUBE. The distribution for both SCDF and SLERF appears generally uniform. The uncertainty is generally dominated by the hazard uncertainty. Since much of the seismic risk comes from higher seismic intervals (greater ground motion), the failure probabilities at this ground motion are generally very high and therefore will not contribute much in the way of uncertainty. The point estimate mean is calculated for each acceleration interval using mean values for the seismic hazard frequency, mean values for the seismic fragilities, and mean values for the random failures and human error probabilities. These acceleration interval point estimate means are then summed for the total SCDF and SLERF point estimate means. Comparison with the point estimate values indicates that the point estimates provide a reasonable approximation of the mean values.

Model uncertainty is introduced when assumptions are made in 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 VEGP SPRA, the important model uncertainties are addressed through the sensitivity studies described in Section 5.7 to determine the potential impact on SCDF or SLERF.

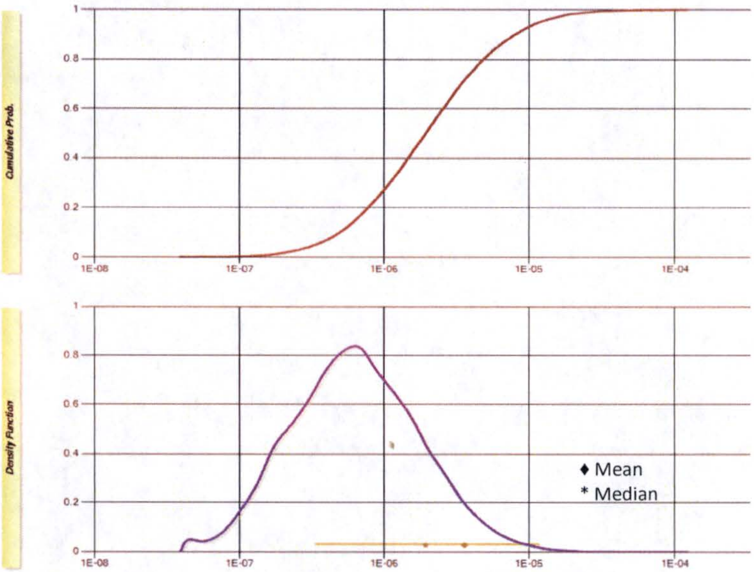

Figure 5.6-1 SCDF – 20,000 Sample Monte Carlo

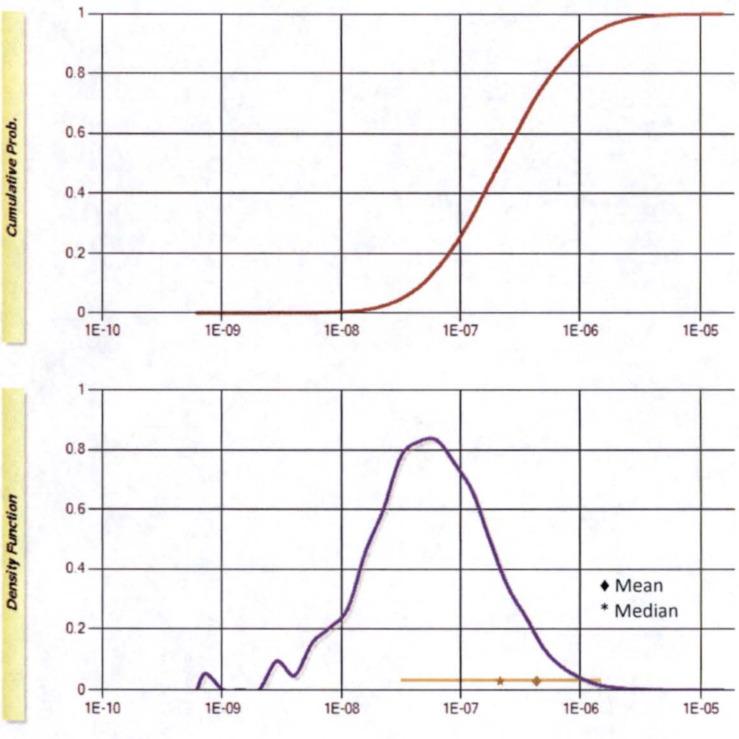

Figure 5.6-2 SLERF – 20,000 Sample Monte Carlo

Completeness uncertainty relates to potential risk contributors that are not in the model. The scope of the VEGP 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. Note that any significant degradation identified during the plant walkdown was included in the fragility calculations. Other potential issues include impacts of plant organizational performance on risk, and unknown omitted phenomena and failure mechanisms. By their very nature, the impacts on risk of these types of uncertainties are not known.

5.7 SPRA Quantification Sensitivity Analysis

Several sensitivity studies were performed to examine different input information and assumptions on the VEGP SPRA results. Among the sensitivity studies examined were the following:

- Model Truncation and Convergence
- Small-Small LOCA
- Control Rod Insertion
- Offsite Power Impact with Improved Plant Wilson fragility
- Human Reliability
- Increased Seismic Capacity of all Four 125 VDC 1E Distribution Panels
- Preventing ESFAS Signal Failure on Loss of Vital AC Inverters
- Auxiliary Building Failure

Table 5.7-1 provides a summary of the sensitivity studies. It also describes the sensitivity and lists any change in median seismic capacity (if applicable). The results for the base case and the sensitivity are shown for both SCDF and SLERF. A percent difference is shown to illustrate more important cases.

5.7.1 Model Truncation and Convergence

The baseline SPRA was quantified at 1E-11 for seismic initiators %G01-%G11 and 5E-10 for %G12-%G14. The reason for the two different truncation levels is that the higher bins have a CCDP of approximately 1.0 and the quantification time drastically increases below 5E-10 for no added benefit. The SLERF model is similar where %G01-%G11 was quantified with a truncation of 1E-12 and %G12-%G14 was set at 5E-10. Model convergence per the criteria in the PRA Standard was achieved at these levels.

5.7.2 Small-Small LOCA

As discussed in the PRA Standard [4] and the EPRI SPRA Implementation Guide [10], the SPRA must consider the potential occurrence of a small-small LOCA. For VEGP the seismic capacity walkdowns evaluated small piping and tubing inside the containment, with the conclusion that the fragility for a small-small LOCA was about the same as for a SLOCA, which is included in the SPRA. A sensitivity study with conservative assumptions was performed by reducing the SLOCA median capacity to 1.02g, which corresponds to assuming that the Small-Small LOCA high confidence of low probability of failure (HCLPF) is about two times the design basis earthquake (DBE) of 0.2g. The impact on CDF was an increase of more than 105% and LERF increased more than 229%.

This sensitivity study confirms the model's reliance on the Small-Small LOCA modeling assumptions. However, there are two factors that over-emphasize this apparent sensitivity. First, the HCLPF value used in this sensitivity is much lower than anticipated based on the SPRA walkdowns. Second, the model includes a conservative assumption that failure of any of the Class 1E Inverters will lead to the failure of ESFAS to send a safety injection (SI) signal, thus failing to auto-start AFW in CDF sequences. In LERF sequences, failure of the inverters will also lead to the failure to send a CI signal and several AOVs will remain open leading to a large early release. During a small-small SLOCA, the time to available to restart AFW is more than 24 hours and credit for an operator action would substantially reduce the small small LOCA contribution to SCDF and SLERF. These conservative assumptions directly affect all SLOCA sequences where ESFAS is relied upon for response to prevent core damage and subsequently a large early release. Since the VEGP SPRA SCDF and SLERF results are already relatively low, no further model refinement was deemed necessary.

5.7.3 Control Rod Insertion

The model assumes that given a seismic LOSP with failure of the CRDM or RV internals, the control rods would not insert due to mechanical interference, and an ATWT would occur. (The internal events PRA does not question ATWT for LOSP sequences, since the rods would drop on loss of power to the gripper coils and mechanical binding is unlikely). In a LOSP (0.3g), the control rods would be released immediately, so they may insert before the failure of the higher capacity CRDM (2.2g) or RV internals (>4g). A sensitivity study provides an assessment by assuming that only 10% of the seismic failures would result in failure of the control rods to insert. SLERF was not affected, but SCDF was reduced by about 6%. Given the relatively low VEGP SPRA SCDF, no further refinement to the model was made.

5.7.4 Offsite Power Impact with Improved Plant Wilson Fragility

Allen B. Wilson Combustion Turbine Plant (Plant Wilson) is owned by Southern Company and is located in Waynesboro, Georgia. Plant Wilson provides a backup source of offsite power to VEGP. Plant Wilson has several black-start combustion turbine generators, and a dedicated underground line to a small transformer in the Vogtle Units 1 through 4 switchyard. During the walkdown of Plant Wilson, several seismic vulnerabilities were identified, such as unanchored starting batteries. If these vulnerabilities were modified, the seismic capacity of offsite power could be improved. For study purposes, the seismic LOSP capacity in the VEGP SPRA was increased to 0.5g median. The results showed that LERF was not significantly impacted, but SCDF was reduced by about 2%. This adjusted median capacity would mostly benefit the lower seismic hazard scenarios, and the overall result would not change much since these scenarios do not contribute much to SCDF. Based on this result, no plant changes are warranted.

5.7.5 Human Reliability

To examine the uncertainty inherent in the calculation of operator action human error probabilities (HEPs), three sensitivity studies were performed. For the first case, all of the internal event PRA HEPs were increased by a factor of 3, with a cap of 1.0 (failure). SCDF increased by 9% and SLERF increased by 1%. The risk insights from the sensitivity, however, did not change. The action modeling operators failing to borate and start feed and bleed became slightly more important, consistent with the increase in failure probability of those events.

For the second case, all of the seismic-specific operator action HEPs were increased by a factor of 3, with a cap of 1.0 (failure). Neither SCDF nor SLERF was significantly impacted by this increase. This is expected because the major contribution to SCDF and SLERF come from high PGA scenarios where the HEPs are already set to 1.0 (guaranteed failure).

The third case assumed no credit for operator action in all bins above 0.8g. This again did not impact SCDF or SLERF because most of these actions were already set to (or close to) 1.0 in the base model quantification.

The results indicate that the model is not overly sensitive to operator response credit.

5.7.6 Increased Seismic Capacity of all Four 125 VDC 1E Distribution Panels

A sensitivity study was performed to determine the potential reduction in SCDF from the top contributing SSC (correlated failure of the four 125 VDC 1E Distribution Panels). In this case the panel Am value was set to 5g (i.e., assume very high seismic capacity could be achieved). A reduction of about 22% was realized in SCDF; however, the SLERF reduction was only about 1%. The SCDF reduction is related to the conservative modeling of failure of ESFAS due to failure of the panels. The smaller impact to SLERF is because there are also other failures that can fail the containment isolation signal to close AOVs preventing a release. The next section provides a related sensitivity case.

Given the relatively low VEGP SPRA SCDF, no plant changes are warranted.

5.7.7 Preventing ESFAS Signal Failure on Loss of Vital AC Inverters

A potentially conservative portion of the SLERF model fails the ESFAS signal on the failure of Vital AC inverters. This sensitivity evaluates the impact of the conservatism. The dominant sequences in which this scenario arises is following a seismically-induced LOCA (any size) where AFW also fails to start and the containment isolation AOVs (HV-0780, HV-0781) do not close because DC power and instrument air for the valve operators is still available for some time following the event. The model requires the ESFAS signal to close the valves and no operator action is credited for manually closing the valves following a LOCA. Another potential recovery is restoring power to the panels from 480V AC to allow the ESFAS signal to be sent.

Table 5.7-1 Summary of Sensitivity Study Results

Baseline Values							
2.8E-06	**SCDF [1]**						
3.3E-07	**SLERF [1]**						
Sensitivity Item	**Description**	**Am Original**	**Am Sensitivity**	**SCDF [1]**	**% Change**	**SLERF [1]**	**% Change**
Small-Small LOCA Impact	Reduce Seismic Capacity of SLOCA	2.54	1.02	5.7E-06	105%	1.1E-06	229%
CRD Insertion	Decrease the frequency of failure of rods to 10% of challenges	-	-	2.6E-06	-6%	3.2E-07	< -1%
Internal Events PRA HEPs Increase	Multiply all FPIE HEPs by 3	-	-	3.0E-06	9%	3.3E-07	1%
SPRA HEPs Increase	Multiply all seismic specific HEPs by 3	-	-	2.8E-06	NC	3.3E-07	NC
HEPs above 0.8g	Increase all HEPs above 0.8g to 1.0	-	-	2.8E-06	NC	3.3E-07	NC
125 VDC 1E Distribution Panels	1DCBS-PN-CB180-1E median fragility increased to 5.0g	2.07	5	2.2E-06	-22%	3.2E-07	-1%
Plant Wilson Credit	Added Plant Wilson Credit to the Model	0.3	0.3	2.8E-06	-1%	3.2E-07	-1%
Aux Building Failure	Aux Building Leads Direct to Core Damage	4.6	2.8	2.9E-06	4%	3.3E-07	NC

Notes: 1. ACUBE reported value
NC = No change

The sensitivity was performed by preventing the failure of the AOVs to close for seismic events. This reduced LERF by a factor of about 3, to $1x10^{-7}$/yr.

5.7.8 Auxiliary Building Failure

The Auxiliary Building (AB) has walls that could fail seismically and damage equipment. Since a detailed study was not performed to determine the specific impacts of the wall failures, this sensitivity study replaced a direct-to-core-damage event with a surrogate (non-SCDF end state) for the failure of the AB walls. The median capacity was set at 2.8g. The results indicate that SCDF would be increased by about 4% and the impact to SLERF was insignificant. This demonstrates that the current modeling technique is adequate for the treatment of the AB failures.

5.8 SPRA Logic Model and Quantification Technical Adequacy

The VEGP SPRA risk quantification and results interpretation methodology 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 VEGP SPRA seismic plant response analysis is suitable for this SPRA application.

6.0 Conclusions

A seismic PRA has been performed for Vogtle Electric Generating Plant Units 1 and 2 in accordance with the guidance in the SPID [2]. The SPRA shows that the point estimate seismic CDF is 2.8×10^{-6}/yr and the seismic LERF is 3.3×10^{-7}/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 Vogtle Electric Generating Plant Units 1 and 2 as of the SPRA freeze date, August 31, 2015. 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.

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8.0 Acronyms

AB	Auxiliary Building
ACCW	Auxiliary Component Cooling Water
ACI	American Concrete Institute
ACU	Auxiliary Cooling Unit
AFW	Auxiliary Feedwater System
Am	Median Seismic Capacity
ANS	American Nuclear Society
AOV	Air Operated Valve
ARS	Acceleration Response Spectra
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ATWT	Anticipated Transient Without Trip
BBM	Blue Bluff Marl
CCDP	Conditional Core Damage Probability
CCU	Containment Cooling Unit
CDFM	Conservative Deterministic Failure Model
CEUS	Central and Eastern United States
CI	Containment Isolation
CRDM	Control Rod Drive Mechanism
DBE	Design Basis Earthquake
EDG	Emergency Diesel Generator
EPRI	Electric Power Research Institute
ESEP	Expedited Seismic Evaluation Program
ESF	Engineered Safety Features
ESFAS	Engineered Safety Features Actuation System
ESM	Extended Subtraction Method
ESP	Early Site Permit
ESW	Essential Service Water
FEM	Finite Element Model
FIRS	Foundation Input Response Spectra
FT	Fault Tree
FV	Fussell-Vesely (risk importance measure)
GERS	Generic Equipment Ruggedness Spectra
GMPE	Ground Motion Prediction Equation
GMRS	Ground Motion Response Spectra
HCLPF	High Confidence of Low Probability of Failure

HEP	Human Error Probability
HF	High Frequency
HRA	Human Reliability Analysis
HVAC	Heating Ventilation and Air Conditioning
IE	Initiating Event
IPEEE	Individual Plant Examination for External Events
ISRS	In-Structure Response Spectra
LF	Low Frequency
LLOCA	Large Loss of Coolant Accident
LMSM	Lumped Mass Stick Model
LOCA	Loss of Coolant Accident
LOSP	Loss of Offsite Power
MDAFWP	Motor Drive Auxiliary Feedwater Pump
MFFF	MOX Fuel Fabrication Facility
MLOCA	Medium Loss of Coolant Accident
MSL	Mean Sea Level
MSM	Modified Subtraction Method
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NSCW	Nuclear Service Cooling Water
NSSS	Nuclear Steam Supply System
NTTF	Near Term Task Force
PGA	Peak Ground Acceleration
PSHA	Probabilistic Seismic Hazard Analysis
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RLME	Repeated Large Magnitude Earthquake
RPS	Reactor Protection System
RV	Reactor Vessel
SA	Spectral Acceleration
SBO	Station Blackout
SCDF	Seismic Core Damage Frequency
SEL	Seismic Equipment List
SFP	Spent Fuel Pool
SFR	Seismic Fragility Element Within ASME/ANS PRA Standard
SG	Steam Generator
SHA	Seismic Hazard Analysis Element Within ASME/ANS PRA Standard

SHS	Seismic Hazard Submittal
SI	Safety Injection
SLERF	Seismic Large Early Release Frequency
SLOCA	Small Loss of Coolant Accident
SMA	Seismic Margin Assessment
SOV	Separation of Variables
SPID	Screening, Prioritization and Implementation Details
SPR	Seismic PRA Modeling Element Within ASME/ANS PRA Standard
SPRA	Seismic Probabilistic Risk Assessment
SQUG	Seismic Qualification Utility Group
SRP	Standard Review Plan
SRSS	Square Root of Sum of Squares
SRT	Seismic Review Team
SSC	Structure, System or Component
SSEL	Safe Shutdown Equipment List
SSI	Soil Structure Interaction
TDAFWP	Turbine Driven Auxiliary Feedwater Pump
TSCR	Truncated Soil Column Response
UHRS	Uniform Hazard Response Spectra
UHS	Ultimate Heat Sink
USI	Unresolved Safety Issue
VEGP	Vogtle Electric Generating Plant
VEGP 1 and 2	Vogtle Electric Generating Plant Units 1 and 2
VEGP 3 and 4	Vogtle Electric Generating Plant Units 3 and 4
WUS	Western United States

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 has two purposes:

1. Provide a summary of the SPRA peer review
2. Provide the bases for the technical adequacy of the SPRA for the 50.54(f) response.

The VEGP 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].

The information presented here establishes that the SPRA has been peer reviewed by a team with adequate credentials to perform the assessment, establishes that the peer review process followed meets the intent of the peer review characteristics and attributes in Table 16 of RG1.200 R2 [11] 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 assessment [6], and subsequent disposition of peer review findings, is summarized here. The scope of the review encompassed the set of technical elements and supporting requirements (SR) for the SHA (seismic hazard), SFR (seismic fragilities), and SPR (seismic PRA modeling) elements for seismic CDF and LERF. The peer review therefore addressed the full set of SRs identified in Tables 6-4 through 6-6 of the SPID [2].

The VEGP SPRA peer review was conducted during the week of November 17, 2014. As part of the peer review, a walk-down of portions of VEGP Units 1 & 2 was performed on November 17, 2014 by members of the peer review team who have the appropriate SQUG training.

A.2. Summary of the Peer Review Process

The peer review was performed against the requirements in Part 5 (Seismic) of Addenda B of the 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 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 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 members of the peer review team were Mr. Kenneth Kiper and Dr. Andrea Maioli of Westinghouse, Dr. Martin McCann of Jack R. Benjamin & Associates, Dr. Richard Quittmeyer of Rizzo Associates, Mr. Steve Eder of Facility Risk Consultants, Mr. William Horstman of Pacific Gas & Electric Company, and Mr. Aaron Quaderer of FirstEnergy Nuclear Operating Company. The peer review team members met the peer review independence criteria in NEI 12-13 [5] and had no involvement in the development of the Vogtle Units 1 &2 SPRA.

Mr. Kiper, the team lead, is a Technical Manager at Westinghouse after a 31-year career at Seabrook Station. He has experience in virtually every aspect of PRA modeling and applications, including upgrading and maintaining the Seabrook seismic PRA.

Dr. Martin McCann was the lead for the Seismic Hazard Analysis (SHA) technical element. He has 30-years' experience in engineering seismology including site response analysis and specification of ground motion. He was assisted in the hazard review by Dr. Richard Quittmeyer, an internationally-recognized expert in seismicity, seismic hazard, and site characterization.

Mr. Steve Eder was the lead for the Seismic Fragility Analysis (SFR) technical element. Mr. Eder has more than 30-years' experience in the fields of natural hazards risk assessment, seismic fragility analysis, structural performance evaluation, and retrofit design. He was assisted by William Horstman, a senior consulting civil engineer for Diablo Canyon. Mr. Horstman has more than 30-years' experience in the fields of structural engineering and structural mechanics.

Dr. Andrea Maioli was the lead for the Seismic Plant-Response Analysis (SPR) technical element. Dr. Maioli has over 10-years' experience in the nuclear safety area generally and seismic PRA specifically. Mr. Aaron Quaderer assisted in the review of the Seismic Plant Response technical elements. He has over ten years of experience in plant and design engineering, including responsibility for maintenance and application of the Davis Besse PRA model.

A.4. Summary of the Peer Review Conclusions

The review team's assessment of the SPRA elements is excerpted from the peer review report [6] 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.

SHA

- As required by the Standard, the frequency of occurrence of earthquake ground motions at the site was based on a site-specific probabilistic seismic hazard analysis (PSHA). The Senior Seismic Hazard Analysis Committee (SSHAC) process of conducting a PSHA was used to develop the regional seismic source characterization (SSC) model and the ground motion model (GMM) inputs to the analysis. The SSC inputs to the PSHA are based on the recently completed Central and Eastern U.S. (CEUS) seismic source model. The ground motion model inputs to the PSHA are based on the CEUS ground motion update project. The requirements of the SSHAC process satisfy the requirements of the standard. The SSHAC process defines a method for utilizing structured expert elicitation and minimum technical requirements to complete a PSHA.
- The "SSHAC level" of a seismic hazard study 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 level of study is not mandated in the standard; however, both the SSC and the GMM parts of the PSHA were developed as a result of SSHAC level 3

analyses. In the case of the GMM, 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.

- As a first step to performing a PSHA, the Standard requires an up-to-date database, including regional geological, seismological, geophysical data, and local site topography, and a compilation of surficial geologic and geotechnical site properties. These data include a catalog of relevant historical, instrumental, and paleo-seismic information within 320-km of the site. The CEUS SSC study involved an extensive data collection effort that satisfies the requirements of the standard as it relates to developing a regional-scale seismic source model.
- No effort was made to compile new (relative to the data used in the CEUS SSC study) or local (relative to the regional scale) information beyond what was considered in the development of the CEUS SSC regional scale seismic sources. This includes:
  - Updating the earthquake catalog and evaluating the potential impact on the estimate of seismicity parameters, and
  - Collecting and evaluating geologic, seismologic and geophysical information to assess whether new information or information at a local scale exists that would indicate that new, local seismic sources or modifications to the CEUS regional scale seismic sources are required.
- In the implementation of the CEUS SSC model for the Vogtle site, all distributed seismic sources in the CEUS SSC within 640 km and all RLME sources within 1000 km were included in the PSHA calculations. By including these seismic sources in the analysis, the contribution of "near-field" and "far-field" earthquake sources to ground motions at Vogtle were considered.
- The seismic hazard analysis for the Vogtle site also took into account the effects of local site response. However, the review team determined that the site response analysis did not fully evaluate and model aleatory and epistemic uncertainties in the site response analysis.
- 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" source spectral shapes.
- The standard requires that sensitivity calculations be performed to document the models and parameters that are the primary contributors to the site hazard. In the Vogtle PSHA, only a limited number of sensitivity calculations and deaggregation results were presented. As a result, this requirement was not met.
- The standard requires that a screening analysis be performed to determine whether hazards other than earthquake ground motion pose a hazard to the site. A screening analysis was performed for some but not all of the other seismic hazards. Therefore, this assessment was incomplete.
- 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 Vogtle site.

- The standard requires that documentation of the PSHA be provided that supports the PRA applications, peer review and upgrades. 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 assess that the approach is appropriate, the analysis was performed correctly, and the results are reasonable. The Vogtle PSHA documentation is minimal and therefore this requirement is not met. The findings with regard to this requirement point out that the way the CEUS SSC and GMM models were implemented in the PSHA was not described; the most recent documentation is incomplete in the sense that it does not include all of the analyses (i.e., results of sensitivity calculations) performed in earlier versions.

SFR

- The Standard requires that all the structures, systems and components (SSCs) that play a role in the seismic PRA be identified as candidates for subsequent seismic fragility evaluation. A seismic equipment list was developed. Generic seismically insensitive items as well as seismically rugged items identified by the walkdown team were screened out. A screening level of 2.5g median capacity was then established for fragility evaluation based on insignificant contribution to risk.
- Generic data and conservative simplifying assumptions were utilized to establish preliminary seismic fragilities for all SSCs, with the goal of demonstrating capacity above this satisfy level. The separation-of-variables methodology was employed. The fragility evaluations were refined on an as-needed basis as requested by the quantification team. Refinements were incorporated until favorable quantification results were achieved. As a result of this process, the seismic fragilities as a whole are not realistic as required by the standard.
- Excess conservatism was noted in essentially all fragility calculations. This approach was possible for Vogtle due to the high seismic capacity of the SSCs.
- The Standard requires that the seismic-fragility evaluation be based on realistic seismic response that the SSCs experience at their failure levels. The building response spectra were developed using new 3-D dynamic building models and soil-structure-interaction analyses, and used in the evaluation of seismic fragilities. A deterministic method was employed to establish median-centered response corresponding to structural model properties associated with the 1E-4 uniform hazard response spectrum shaking level. As a result, seismic response is overestimated for higher levels of shaking. Higher shaking levels cause additional cracking, leading to softening of members and increased structural damping. Strain dependent soil properties also change. A structural response modification factor was not used in the fragility evaluations to adjust for this conservatism.
- A series of walkdowns, focusing on the anchorage, lateral seismic support, functional characteristics, and potential systems interactions were conducted and documented appropriately in support of the fragility analysis. The walkdowns also identified the potential for seismic-induced fires and floods. The walkdown observations were subsequently incorporated in the seismic fragility evaluations. Some improvements in documentation were recommended.

- The SPRA identifies the relevant failure modes for the SSCs through a review of plant design documents, earthquake experience data, and walkdowns. Seismic-fragility evaluations were performed for these critical failure modes. The review team found that in general this requirement was satisfied, but noted, that the failure modes associated with sloshing of water and soil settlement were overlooked.
- The Standard requires that the seismic-fragility parameters be based on plant-specific data supplemented as needed by earthquake experience data, fragility test data, and generic qualification test data. The review team found that this requirement was satisfied. Use of generic data was justified via the iterative process of refining fragility on an as-needed basis.

SPR

- The seismic PRA model was developed by modifying the Full Power Internal Events (FPIE) PRA model to incorporate specific aspects of seismic analysis that are different from the FPIE. The logic model appropriately includes seismic-caused initiating events and other failures including seismic-induced SSC failures, non-seismic-induced unreliability and unavailability failure modes (based on the FPIE model), and human errors.
- Additional documentation was needed regarding the details of the modification performed on the internal events PRA model to generate the seismic model. For example, some sequences have been added (apparently correctly) but without explaining the process. The reviewers felt that additional documentation would be needed to ensure future re-generation and update of the model.
- A systematic identification of all the potential seismic induced initiators is presented and, while a hierarchy tree was not generated, the documentation supports the conclusion that the complete spectrum of seismic-induced scenarios was modeled.
- The "Surry method" was applied to the Vogtle SPRA for the modification of existing human actions. Some actions were added to the logic specifically for the seismic logic (e.g., seismic-induced flooding specific actions). The reviewers questioned whether the treatment of seismic-specific performance shaping factors (PSF) for these actions was done in a manner consistent with the multiplier method that was selected. They felt there was not a systematic evaluation of the applicability and impact of the selected multiplier method on the overall results (e.g., sensitivities associated with location of the breaking points and/or the multiplier). In addition, the use of a multiplier method with only one breaking point at 0.8g is being now replaced in the industry by multipliers with more breaking points. While not stating that the Vogtle SPRA should adopt a new method, the reviewers felt there is a need for a more systematic assessment of the sensitivity of the model to the selected multiplier method.
- The Vogtle SPRA team relied on the internal events PRA for the evaluation of HEP dependency. The reviewers felt that the dependency evaluation should be revised for seismic specific new potential dependencies.
- The Vogtle SPRA adopts a standard full correlation of seismic failures, with a number of notable exceptions, that generate potentially non-minimal cutsets. While it is

apparent that the Vogtle SPRA team considered this effect, more documentation is needed to describe the process and the rationale for accepting the mathematical limitations of the selected approach.

- The relay identification and screening was adequately performed, however, at the time of the peer review of the Vogtle SPRA, the effect of chatter of those relays that were not screened was not yet included in the logic model. Recent SPRAs have shown that relay chatter is an important contributor to the overall seismic risk profile and therefore this model limitation needs to be addressed.
- The Vogtle SPRA has extensive documentation of the screening process used for the inclusion of seismic-induced failure in the model. The only notable exception is on the inclusion of very small LOCA. Uniquely from a number of other seismic PRAs, the Vogtle SPRA relied on actual walkdowns of RCS lines to support not modeling very small LOCA. This is a notable effort but the unique approach of the Vogtle SPRA in treating and screening very small LOCA requires more supporting documentation.
- The model provided for peer review was not fully completed in the sense that (apart from the missing relay information) the refinement of fragilities was not yet completed for the most significant contributors. At the time of the peer review, the most significant contributors were associated with seismic-induced floods that were then judged to be extremely conservative. While the fragility analysis of these components is being refined, potential modeling considerations also need to be performed. For example, if the current lead contributors would stand, the modeling of operator actions in those scenarios would not be consistent with the Capability Category II of the associated supporting requirement from Part 2.
- Documentation of the Vogtle SPRA seismic equipment list (SEL) was judged to be best practice by the peer review team. The database generated to track SEL items and to link it to relay assessment, fragilities and modeling is extremely powerful and well designed and conceived.
- The Vogtle SPRA is quantified with the CAFTA suite of codes (i.e., CAFTA, FRANX, ACUBE, and UNCERT) but also requires some post-processing to address code limitations. The documentation of the post-processing is sometimes missing and will be essential to be able to re-produce the results.
- The quality of the documentation of the results and insights is impacted by the model not having reached a steady state (i.e., fragility refinements, relays, etc.) and some documentation details were missing for that reason (e.g., documentation of insignificant cutsets). Nevertheless, the Vogtle SPRA team demonstrated the capability to use the model to guide the refinement of the analysis at different levels.
- The Vogtle SPRA team essentially relied on the UNCERT code for the propagation of the uncertainties in the SPRA. There is little explanation or documentation of the meaning of the uncertainties results.

The peer review concluded that the Vogtle seismic PRA 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 peer reviewers also noted that the relay modeling for the SPRA model that was reviewed was not yet fully completed but stated that the Vogtle SPRA team demonstrated the ability to use the existing SPRA to obtain results and interpret the insights. Some limitations in the documentation were noted, primarily attributed to the closeness in time of the peer review to the completion date of the model but the Vogtle SPRA team demonstrated good knowledge of the inherent limitations of the model and of the conservatism applied. The SPRA Database developed by the team was recognized as an outstanding tool which demonstrates how the Vogtle SPRA team is managing the model and its insights.

No new methodologies have been incorporated into the SPRA model since the peer review.

A.5. Summary of the Assessment of Supporting Requirements and Findings

Table A-1 presents a summary of the SRs graded as "Not Met" or not "Capability Category II," and lists the Finding F&Os associated with those SRs along with the disposition for each. Table A-2, provided at the end of this document due to its size, presents a summary of all the 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 VEGP SPRA Peer Review

SR	Assessed Capability Category	Associated Finding F&Os	Disposition to Achieve Met or Capability Category II
SHA			
SHA-C4	Not Met	12-18, 12-36	Associated F&O Findings have been resolved as noted in Table A-2. SR is judged to be Met.
SHA-H1	Not Met	12-18, 12-36	Associated F&O Findings have been resolved as noted in Table A-2. SR is judged to be Met.
SHA-I1	Not Met	12-15	Associated F&O Findings have been resolved as noted in Table A-2. SR is judged to be Met.
SHA-I2	Not Met	12-15	Associated F&O Findings have been resolved as noted in Table A-2. SR is judged to be Met.
SHA-J1	Not Met	12-1, 12-2, 12-11, 12-16	Associated F&O Findings have been resolved as noted in Table A-2. SR is judged to be Met.
SHA-J3	Not Met	12-8	Associated F&O Findings have been resolved as noted in Table A-2. SR is judged to be Met.
SFR			
SFR-A2	CC-I	14-1, 14-7, 14-10	Associated F&O Findings have been resolved as noted in Table A-2. SR is judged to be Met at CC-II.
SPR			
SPR-B2	Not Met	16-4, 16-6	Associated F&O Findings have been resolved as noted in Table A-2. SR is judged to be Met.
SPR-B4	Not Met	16-1	Associated F&O Findings have been resolved as noted in Table A-2. SR is judged to be Met.
SPR-F1	Not Met	12-31, 16-5	Associated F&O Findings have been resolved as noted in Table A-2. SR is judged to be Met.

A.6. 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 VEGP SPRA model meets the expectations for PRA scope and technical adequacy as presented in RG 1.200, Revision 2 [11] 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 SCDF and SLERF (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 is available if required to facilitate the NRC staff's review of this submittal.

The Vogtle Electric Generating Plant Units 1 and 2 SPRA reflects the as-built and as-operated plant as of the cutoff date for the SPRA, August 31, 2015. 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. Certain aspects of FLEX are permanently installed and operational without operator intervention, i.e., notably improved RCP seals, and these are reflected in the internal events PRA model, and therefore the SPRA model.

The peer review observations and conclusions noted in Section A.4, the F&O finding dispositions noted in the discussion in Section A.5, and the discussion in Section A.7 demonstrate that the VEGP 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 January 2017. No changes were made in updating the model that would require a subsequent focused peer review.

A.7. 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 VEGP internal events PRA and internal flooding PRA that forms the basis for the SPRA to determine compliance with ASME PRA Standard, RA-S-2002, including Addenda RA-Sb-2005 [27] and RG 1.200 Rev. 1 [28] in in May 2009. This review found that all but 3 supporting requirements (SRs) met at least Capability Category II. All of the internal events and internal flooding PRA peer review findings that may affect the SPRA model have been addressed. Since the time of the peer review, the internal events PRA has been maintained in accordance with the requirements for model configuration control in the PRA Standard.

The PWR Owners Group peer review of the VEGP SPRA was conducted in November 2014. The results of this peer review are discussed above, including resolution of SRs not assessed as meeting Capability Category II by the peer review, and resolution of peer review findings pertinent to this submittal. The peer review team expressed the opinion that the VEGP seismic PRA 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 review was that the VEGP SPRA is judged to be suitable for use for risk-informed applications.

- Table A-1 provides a summary of the disposition of SRs judged by the peer review to be not met, or not meeting Capability Category II.
- Table A-2 provides a summary of the disposition of the SPRA peer review findings.
- Table A-3 provides an assessment of the expected impact on the results of the VEGP SPRA of any SRs and peer review Findings.

Of the peer review finding-level Facts and Observations (F&Os) listed in Table A-2, most were associated with PRA Standard supporting requirements (SRs) that were deemed by the peer reviewers to be either "Met" or met at "Capability Category II." This indicates, as can be seen from the finding details, that these findings deal with relatively focused issues that have been adequately dispositioned within the reviewed methodologies, for the SPRA and for future risk-informed application. Many of these were documentation-related.

The remaining finding-level F&Os are associated with SRs deemed by the peer reviewers to be "Not Met", or to not meet "Capability Category II." These are as listed in Table A-3.

Table A-3 Findings associated with Not Met/CC-I SRs

SR	Findings	Summary of Issue Not Fully Resolved	Impact on SPRA Results
SHA-C4	12-18, 12-36	Finding issues are resolved.	No impact on SPRA results.
SHA-H1	12-18, 12-36	Finding issues are resolved.	No impact on SPRA results.
SHA-I1	12-15	Finding issues are resolved.	No impact on SPRA results.
SHA-I2	12-15	Finding issues are resolved.	No impact on SPRA results.
SHA-J1	12-1, 12-2, 12-11, 12-16	Finding issues are resolved.	No impact on SPRA results.
SHA-J3	12-8	Finding issues are resolved.	No impact on SPRA results.
SFR-A2	14-1, 14-7, 14-10	Finding issues are resolved.	No impact on SPRA results.
SPR-B2	16-4, 16-6	Finding issues are resolved.	No impact on SPRA results.
SPR-B4	16-1	Finding issues are resolved.	No impact on SPRA results.
SPR-F1	12-31, 16-5	Finding issues are resolved.	No impact on SPRA results.

As this list indicates, there were only 10 Not Met / Capability Category I SRs associated with the finding F&Os.

- Of these, 6 are seismic hazard-related SRs, for which the findings were associated with: (a) inadequate documentation of the hazard analysis performed; (b) demonstration that sufficient consideration has been given to more recent geologic events and associated modeling; or (c) sensitivity calculations for the models and parameters used in the site hazard. The identified issues have been addressed, as noted in the dispositions for the affected findings in Table A-2.
- One of the SRs is fragilities-related. Two of the 3 findings associated with this SR deal with conservatisms that the reviewers noted, which have now been addressed within the analytical methodology that the peer reviewers found acceptable. The remaining finding is associated with a specific polar crane fragility issue, which has also been addressed within the reviewed methodology.
- Three of the SRs are PRA modeling-related. Three of the findings associated with this SR are related to implementation of the seismic performance shaping factor approach in the human reliability analysis. The comments in those findings have been addressed and implemented in the SPRA model, within the reviewed methodology, without significant impact on the results. One finding was related to the relay chatter evaluation, for which the model update resolves the finding. The last finding was related to the SPR documentation, which has been updated to resolve the finding.

The SPID [2] defines the principal parts of an SPRA, and the VEGP SPRA has been developed and documented in accordance with the SPID. The information in the tables identified above demonstrates that the VEGP SPRA is of sufficient quality and level of detail for the response to the NTTF 2.1 Seismic SPRA submittal.

A.8. Identification of Key Assumptions and Uncertainties Relevant to the SPRA Results.

The 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 [13] and EPRI 1016737 [14] provide guidance on assessment of uncertainty for applications of a PRA. As described in NUREG-1855 [13], 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 VEGP 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 VEGP SPRA technical elements are noted in the SPRA documentation that was subject to peer review, and a summary of important modeling assumptions and associated sensitivity evaluations 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 VEGP SPRA is listed in Table A-4.

Table A-4 Summary of Potentially Important Sources of Uncertainty

PRA Element	Summary of Treatment of Sources of Uncertainty per Peer Review	Potential Impact on SPRA Results
Seismic Hazard	The VEGP 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 VEGP site. The review team commented that the site response analysis did not fully evaluate and model aleatory and epistemic uncertainties in the site response analysis.	With regard to aleatory and epistemic uncertainties in site response analysis, there is an abundance of site-specific data from VEGP Units 3&4 that reduces epistemic uncertainty to an insignificant level. The documentation has been expanded to demonstrate that the current analysis adequately represents the Vogtle site. The characterization of the seismic hazard reasonably reflects sources of uncertainty.
Seismic Fragilities	No specific peer review team comments on sources of uncertainty in fragilities.	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 commented that the Vogtle SPRA team relied on the UNCERT code for the propagation of the uncertainties in the SPRA with little explanation or documentation of the meaning of the uncertainties results.	The discussion of uncertainty has been expanded in the SPRA Quantification (QU) report, including a discussion of sources of model uncertainty, and potentially important sources have been addressed in the sensitivity analysis. 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, and sensitivities performed to evaluate the impact of possible changes to address these.

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

The VEGP SPRA reflects the plant as of the cutoff date for the SPRA, which was August 31, 2015. Table A-5 lists provides a summary of plant changes not included in the model and provides a qualitative assessment of the likely impact of those changes on the SPRA results and insights. There are no significant plant changes that have not been reflected in the current SPRA model.

Table A-5 Summary of Significant Plant Changes Since SPRA Cutoff Date

Description of Plant Change	Impact on SPRA Results
Safety-related battery chargers are no longer operated in a load-share configuration. Instead, a single charger will be in service and if it fails, the other charger will be placed in service by operator action.	An assessment of this change on the VEGP internal events PRA model indicated no significant impact. Further, the battery chargers are modeled as seismically correlated. Thus, modeling of the change in the SPRA would not affect the SPRA results.
Permanently installed and portable FLEX equipment other than low leakage RCP seals have not been modeled in the SPRA.	Credit for such equipment is likely to improve the SPRA results (SCDF and SLERF) but the impact is difficult to quantify without detailed modeling.

Table A-2 Summary of Finding F&Os and Disposition Status [1]

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
SHA-E2	11-3	While variability in the mean base-case Vs profile is incorporated in the site response analysis, no epistemic uncertainty in the base-case profile is represented. Documentation of the justification for this assessment should be expanded. (This F&O originated from SR SHA-E2)	To maintain hazard-consistent ground motion hazard at the control point, the site response analysis needs to incorporate appropriate epistemic uncertainty and aleatory variability in its inputs. The Vs profile for the Vogtle Units 1&2 site is represented by a single Vs profile, indicating there is no epistemic uncertainty in the mean base-case profile. Documentation of this assessment needs to be expanded. Discussion with staff indicates that consideration of the combined data for the Vogtle site (Units 1&2, Units 3&4, ISFSI) provides sufficient confidence that a single mean base-case profile characterizes the site. This conclusion is based on the quantity and quality of the combined data and an evaluation showing the site is relatively uniform with respect to Vs. For some depth ranges, data from the nearby Savannah River Site (SRS) are used to support the	Expand documentation to demonstrate that a single base-case Vs profile adequately represents the Units 1&2 site. Or if that is not the case, include epistemic uncertainty in the characterization of Vs profile and evaluate the impact on control point ground motions.	There is an abundance of site-specific Vs data from VEGP Units 3&4, which reduces epistemic uncertainty to an insignificant level. Additional discussion of the rationale for use of a single base-case Vs profile for the site has been included in the documentation. The added discussion demonstrates that a single base-case shear-wave velocity (Vs) profile adequately represents the Vogtle site, based on the availability of Vs data, which reduces the epistemic uncertainty for this particular parameter. This finding has been resolved with no significant impact to the SPRA results or conclusions.

[1] In Table A-2, all but the last column are extracted directly from the Peer Review report. The last column provides the disposition for the Findings.

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
			profile interpretation. Bechtel Document 23162-000-G65-GEK-00010 (SNC #SV0-GB-X7R-011-001) presents summaries of velocity data, but does not provide sufficient information to support the lack of epistemic uncertainty at the Units 1&2 site over the complete depth range of the Vs profile. This would typically require multiple measurements throughout the depth range that provide a consistent picture of natural variability about a single mean base-case profile. The technical basis and justification that a single base-case profile is appropriate should be provided in more detail. This should include the basis for applying conclusions from other Vogtle locations to the Units 1&2 site. [A related Suggestion 11-2 addresses specifically potential epistemic uncertainty in the Blue Bluff Marl stratum.]		

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
SHA-E2	11-8	Upper crustal site attenuation of ground motion (kappa) is, generally, an uncertain parameter. Thus, to maintain hazard-consistent ground motion at the control point, this uncertainty should be incorporated in the site response analysis, or the basis for not including it should be provided. In either case, the technical basis and justification should be documented. (This F&O originated from SR SHA-E2)	Calculation X2CFS129 Ver2 notes that the damping associated with the base-case profile corresponds to a total kappa value for the soil column of 0.01 sec. The report does not address epistemic uncertainty in kappa. In discussion with staff during the peer review, it was noted that randomization of the damping associated with the profile layers represents both random variability and epistemic uncertainty. It was also noted that kappa was expected to be small for the Vogtle site and uncertainties in that small value would not be expected to have a significant impact on site amplification. Staff also noted that the approach used had been reviewed by the NRC for the Vogtle ESP and COLA. The SPID (EPRI, 2013) provides guidance accepted by the NRC for response to NTTF 2.1 Recommendation: Seismic that indicates kappa is difficult to measure and thus subject to large uncertainty (SPID Section B-5.1.3.2). Documentation of the technical	Provide a basis in the documentation for representing base-case kappa at the site by a single value. The basis might include sensitivity analyses to show the impact of epistemic uncertainty in kappa.	A discussion of the range of possible values of deep soil damping has been included in the documentation. A sensitivity study on the epistemic uncertainty of deep soil damping has been performed using median, lower range, and upper range alternatives for deep rock damping. Site response analysis was performed using 1E-4 HF and LF rock input motion. The resulting amplification functions and log-standard deviation were weight-averaged and compared to the original base case for each of BBM High PI and BBM Low PI soil columns. It was concluded that the inclusion of alternative base cases for deep soil damping to account explicitly for the epistemic uncertainty associated with site kappa does not have any significant effects on

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
			basis for kappa characterization should be expanded.		the resulting seismic hazard curves and UHRS. The sensitivity study has been added to the SPRA documentation. This finding has been resolved with no significant impact to the SPRA results or conclusions.
SHA-J1	12-1	As part of the PSHA implementation, the analyst has different alternatives for modeling the earthquake occurrences in the calculations. The PSHA documentation does not describe the approach that was used to model earthquakes. (This F&O originated from SR SHA-J1)	The approach that was taken to model earthquakes in the PSHA calculation was not identified. There are two basic alternatives that can be used to model earthquake events; as extended fault ruptures, or as point sources. The approach that is used influences how the CEUS ground motion model is implemented. No documentation is provided on either of these subjects (earthquake source modeling and use of the ground motion attenuation models). From questions posed to the PSHA analysts, it is our understanding that earthquakes were modeled as point sources and the appropriate ground motion	Documentation should be provided that describes how seismic sources are modeled in the PSHA (i.e., how the SSC and GMMs) were implemented in the Vogtle PSHA.	A PSHA report has been prepared that describes how earthquake events were modeled for area sources in the PSHA calculations. This was by modeling each earth-quake as a point source, and using correction factors for distance and ground motion uncertain-ty that modify the ground motion estimate to in-clude the effect of a closer distance to a fault rupture (because the rupture may be closer to the site than the single point used to represent that event) and the un-certainty in ground

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
			aleatory uncertainty was used in the calculation.		motion because the azimuth of the rupture is unknown. These correction factors were published in EPRI (2004) [12]. This finding has been resolved with no significant impact to the SPRA results or conclusions.
SHA-J1	12-11	As part of the PSHA implementation, the analyst has alternatives for modeling the earthquake occurrences in the calculations. The PSHA documentation does not describe the approach that was used to model earthquakes in RLME sources. (This F&O originated from SR SHA-J1)	The PSHA analysts were asked to describe the approach that was used to model earthquakes in the Charleston RLME seismic source. The response indicated that earthquakes in the Charleston RLME source were modeled using 'pseudo faults'. The PSHA report does not: 1. Describe that a 'pseudo fault' approach was used to model earthquakes in the Charleston RLME source. 2. Provide a definition of 'pseudo faults'. 3. Describe how the 'pseudo fault' approach was implemented for the Charleston RLME seismic source (e.g., what was the fault spacing that was used; how was the earthquake rate distributed to	Provide a description of the earthquake modeling approach that was used to model the Charleston RLME seismic source and how the approach was implemented.	A PSHA report has been prepared that describes how pseudo-faults were implemented to represent the Charleston RLME source. This includes: 1. A description of the pseudo-faults. 2. A definition of pseudo-faults as constructed faults that represent possible sources of future large earthquakes. 3. Implementation of the pseudo-faults including spacing and limits at the borders of the Charleston source. 4. Documentation of the rupture area, length, and width that were estimated for possible

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
			the faults, etc.). 4. Document the fault rupture model that was used. 5. Describe how earthquake events are distributed on the faults.		future earthquakes. 5. A description of how earthquake ruptures are distributed on the faults. This finding has been resolved with no significant impact to the SPRA results or conclusions.
SHA-I1, SHA-I2	12-15	A screening assessment was performed for soil liquefaction and is described in seismic fragility calculation (PRA-BC-V-14-025). A screening assessment was not performed for other potential seismic hazards. (This F&O originated from SR SHA-I1)	A screening analysis was not performed for hazards such as settlement, fault displacement, tsunami, seiche, etc. It is anticipated these other seismic hazards will be screened out.	A screening analysis for other seismic hazards should be performed and documented as part of the PSHA and SPRA. It is expected that information in the FSAR for Vogtle 1 & 2 and in the COLA for Units 3 & 4 can be used to support this requirement.	This evaluation was done for the Vogtle 3&4 COLA [38] and is noted in the ESP SAR [36]. The Vogtle 3&4 evaluation is applicable to, and has been cited in, the Vogtle 1&2 SPRA Fragility report. This finding has been resolved with no significant impact to the SPRA results or conclusions.
SHA-J1	12-16	The Vogtle PSHA has gone through a number of changes and revisions since 2012 due to changes in models, input data, etc. As new calculations were performed and reports generated, sensitivity results, were not carried forward. As a result, there does not exist a current report that includes all	The documentation of the PSHA is provided in a collection of documents that were prepared in the 2012-2014 time frame. There does not exist a single document that contains a set of results that is based on the current PSHA model.	Prepare a complete and up-to-date PSHA document that includes all results, sensitivity calculations, deaggregation results, etc. that is based on the current model.	A PSHA report has been prepared that includes hazard results, uncertainties in hazard, and sensitivities to input uncertainties; this summarizes hazard results for the Vogtle site.

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
		PSHA results, deaggregations, etc. that is based on the current PSHA model. (This F&O originated from SR SHA-J1)			This finding has been resolved with no significant impact to the SPRA results or conclusions.
SHA-B2, SHA-C4, SHA-H1	12-18	The Vogtle PSHA is based on the CEUS SSC seismic source model which was completed in 2012. The SSC model was a developed at a regional scale that was based on data gathered up until about 2010. (Note, the date when data was gathered varied; for example the earthquake catalog was complete through 2008.) In the sense that the CEUS SSC model was not specifically performed as a site-specific PSHA for the Vogtle site. (This F&O originated from SR SHA-B2)	As part of a site-specific PSHA, there is a need to gather, review and evaluate new geological, seismological, or geophysical information or information that is defined at a scale that was not considered in the development of the CEUS SSC model. As part of the Vogtle SPRA, no effort was made to gather up-to-date and local (local to the Vogtle site) information to evaluate whether any new information has become available on active faulting and/or the development new seismic sources or the revision of sources in the CEUS SSC model in the vicinity of the Vogtle plant. Since up-to-data was not gathered, consideration of alternatives could not be addressed.	A data gathering effort should be undertaken to identify new information that post-dates the CEUS SSC data collection effort. The data gathering effort should also look for information local to the Vogtle site region that was not considered, or at a scale that was not addressed as part of the CEUS SSC regional evaluation. Some of this information may be available in the COLA for Vogtle Units 3 & 4.	A detailed study of new geological, seismological, and geophysical information was conducted, to determine if any information subsequent to the EPRI SSC model (EPRI, 2012 [35]) is available that should be incorporated into the seismic hazard results for Vogtle. This study is described in the SPRA documentation. While the area around the site continues to be studied by many earth scientists, there was no new information identified that would change the estimate of seismic hazard for Vogtle. This finding has been resolved with no significant impact to the

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
					SPRA results or conclusions.
SHA-J1	12-2	The method that is used in the Vogtle PSHA to estimate the soil site hazard is not described or referenced. (This F&O originated from SR SHA-J1)	For soil sites, the soil hazard is generally (though not exclusively, since other methods could be used) determined in two steps; probabilistic rock hazard results are estimated which are then combined with probabilistic estimates of the site response. The method used in the Vogtle PSHA to estimate the soil hazard is not described.	The documentation should include a description of the methodology that is used to combine the rock hazard results and the site amplification factors to determine the soil hazard at the Vogtle site.	The methodology used for the surface hazard calculation has been described in detail, and a comparison made between the GMRS using the two approaches 2A and 3. Approach 2A was used for the calculation of SSI input motions at foundation elevations and Approach 3 was used for the calculation of surface hazard and GMRS at the ground surface, as defined in NUREG/CR-6728 [34]. It was concluded that the use of Approach 2A USHRS as input to the SSI analysis of the Vogtle plant is considered acceptable and does not present any significant inconsistency with the seismic hazard curve and GMRS at the ground surface, which were

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
					calculated using Approach 3. This finding has been resolved with no significant impact to the SPRA results or conclusions.
SHA-E2	12-22	The site response Calculation X2CFS129 Ver. 1 (2012) and Ver. 2 (2014) does not describe a framework for evaluating and characterizing sources of aleatory and epistemic uncertainty and how the approach was implemented. (This F&O originated from SR SHA-E2)	The site response calculation does not present a clear description of how aleatory and epistemic uncertainties are identified and evaluated. As a result it is difficult to track the propagation of uncertainties is carried out in the site response analysis. It is worth noting that there is some epistemic site response uncertainty that is accounted for in the rock GMPEs.	A framework and approach for evaluating and modeling uncertainties in the site response should be developed and implemented. The site response calculation documentation should fully describe the methodology and its implementation.	A description of the methodology used to account for epistemic and aleatory uncertainties in soil hazard has been added to the documentation. This finding has been resolved with no significant impact to the SPRA results or conclusions.
SPR-E5	12-23	The quantification process has included the uncertainties in the seismic hazard, fragility and systems-analysis elements of the SPRA. The results in Table 5.1 are internally inconsistent and are inconsistent with the results reported in Sections 3 and 4 for CDF and LERF, respectively.	Table 5-1 presents the results of three different uncertainty calculations for CDF and LERF. In addition, point estimates for CDF and LERF are calculated and reported in Section 5.1.1. Thus the table reports two estimates of the mean CDF and LERF respectively from different uncertainty calculations and a 'Point Estimates' result for each.	Develop and document an understanding of the earlier point estimate results for CDF and LERF (as reported in Sections 3 and 4) and of uncertainty results.	Additional detail has been added to the SPRA Quantification report to document the uncertainty, importance, and sensitivity analyses and relate the uncertainty analysis mean CDF and LERF to the point estimate values.

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
		(This F&O originated from SR SPR-E5)	All of these results are different than the point estimate (approximate mean) reported in Sections 3 and 4 for CDF and LERF, respectively. The documentation in the report does not describe the basis (inputs) for these calculations, or offer an interpretation of the results.		This finding has been resolved with no significant impact to the SPRA results or conclusions.
SPR-E5	12-24	The Quantification report does not provide documentation of the uncertainty analysis results. (This F&O originated from SR SPR-E5)	The uncertainty analysis is presented in Section 5.1 with the results reported in Table 5.1. The report provides limited discussion of the results and the insights that might be gained from them. The two sets of results that are reported in Table 5-1 are not discussed in terms of their relationship to each other. For instance the mean values should be the same (but are not). The uncertainty estimates provide insight to the total uncertainty and the contribution of the basic event uncertainty to the total. In addition, neither Table 5.1 or the discussion identifies what is the 'final' uncertainty result that includes the propagation of uncertainties of all elements of the	Provide documentation of the uncertainty analysis that describes the results, how they are being interpreted and the insights that are derived from them.	Additional detail has been added to the documentation of the seismic plant response model, model implementation, and quantification in the QU report. In addition, the uncertainty, importance, and sensitivity analyses are described in more detail. This finding has been resolved with no significant impact to the SPRA results or conclusions.

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
			SPRA to the estimates of CDF and LERF.		
SPR-E5	12-26	There are differences in the results for CDF and LERF that are reported in Table 5.1. A possible contributor to these differences may be due to the number of Monte Carlo simulations that were performed. (This F&O originated from SR SPR-E5)	The report does not present the results of sensitivity calculations with regard to the number of Monte Carlo simulations that are needed to produce stable results. It is our understanding from discussion with the PRA staff that these types of sensitivity calculations were performed.	Document the results of sensitivity calculations on the number of Monte Carlo simulations required to produce stable results.	Updated Monte Carlo uncertainty runs have been performed with 20,000 iterations for SCDF and SLERF. This is a sufficiently high number of simulations to produce a stable result. The SPRA documentation has been updated to clearly indicates the results. This finding has been resolved with no significant impact to the SPRA results or conclusions.
SPR-F2	12-27	Documentation should be provided that describes how the plant model analysis is quantified. (This F&O originated from SR SPR-F2)	The current quantification document does not provide a clear description of the how the plant model is quantified. For example the discussion does not identify how calculations are performed, what the limitations of these quantifications are and how they affect the results.	Provide clear and complete documentation of the approach used to quantify the seismic plant response model, to perform the risk quantification, uncertainty analysis, and importance analysis.	The QU report documentation has been updated to describe the quantification process, including the technique for combining cutsets over the 14 acceleration intervals, and obtaining the importance measures. This finding has been resolved with no significant impact to the

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
					SPRA results or conclusions.
SPR-E2	12-29	The Quantification report provides limited documentation of the process and methods that were used to perform the uncertainty analysis. (This F&O originated from SR SPR-E2)	There is limited documentation of the process and the numerical methods that were used to perform the uncertainty analysis. Based on the documentation that is provided and discussions with the PRA staff there is limited but not complete understanding of the methods that were used and the relationship of these methods to the results were obtained (reported in Table 5.1). In some cases (as described in the documentation) the results from the uncertainty analysis (Table 5.1) are not the same as the results reported in Sections 3 and 4 for CDF and LERF (though this connection is not clearly stated in the report). However, it would seem the results in Table 5.1 should be internally consistent.	Document the process and methods that were used to perform the uncertainty analysis. Where appropriate document where consistencies and potential inconsistencies in results might be expected.	Additional detail has been added to the QU report to document the uncertainty, importance, and sensitivity analyses and relate the uncertainty analysis mean SCDF and SLERF to the point estimate values. This finding has been resolved with no significant impact to the SPRA results or conclusions.
SPR-F1	12-31	The standard requires a level of documentation that provides an understanding of the seismic plant response model and the quantification. This requirement is not met.	There is limited documentation that describes the seismic plant response analysis and quantification; how the model was implemented, how the quantification was performed and a discussion of the analysis	Documentation should be provided in sufficient detail that describes the seismic plant model, how it is implement and quantified.	Additional detail has been added to the documentation of the seismic plant response model, model implementation, and quantification in the QU

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
		(This F&O originated from SR SPR-F1)	results. To meet this requirement, the documentation must be in considerable detail in order to support the review process and future updates. Part of the documentation should include a detailed discussion of the results, sensitivity calculations, and the uncertainty analysis.		report. In addition, the uncertainty, importance, and sensitivity analyses are described in more detail. This finding has been resolved with no significant impact to the SPRA results or conclusions.
SPR-F3	12-32	The documentation of the sources of model uncertainty and a description of the analysis assumptions is not complete in the SPRA quantification report. In addition, there is not a clear description of the uncertainty analysis and the contributors to the total uncertainty beyond a simple report from UNCERT. (This F&O originated from SR SPR-F3)	The purpose of this supporting requirement is that documentation should be presented that addresses the sources of epistemic (knowledge) uncertainty that are modeled and their contribution to the total uncertainty in CDF and LERF. In addition, the documentation should discuss elements of the seismic plant model where there may be latent sources of uncertainty that are not modeled and assumptions that are made in performing the analysis.	Document and discuss the contribution of the different sources of uncertainty that are modeled in the SPRA.	The documentation of the uncertainty analysis has been expanded in the Quantification report. A discussion of sources of model uncertainty has been added to the report, and potentially important sources have been addressed in the sensitivity analysis. This finding has been resolved with no significant impact to the SPRA results or conclusions.

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
SHA-B3, SHA-C4, SHA-H1	12-36	As part of a site-specific PSHA, an up-to-date earthquake catalog should be used. The CEUS SSC study involved the development of a comprehensive earthquake catalog based on data through 2008. The Vogtle site-specific PSHA should consider the impact SSC of any additional seismicity since 2008 up to the time the study started. (This F&O originated from SR SHA-C4)	As part of the Vogtle PSHA an effort was not made to gather data on earthquakes that occurred since 2008. As such, the analysts did not assess whether more recent seismicity is consistent with the characterization parameters estimated as part of the CEUS SSC study (NRC, 2012). We note that as part of the Vogtle PSHA, calculations were performed to recompute the seismic hazard at the site to take into account changes in the CEUS SSC earthquake catalog through 2008 that were made following the completion of the CEUS SSC study. These changes reflect the identification of reservoir induced seismicity earthquakes and the re-interpretation of the location of some earthquakes in the Charleston, SC area that occurred in the 1880's (EPRI, 2014). References EPRI (2014). Review of EPRI 1021097 Earthquake Catalog for RIS Earthquakes in the Southeastern U. S. and Earthquakes in South Carolina	An up-to-date earthquake catalog for the Vogtle site region should be developed to assess whether modifications to the seismic source recurrence parameters or required. The updated catalog, resources used in compiling the update and the results of the evaluation should be documented as part of the PSHA. If more recent seismicity is not consistent with the existing CEUS SSC seismic source parameters, the parameters should be updated and the PSHA should be updated.	An update to the earthquake catalog was prepared from the time of the CEUS SSC catalog (through 2008) through February 2016. The rate of occurrence of earthquakes within 320 km of the Vogtle site was compared to the rate of earthquakes represented by the CEUS SSC seismic source model for that same area, this comparison being made for M>2.9. It was found that the updated catalog implied a rate of earthquakes that is lower than the mean rate from the CEUS SSC seismic sources. Therefore, incorporating the effects of a updated catalog on the hazard at Vogtle would decrease the hazard slightly, and was not undertaken. This comparison is documented in the SPRA documentation. This finding has been resolved with no

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
			Near the Time of the 1886 Charleston Earthquake Sequence, transmitted by letter from J. Richards to R. McGuire on March 5, 2014.		significant impact to the SPRA results or conclusions.
SHA-J3	12-8	A foundational element of PSHA as it has evolved over the past 30 years is the development and implementation of methods to identify, evaluate, and model sources of epistemic (model and parametric) uncertainty in the estimate of ground motion hazards. As such fairly rigorous analyses are carried out (SSHAC studies) to quantitatively address model uncertainties. At the same time there is within any analysis sources of uncertainty that are not directly modeled and assumptions that are made for pragmatic or other reasons. There are also sources of model uncertainty that are embedded in the context of current practice that are 'accepted' and typically not subject to critical review. For instance, in the PSHA it is standard practice	The documentation of the sources of model uncertainty analysis and a description of the analysis assumptions is not complete in the PSHA report in its current form such that a clear understanding of the contribution of individual sources of uncertainty to the estimate of hazard are understood. Limited information on the contribution of seismic sources to the total mean hazard is presented, but information on the contributors to the uncertainty is not provided. With respect to addressing model uncertainties and associated assumptions there are some examples that can be identified in the Vogtle PSHA. For example, in the site response analysis the assumption is made that the 1D equivalent linear model (SHAKE type) to estimate the site amplification and ground motion input to plant structures is appropriate.	The resolution to this finding could involve: 1. Documentation and discussion of the contribution of different sources of uncertainty that are modeled in the PSHA. The documentation of the contribution of different sources of uncertainty can be shown by means of 'tornado plots' that quantify the sensitivity of the hazard at different ground motion levels to the various branches in the logic tree. These plots show which sources of epistemic uncertainty are most important. It should include the source model uncertainty, ground motion model uncertainty, and site response uncertainty. Currently, the total uncertainty is shown by the hazard fractiles, but it is not broken down to provide understanding as to what is most important.	Sources of uncertainty in the seismic hazard analysis for Vogtle are discussed in the updated SPRA documentation. These include uncertainty in seismic source model (for background earthquake sources and for the Charleston RLME), in maximum magnitude for background seismic sources and for the Charleston RLME, in ground motion prediction equation, in smoothing assumptions for seismicity parameters in background sources, and in site amplification model. "Tornado plots" are included in the updated SPRA documentation that show the contribution to total uncertainty in seismic hazard from source model uncertainty,

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		to assume that the temporal occurrence of earthquakes is defined by a Poisson process. This assumption is well accepted despite the fact that it violates certain funda-mentally understanding of tectonic processes (strain accumulation). A second practice is the fact that earthquake aftershocks are not modeled in the PSHA, even though they may be significant events (depending on the size of the main event). In the spirit of the standard it seems appropriate that sources of model uncertainty that are modeled as well as sources of uncertainty and associated assumptions as they relate to the site-specific analysis should be identified/ discussed and their influence on the results discussed. As SPRA reviews and the use of the standard has evolved, it would seem the former interpretation is reasonable, but potentially incomplete. It is reasonable from the perspec-tive that document-tation of the sources of model uncer-		2. Identification and discussion of model assumptions that are made.	maximum magnitude uncertainty, ground motion prediction equation uncertainty, smoothing assumptions for seismicity parameters in background sources, and site response uncertainty. These plots are presented for 10 Hz and 1 Hz spectral acceleration, for ground motion amplitudes corresponding to mean annual frequencies of exceedance of 1E-4 and 1E-5. These "tornado plots" show that ground motion prediction equation is the major contributor to seismic hazard uncertainty for both 10 Hz and 1 hz spectral acceleration, and maximum magnitude of the Charleston RLME source is an important contributor for 1 Hz spectral acceleration. The use of equivalent linear one-dimensional site response analysis, and its associated

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
		tainty and their contribution to the site-specific hazard results is a valuable product that supports the peer review process and assessments in the future as new information becomes available). Similarly, documenting assumptions provides similar support for peer reviews and future updates. The notion that model uncertainties and related assumptions that are not addressed in the PSHA is at a certain level an extreme requirement that may not be readily met and may not be particularly supportive of the analysis that is performed. For purposes of this review, the following approach is taken with regard to this supporting requirement: 1. The documentation should present quantitative results and discussion the sources of epistemic uncertainty that are modeled and their contribu-tion to the total uncertainty in the seismic hazard. 2. The documentation should discuss elements of the			assumptions, and its adequacy for the Vogtle site are documented in the hazard calculation. This finding has been resolved with no significant impact to the SPRA results or conclusions.

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
		PSHA model where their may be latent sources of model uncertainty that are not modeled and assumptions that are made in performing the analysis. (This F&O originated from SR SHA-J3)			
SFR-A2	14-1	The conservatisms that exist in structural demand were not properly accounted for in the estimation of component and structure fragilities. (This F&O originated from SR SFR-A2) .	SFR-A2 requires that seismic fragilities be based on plant-specific data and that they are realistic and median centered with reasonable estimates of uncertainty. The structural response factor used in all component fragilities reviewed is reported as 1.0. This factor will be greater than 1.0 because of the conservatism introduced in the demand through the structural analysis. Because of this, the component and structural fragilities are biased low. The fragilities developed for structures and components that are mounted in those structures will be biased low because the input structural demands include conservatisms. Time histories used for the SSI analysis have been processed such that each	Account for conservatism in the building response analyses in the structure response factor for component fragility evaluations. Use clipped spectra for assessing anchorage capacities.	Evaluation of anchorage has been updated to include clipping of in-structure response spectra, and the methodology is documented in the fragility notebook. Structure response is dominated by the soft soil on which Vogtle 1 and 2 structures are founded. This would cause higher damping at lower hazard frequency levels and lead to stress similar to the stress calculated for the buildings at 1E-4. As a result the structural response factor is close to 1 and is accounted for appropriately in the fragility evaluations.

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
			record envelopes the target UHRS. This will introduce some level of conservatism. The input motion at the control point has been scaled to produce resultant FIRS that envelopes the FIRS coming out of the site-consistent input motion analysis. In structure response spectra coming out of the SSI analyses were not peak clipped when computing anchorage demands. Structure response at the calculated equipment fragility levels is considerably higher than the 1E-4 UHRS considered in the building response analyses. The structure will have additional cracked shear walls and higher associated levels of damping at these higher ground motions.		The input time history motion at the control point in the SSI analysis has been modified to reasonably match the corresponding 1E-4 UHRS from the site-consistent input motion analysis. This finding has been resolved.
SFR-A2	14-10	Significant conservatisms were noted in several sampled fragility calculations. (This F&O originated from SR SFR-A2)	In the fragility calculations of heat exchangers (PRA-BC-V-14-009 Appendix A), nozzle loads significantly contribute to the seismic demands which form the basis for the median capacities. Based on in-plant walkdowns by the peer review teams and also noted in the walkdown report, the piping is well supported in all directions and will not impose significant nozzle loads during a seismic event. The CCW and	Realistic nozzle loads should be determined for fragility evaluation of heat exchangers. The equipment capacity factor should be based on the frequency range of interest. That frequency range of interest is centered at the fundamental frequency of the	The CCW and ACCW heat exchanger capacities have been updated to reflect realistic nozzle loads. The equipment fragilities have been updated to account for appropriate frequency, and uncertainty has been considered in these updates.

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
			ACCW capacities are below the 2.5g screening level and are significant contributors to risk so more realistic fragilities are required. Battery rack 11806B3BN3 in calculation PRA-BC-V-14-010 Appendix J2 is governed by GERS capacity. The GERS capacity is taken to be 1g, which corresponds to a frequency of 1 Hz. This is not realistic. The actual capacity is about 4g. The median capacity reported in the calculation is well below the 2.5g screening level and is not realistic. The median capacity reported for the Turbine Driven Auxiliary Feedwater Pump is reported in Calculation PRA-BC-V-14-008 as 1.56g. This fragility is based on the seismic qualification document. The frequency range of interest for the fragility evaluation should be centered around the fundamental frequency of the assembly and not consider the entire frequency range.	pump, and considers some uncertainty in that frequency.	This finding has been resolved.

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
SFR-G2	14-14	The iterative process used for developing realistic fragilities is not well documented. (This F&O originated from SR SFR-G2)	In review of the seismic fragility calculation for the safety features sequencer (11821U3001), it was discovered that an iterative process was used. The initial fragility is based on EPRI 6041 screening methodology and an equipment capacity factor that is equal to the EPRI 6041 median capacity divided by the peak in structure demand. If this value is less than the screening capacity (2.5g), then the fragility may be refined by examining the component fundamental frequency. The fragility may be further refined by examining component specific qualification test reports. However, the fragility used in the logic tree by the systems analyst is generally the highest of these computed. This is reasonable and appropriate, however, this process is not described in the fragility notebook or fragility calculations.	Add a description of the iterative process for computing the component fragilities in the SPRA documentation	The description of the iterative process for computing fragilities has been documented. This finding has been resolved with no significant impact to the SPRA results or conclusions.
SFR-D2	14-17	Inconsistencies and errors in NSSS fragility development. (This F&O originated from SR SFR-D2)	Fragilities for the Vogtle 1&2 Nuclear Steam Supply System (NSSS) are based on the results of the Westinghouse analysis of record (AOR) associated with the safe shutdown earthquake (SSE). In general, fragilities are developed through scaling of the	Update SNC calculation no. PRA-BC-V-14-015 to incorporate corrections and enhancements.	The following changes have been made: NSSS fragility calculations have been updated to reflect Westinghouse-provided critical loads and support capacities represented in the critical failure modes;

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
			SSE demands to the RLE and using the AOR seismic margins. Various deficiencies were noted in the development of the fragilities associated with these components. Basis: The NSSS Seismic fragility evaluation (SNC calculation no. PRA-BC-V-14-015) includes detail calculations for each of the major NSSS components. It indicates that the critical failure modes for the components are controlled by the support capacities. During the Peer Review, the team members discussed these issues with SNC staff to obtain insights and develop potential resolution paths. Key issues included: (a) Basis for assumption that the support capacities represented the critical failure mode was not documented. SNC indicated that this was based on input from Westinghouse and NUREG-3360 and will update the fragility evaluation of provide this information. (b) Inelastic energy absorption was not credited to increase the median capacities - this does not result in realistic median capacities (overly conservative). (c) Reactor Coolant Pump fragility		the effect of inelastic energy absorption is factored in and documented in fragility calculation as appropriate; the Reactor Coolant Pump fragility has been updated to reflect the failure of the pump associated with LOCA; the reactor internals fragility has been updated in the calculation; and the new fragilities have been reflected in the updated SPRA model. This finding has been resolved.

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
			was based on consideration of the failure of the attached CCW piping, due to an assumption that a small-break/RCP seal LOCA was critical. It was learned during the Peer Review that failure in the system model was linked to a large-break LOCA, so the failure mode considered in the fragility evaluation is not consistent with the system model - SNC indicated that they will revise the fragility evaluation. (d) Reactor Internal fragility evaluation determined the demand based an average spectral acceleration over the range of 2 to 3 Hz, rather than using the peak acceleration in this range of the ISRS, and did not consider the contribution of higher modes. SNC indicated that this was done to avoid an overly conservative capacity, but agreed that the contribution of higher modes should be addressed, and will revise the calculation. (f) Control Rod Drive Mechanism fragility evaluation assumed that material stresses were the critical failure mode, and did not address the potential impact of deflections on rod drop. SNC indicated that information provided by Westinghouse (based on a		

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
			Japanese testing program) indicated that the deflection levels associated with seismic loading does not impact rod drop, and agree to add this discussion to the calculation.		
SFR-E4, SPR-B9	14-20	Seismic induced fire evaluations are not documented in the walkdown report or fragility calculations. (This F&O originated from SR SFR-E4)	The only mention for seismic induced fire evaluation is contained in the quantification notebook. Based on discussions during the peer review, it is understood that seismic induced fire was a key consideration during the walkdowns. However, detail of the walkdown procedure for fire following earthquake is missing. The write up should include team composition, methodology, screening criteria, and results,	Seismic induced fire is an important element of the fragility evaluation process and this should be clearly documented.	The seismic-induced fire and flood evaluations have been updated, and documented in the fragility and quantification report. This includes the details of the walkdown procedure used to evaluate the potential for seismically induced fires, including the methodology, screening criteria and results. This finding has been resolved with no significant impact to the SPRA results or conclusions.

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
SFR-D1	14-4	A potential for sloshing induced inundation of the NSCW Pumps (11202P4007, 11202P408) and associated discharge motor operated valves (1HV11600, 11606, 11607, 11613) in the NSCW exists and was not identified either in the walkdowns or subsequent analysis. (This F&O originated from SR SFR-D1)	SFR-D1 requires that realistic failure modes of structures and equipment that interfere with the operation of that equipment be identified. The potential for earthquake induced sloshing of the water within the NSCW tower exists. From field walkdowns of the NSCW it was observed that there is a potential for sloshing of contents to potentially splash onto or flood the pumps and or motor operated valves on the attached discharge piping.	Evaluate the potential for flood induced failure of the NSCW Pumps or NSCW discharge MOVs.	The evaluation for potential flood induced failure of the NSCW pumps or the NSCW discharge MOVs has been performed and documented in the fragility calculation for the NSCW tower. There was no significant impact on the pump or MOV fragilities. This finding has been resolved with no significant impact to the SPRA results or conclusions.
SFR-D1	14-5	The potential for seismically-induced differential settlements between structures was not addressed. (This F&O originated from SR SFR-D1)	Vogtle 1&2 is a soil site, with engineered fill from the rock interface to the finished grade. The in-scope Seismic Category I structures have foundations with varying embedment depths, ranging from surface founded (elev. 220 ft.) to a foundation embedment of 110 ft. (elev. 110 ft.). Since soils, including engineered fill, will consolidate/settle to some extent when subjected to high level earthquake ground motion, and the amount of settlement is proportional to the thickness of	Develop estimates of the differential settlements between adjacent structures and assess the fragility of commodities based on their ability to accommodate the associated differential displacements.	Documentation has been updated to include the effects of earthquake induced settlement; no significant differential settlements were computed between the structures. This finding has been resolved with no significant impact to the SPRA results or conclusions.

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
			the soil layer under the foundation, the settlement of one structure relative to another structure is dependent on the depth of the foundation embedment. The Fragility Notebook (PRA-BC-V-14-025) does not address the potential differential settlement between buildings, or the potential effect on commodities (e.g., piping, electrical raceways, HVAC ducts, etc.) that cross the separation between adjacent structures. During the performance of the Peer Review, SNC personnel indicated that the consideration of differential settlements was not required, since the structures were founded on engineered fill.		
SFR-G2	14-6	The results of the seismic gap/shake space walkdowns are not documented. (This F&O originated from SR SFR-G2)	The walkdown guidance provided in Appendix F (Checklists and Walkdown Data Sheets) of EPRI NP-6041 includes attributes of seismic gaps between structures which should be addressed in the performance of the walkdowns. These include the clearance between adjacent structures and the ability of any subsystems (e.g., piping, cable trays, HVAC ducts) spanning the gap to	Provide documentation of the results of the seismic gap walkdowns.	As noted in the Finding basis, inspection of the seismic gaps was included in the seismic walkdowns. Piping across seismic gaps is designed with adequate flexibility to accommodate building motions, and pipe sleeves provide adequate gaps for piping

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
			accommodate the differential seismic displacements. The Seismic Walkdown Report (PRA-BC-V-14-005) does not include documentation of the results/findings/observations associated with the inspection of the seismic gaps between structures or the subsystems spanning the gap. During the performance of the Peer Review, SNC personal indicated that inspection of the seismic gaps was included in the seismic walkdowns, but not explicitly described in the report. The ability of components to accommodate potential differential movement at the building separations is implied in the discussion of rugged components (piping, cable trays, and HVAC ducts) in Section 2.1 (Rationale for Screening) of the report. In addition, information from the Vogtle IPEEE Report (page 3.1-37) indicated that the seismic gaps had been inspected during the IPEEE.		movement. The documentation has been updated to reflect the inspections performed during the walkdowns. This finding has been resolved with no significant impact to the SPRA results or conclusions.

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
SFR-A2, SFR-F4	14-7	The fragility evaluation for the Containment Polar Crane (in fragility notebook) did not address the impact of variation in the fundamental frequency on the applicable seismic demand. (This F&O originated from SR SFR-A2)	The determination of the fundamental frequency of structures and components involves a certain degree of uncertainty. This uncertainty must be accounted for in the determination of the seismic accelerations from the applicable in-structure response spectra (ISRS). Section 7.4 (Vogtle 1 and 2 Polar Crane) of the Fragility Notebook (SNC calculation no. PRA-BC-V-14-025) evaluates the polar crane as a potential seismic interaction source relative to the reactor vessel and other NSSS components inside the containment structure. In the determination of the vertical spectral acceleration applicable to the polar crane, the computed fundamental frequency falls within a valley in the applicable ISRS, on the low frequency side of the primary spectral peak. Uncertainty in the calculated frequency, and the contribution of high modes, could result in an increase in the applied vertical acceleration. During the performance of the Peer Review, SNC personnel provided a written response indicating that it is	Update the fragility evaluation for the polar crane to address potential uncertainty in the fundamental frequency and the contribution of higher modes.	The fragility evaluation of the polar crane has been updated to address potential uncertainty in the fundamental frequency and contribution of higher modes. This finding has been resolved.

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
			appropriate to increase the applied acceleration by 50%, which will result in a 20% decrease in the median capacity of the polar crane.		
SFR-F3	14-8	Relay fragility calculations include conservative assumptions. (This F&O originated from SR SFR-F3)	The relay evaluation for the turbine driven auxiliary feedwater pump control panel in calculation PRA-BC-V-14-008 is based on a generic capacity for motor starters and contactors (intended for motor control centers) and an amplification factor associated with center of door panel response. Based on walkdown observations the relay is not mounted on the door panel so is likely on an internal bracket. The median capacity of 0.627g is well below the screening level and is not realistic. The relay evaluations in calculation PRA-BC-V-14-009 are governed by response in the vertical direction, and the in-cabinet amplification factors used in the calculation are associated with horizontal response. The resulting median capacities of 0.762g (Appendix M1) and 1.026g (Appendix M2) are well below the screening level and are not realistic.	Perform more realistic relay fragility evaluations.	The relay fragilities have been updated using the appropriate response and in-cabinet amplification factors, and are realistic. This finding has been resolved.

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
SFR-D2	14-9	The seismic walkdown report includes a number of open items that are not are not traceable to a resolution (This F&O originated from SR SFR-D2)	The summary of the seismic walkdowns documents a number of issues identified during the performance of the walkdowns that required follow-up actions (31). These include spatial interaction issues, housekeeping issues, anchorage issues, valves having configurations that do not meet the EPRI guidelines, configuration issues, installation errors, etc. The Seismic Walkdown Report (PRA-BC-V-14-005) does not document how the issues identified during the walkdowns have been addressed, either in the field (e.g., correction of installation errors, resolution of housekeeping issues) or in the fragility evaluations (e.g., valve configurations, anchorage issues). During the performance of the Peer Review, the Peer Review Team provided a list of the walkdown issues to SNC personnel, and SNC provided a summary of how they were addressed. Most issues had been adequately addressed during the development of the SPRA, but it was determined that the following would require further effort for resolution:	Perform resolution of open items and provide documentation of the resolution associated with each of the issues, either in the Fragility Notebook or the SPRA Database.	The noted walkdown issues have been evaluated and reflected in the revised documentation: - potential piping interaction; - the difference in inverter anchorage configuration; - potential interaction concerns with the overhead heater; this evaluation is in the fragility notebook in section 3.4.2. Valve operator heights & weights that were outside EPRI guidelines have been taken into account in the fragility analysis for these components. The Diesel Generator Exhaust Silencer was re-evaluated to the as-operated condition. The fragility analysis for these components has been completed for the as built condition.

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
			(a) Potential interaction between piping and deluge valve (page 19) - follow-up walkdowns required. (b) Anchorage configuration on inverter (page 40) - follow-up revision to fragility evaluation required (c) Overhead heater poses potential interaction issue (page 60) - follow-up walkdown required. (d) Valve operator heights/weights outside of EPRI guidelines (page 74) - follow-up walkdown required. (e) Diesel Generator exhaust silencer anchor bolt nuts (page 96) - not addressed in fragility evaluation, further evaluation required. (f) Valve operator heights outside of EPRI guidelines and potential lack of yoke support (page 105) - these valves are part of the unfinished scope described in the Fragility Notebook, which will be completed in the future. (g) Valve operator heights outside of EPRI guidelines (page 107) - further evaluation required.		This finding has been resolved.
SFR-F3, SPR-B4, SPR-E5	16-1	The model presented for peer review did not incorporate the effects of relay chatter as the analysis was not yet	Relay chatter is consistently being observed as a significant contributor to risk profile in recently peer reviewed S-PRAs	Complete the analysis and incorporate the effects of relay	The approach to screening and modeling of seismically-induced relay failures and chatter

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
		complete. (This F&O originated from SR SPR-B4)	and it is therefore realistic to expect that relay chatter is a potential significant contributor. During the peer review it was discussed that the SPRA team does not believe relays will be a significant contributors but it was also said that this conclusion/ expectation is based on potentially crediting operator actions. Thus, the effects of relay chatter per se may be significant (and provide some insights) while the combination of relays and a number of HEP may not be.	chatter and similar devices in the PRA logic model.	was provided to the peer review team and determined to have been performed appropriately; only the incorporation into the model of the impacts of relay chatter from unscreened relays was not complete. The final screening resulted in only 2 relays being incorporated into the model, with one having an operator action. Relay chatter fragilities and impacts have been incorporated into the seismic model, in a manner consistent with that used for other failures. This finding has been resolved.
SPR-B6	16-10	The documentation about the walkdowns in support to seismic impact on HRA appear limited. (This F&O originated from SR SPR-B6)	There is only a short sentence supporting the discussion on alternative access pathways.	More detailed documentation is suggested to support the conclusion on accessibility, alternative route, availability of tools/keys, clear identification of equipment manipulated in each local action. Obviously, the goal of the enhanced documentation is	Walkdown documentation on accessibility for operator actions, including photos, has been improved. Potential failure of block walls has been reviewed and documented. Required tools and equipment,

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
				not to convince the peer reviewer that the walkdowns were performed but rather to ensure that the analyst is fully convinced of the conclusions. Past SPRAs have shown examples of equipment needed for the HFE that was not in the SEL, or that has different actuators when manually actuated, or that needed ladders that were not easily accessible or that were close to block walls (or under ceiling that could collapse) that were not considered an issue because the block walls were not near safety related equipment (and therefore not addressed in the rest of the SPRA work). In this perspective, a more systematic documentation of the feasibility and accessibility analysis for each of the HFE credited in the SPRA is suggested.	such as ladders, have been identified with locations when needed. The documentation supports the seismic HRA assumptions and modeling. This finding has been resolved with no significant impact to the SPRA results or conclusions.

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
SPR-E2	16-11	Missing review of the potential for additional dependencies introduced by the SPRA models (QU-C1&2) (This F&O originated from SR SPR-E2)	It is understood that the investigation performed in internal events to identify potential HFE dependency has been relied upon in the Vogtle SPRA. The SPRA logic may identify additional dependencies trends that were not identified in the internal events.	As this exercise was apparently performed for the Fire PRA (as discussed during the peer review), it is suggested that a review of the potential for unforeseen dependencies trends is performed. As it is understood that the plan is to transition to a different dependency analysis method (based on HRA calculator), this may be addressed within the same transition as it is realistic to expect that not too many (if any) new dependencies would be identified.	A detailed quantitative HRA dependency analysis based on using the HRA calculator was performed and documented. There was no significant impact on results since human actions are not significant contributors in the Vogtle SPRA. This finding has been resolved.
SPR-E2	16-12	Missing documentation of the review of non significant cutsets QU-D5. (This F&O originated from SR SPR-E2)	It is an industry expectation (as discussed in NEI peer review task force meetings) that review of the non significant cutsets is explicitly documented. Based on discussion during the peer review, two reviews were performed to validate the overall model and cutsets. The first was a random review of cutsets at midpoints and low significance for each of the %Gxx initiators to verify that the cutsets are valid cutsets, and that the patterns are	It is understood that the SPRA documentation will be revised to incorporate explicitly the two reviews discussed in the basis for this F&O. It is also recommended to document the review of cutsets following guidance from the NEI peer review task force.	The QU report has been updated to document the review of both dominant cutsets and non-significant cutsets for both CDF and LERF. This finding has been resolved with no significant impact to the SPRA results or conclusions.

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
			appropriate. That is, if one cutset is valid, then another cutset with slightly different seismic failures (or random failures) should also be nearby. The second review, more importantly, lowered the median seismic capacity for each of the seismic initiators and some of the other seismic failures to ensure that the model would properly generate valid cutsets. For example, the LLOCA fragility was reduced to 0.5g to generate LLOCA cutsets. For ATWT, the fragility of the CRDs and RV internals were reduced to 0.5g to verify that valid ATWT cutsets were generated.		
SPR-E6, SPR-F2	16-15	Documentation of LERF model applicability review. (This F&O originated from SR SPR-F2)	The current documentation does not explain what are the basis for retaining the LERF logic and analysis unchanged within the SPRA logic. During the peer review the following explanation was provided by the SPRA team: "The internal events Level 2 notebook (Chapter 9) was reviewed to ensure that the definition of LERF would be	Expand the documentation to ensure that the criteria used to retain the LERF analysis in the SPRA is explained so that the same applicability review can be performed following future potential revisions of the LERF modeling.	The LERF documentation in the QU report was expanded to describe the review of applicability of the internal events PRA LERF analysis to the seismic PRA. This finding has been resolved with no significant impact to the

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
			appropriate for seismic events. Section 9.2 provides the LERF definition, including the use of a 12 hour time period for release after event initiation, to allow for evacuation. This time period is considered to be valid for Vogtle seismic events, particularly due to the very low population density in the area. Other characteristics, such as bypass and scrubbing, are the same for seismic as for internal events. The logic for the internal events LERF model is very straightforward, with sequences from the CDF model ANDed with the appropriate LERF fault tree. This logic is also appropriate for seismic events."		SPRA results or conclusions.
SPR-B8	16-18	Very small LOCA have been screened from the analysis based on walkdowns but little documentation exists of such walkdowns. (This F&O originated from SR SPR-B8)	The DB has a specific entry for the incore thermocouples and provides pictures of them. Still, in-core thermocouple tubing is not the only possible source of very small LOCA that is envisioned and the only documentation of addressing the other potential sources is in section 2.3.3 of the quantification notebook: "For Vogtle 1&2, the seismic walkdowns inspected and	To the peer review team knowledge Vogtle is the only plant that has elected to perform dedicated walkdowns in support of not modeling very small LOCA. This would be a best practice but it also behooves to the SPRA team to provide detailed documentation of such walkdowns and how they supported a systematic	Additional information on the walkdown for very small LOCA has been added to fragility report to provide the basis for the VSLOCA screening. This finding has been resolved with no significant impact to the SPRA results or

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
			photographed a large sample of the small piping and tubing lines connected to the primary system in order to identify any weaknesses. The piping was judged to be rugged."	evaluation of the potential sources of very small LOCA.	conclusions.
SFR-C1, SPR-E1	16-2	Fragilities were not corrected to reflect the 2014 hazard used for quantification. (This F&O originated from SR SPR-E1)	The 2014 hazard was only used as input to FRANX for the final quantification. It is understood that the fragility estimates have been performed based on the 2012 hazard. While it is not expected nor recommended to regenerate all the fragility work with the new hazard, some consideration on the possible change in fragility due to the use of the newer hazard should be made.	During the peer review the SNC staff answered a question on this topic by performing an initial limited investigation of the effect on fragilities correction to reflect the 2014 hazard and concluded that the effect of this scaling is not insignificant (especially for LERF). It is recommended to continue and expand this investigation to make the quantification fully consistent with the fragility values.	The fragilities have been recalculated based on the 2014 hazard [3] and the new values incorporated into the SPRA model and quantification. This finding has been resolved.
SPR-B2	16-4	The effect of seismic impact on performance shaping factors is considered in the analysis by the usage of the Surry method. (This F&O originated from SR SPR-B2)	There is no assessment of the effect of changing the breaking points in the Surry method. The Surry method is based on methods used in the past at SONGS and Diablo Canyon and the 0.8g breaking point was developed for California earthquakes. In the Vogtle analysis there is no indications on whether the breaking point at 0.8g is also applicable to Vogtle. There	While it is recognized that the industry is still developing methods in support to this particular topic (e.g., recently published EPRI HRA method for external events), some additional considerations should be done to understand the effect of HEPs in the model rather than simply implementing the Surry method as is.	The methodology used for the seismic HRA analysis is based on defining PSFs as a function of seismic hazard level (bins), which is consistent with the EPRI seismic HRA guidance in EPRI 3002008093 [9]. The Integrated PSFs and bins (breaking points)

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
			are also no sensitivity analyses that would support whether a change in the breaking points is significant or not.	Three examples for addressing this finding may be the following: 1. Perform sensitivities on the values of the multipliers and the g levels where the breaking point happens. 2. Use a different multipliers method with more breaking points. 3. Apply the impact of seismic specific PSF at the individual PSF level (i.e., timing, stress, etc.) in the HRA calculator.	have been updated with additional breaking points and integrated PSFs to reflect seismic binning applicable to Vogtle, in accordance with this finding and consistent with the EPRI guidance. The updated values have been applied to both internal events HFEs and seismic-unique HFEs within the plant response model. There was no significant impact on the SPRA results. This finding has been resolved.
SPR-B1, SPR-F1	16-5	LOCA modeling and fragility selection not clearly documented. (This F&O originated from SR SPR-F1)	The selection of the fragility data used for all LOCA is discussed in Appendix B.2 of the quantification notebook but is confusing in the mapping of selected fragilities with specific failures. It appears that the fragility selected to represent LOCA sequences are coming from specific components but then they are used to represents sort of	Documentation on the use of fragility in support to LOCA should be clarified to better represent the rationale selected and potentially addresses the modeling uncertainties associated with this selection. While this finding is expected to be addressed via documentation, some	LOCA basis has been re-evaluated and updated. This was partially due to seismic fragility update and partially a matter of adding amplifying information to the LOCA basis. The quantification report includes updated documentation. Although LOCAs are a significant

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
			surrogate events for potential failures along the piping network. Using localized events as surrogate for pipe network failure is probably conservative and may not be fully consistent with the system success criteria and modeling in the internal events modeling. For example, the seismic-induced MLOCA fragility seems to be based on failure of the pressurizer surge line, which is a localized failure. The seismic-induced MLOCA initiator is mapped to the internal events MLOCA initiator. The internal events logic for MLOCA has a split fraction that divides MLOCA (and LLOCA) in four 25% contributors impacting all four CL/HL. Since the seismic-induced MLOCA is a localized failure, the internal events logic is not fully applicable (probably slightly conservative). Because the documentation is potentially leading to a misunderstanding of the selected approach (thus impacting ease on update), this F&O is considered a finding against the documentation SR.	additional suggestions are provided, such as: 1. Perform a sensitivity to show that the modeling approach described is not significantly skew the results for seismic; 2. Modify the logic by mapping the seismic-induced MLOCA to a different position in the logic (e.g., a dummy event can be entered in the model to provide a target for the FRANX injection).	contributor to the SPRA results, the VEGP SCDF and SLERF are sufficiently small that further LOCA modeling sensitivity beyond what has been provided in the updated model quantification is not warranted. This finding has been resolved.

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
SPR-B2	16-6	The effect of seismic impact on performance shaping factors is not considered for any action that was explicitly added for the SPRA (e.g., flood isolation or DG output breaker closure). (This F&O originated from SR SPR-B2)	The Vogtle SPRA elected to use Integrated Performance Shaping Factors (IPSF) multipliers. While this approach was used for the HEPs that were carried over from internal events, it was systematically not done for all the actions explicitly added for seismic. Based on discussion during the peer review, the analyst believed that having designed these actions for specific scenarios following a seismic event, the impact of seismic specific PSF is already included. The objection to this conclusion is that the seismic specific PSF should realistically change with the magnitude of the event. This change addresses the change in the overall context of the plant when a small seismic event happens as opposed to when a very large seismic event happens. This seems not to be captured by the approach selected for the Vogtle SPRA. One example of this is that an action that has a 30 minute Tsw (S-OA-BKR-LOCAL) maintains an HEP of 1.60E-03 at all g levels, including the %G14 interval (i.e., >2g).	Expand the IPSF approach to all the operator actions credited in the SPRA.	The methodology used for the seismic HRA analysis is based on defining PSFs as a function of seismic hazard level (bins), which is consistent with the EPRI seismic HRA guidance in EPRI 3002008093 [9]. The Integrated PSFs and bins (breaking points) have been updated to reflect seismic binning applicable to Vogtle, in accordance with this finding and consistent with the EPRI guidance. The updated values have been applied to both internal events HFEs and seismic-unique HFEs within the plant response model. There was no significant impact on the SPRA results. This finding has been resolved.

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			It is understood that this is not expected to be quantitatively significant because failure of the recovered equipment is taken care by the logic model.		
SPR-E2	16-7	Base case seismic LERF does not meet the truncation requirements from QU-B3. (This F&O originated from SR SPR-E2)	Both CDF and LERF are truncated at 1.0E-09 with 1000 cutsets managed by ACUBE. This meets the QU-B3 requirement for CDF but not for LERF.	LERF at 1E-11 truncation meets the QU-B3 truncation requirement. Rename LERF at 1E-11 as the base case for LERF.	LERF truncation, which was already considered in sensitivity studies, has been revised appropriately to meet QU-B3. A new LERF truncation limit has been established consistent with the LERF results. Quantification is at 1E-12, which is a suitably low value. This finding has been resolved.
SPR-E2	16-8	Missing documentation of cutsets review (cfr. QU-D1) (This F&O originated from SR SPR-E2)	Section 3.1 is the only description of the most important scenarios but there is no cutset-by-cutset review.	While it is understood that the Draft. B version of the quantification notebook is still somewhat a work in process, it is expected that when the model reaches a more stable state documentation of the review of the cutsets is going to be part of the documentation.	The QU report has been updated to document the review of both dominant cutsets and non-significant cutsets for both CDF and LERF. This finding has been resolved with no significant impact to the SPRA results or conclusions.

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
SPR-B1, SPR-B4b	16-9	Screening values used for the HEPs that (at the time of the provided documentation) were in the most significant cutsets. (This F&O originated from SR SPR-B1)	At the time when the documentation was provided for peer review, the most significant operator actions (i.e., flood isolation of ACCW HX) were all screening values, which would only meet CCI for HR-G1 (directly called through SPR-B1). In addition, there is little documentation or supporting evidence to justify screening values as low as 3.00E-2	An appropriate resolution of this F&O is pending the current evolution of the model and the importance of operator actions in the SPRA. Given the expectation that operator actions will be needed to mitigate the importance of relay chatter (not yet included in the SPRA logic model) this F&O was provided to ensure care is used in the generation of HEPs if they appear in important cutsets and also to provide more justification for screening values less than 1.00E-1 because a low screening value may indeed skew the actual importance of the newly generated HEP.	The seismic HRA analysis has been revised to be consistent with the EPRI seismic HRA guidance in EPRI 3002008093 [9]. The original screening HEPs have been updated using the HRA Calculator, consistent with the approach used in the VEGP internal events PRA. The Documentation has been updated. Operator response to relay chatter has been addressed and evaluated within the same process, and not found to be important. This finding has been resolved.
SPR-B1	17-1	The documentation does not specifically address the applicability of the internal events accident sequences and success criteria to the SPRA model, and does not properly document the accident sequences created specifically for the SPRA model.	The modeling approach injected seismic fragilities into fault trees that were modified from the internal events PRA model. It can be inferred from this approach, and it was verified by discussions with the staff, that the internal events sequences and success criteria were considered to be applicable to the SPRA model. This was not specifically stated in	A separate section in the documentation that specifically addresses accident sequences and success criteria is needed to collect the information in one logical place, and is needed to support effective peer reviews and future model updates.	The discussion of accident sequences and success criteria has been expanded, and specific descriptions of the flooding scenarios has been added. This finding is documentation only and does not impact Seismic PRA model results.

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
		(This F&O originated from SR SPR-B1)	the documentation. Further, several additional seismic flooding sequences were added to the fault tree. These sequences are not discussed from an accident sequence and success criteria perspective. Inspection of the fault tree and discussions with the staff indicate that the sequences were appropriately developed with specific success criteria that is different from other internal events sequences. The development of these sequences needs to be included in the documentation. Including event trees for these sequences would also aid in a reader's understanding.		This finding has been resolved with no significant impact to the SPRA results or conclusions.
SPR-E2, SPR-F2	17-2	The processes used to create the presented quantification results are not fully documented. (This F&O originated from SR SPR-F2)	Examples include: The top cutsets shown in table 3-1 of the quantification report are produced by combining the cutsets from all the seismic interval cutsets in a process that is not documented. While the process used to obtain the importance measures in section 5.2 of the quantification notebook is documented in that	Expand the documentation to clearly explain the post-processing of the results generated by CAFTA and FRANX. Examples include: - Explain how the cutsets generated by FRANX are combined into g-level-independent cutsets. - Explain the post-processing used to generate importance	Documentation for QU results has been im-proved to describe the processes used to ag-gregate results over the 14 hazard intervals. The importance calculations have been re-quantified and the method for presentation documented. This finding has been

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
			section, discussions with the PRA staff indicated that importances for some of the basic events were obtained in a different manner (setting to one or zero and requantifying). This is not documented in the notebook.	measures, especially focusing on the deviation from a normal practice that is currently only mentioned in the notebook.	resolved with no significant impact to the SPRA results or conclusions.
SPR-B3, SPR-E4	17-3	Subdividing correlation groups based on weaker/stronger components resulted in retention of non-minimal cutsets in some cases, which could impact CDF/LERF results as well as model importance measures. The magnitude and acceptability of these impacts was not documented. (This F&O originated from SR SPR-E4)	To account for similar equipment that has different fragilities due to different building locations, certain correlation groups were subdivided to assign a seismic capacity to a weaker component that only failed that component. The higher capacity was then assigned to both components, and was effectively the correlated failure of both components. This can result in the retention of non-minimal cutsets in some cases. For example, for the Containment Fan Cooler Units there are cutsets in which, due to other failures, only one containment fan cooler needs to seismically fail to cause core damage. Inspection of the cutsets shows that two otherwise identical cutsets are retained: one in which the 1Fan 'group' occurs, and one in which the 4Fans group occurs. The 4Fans cutset is not minimal, and should not be included in the results. Discussions with the staff	The impact of the retention of these non-minimal cutsets on CDF/LERF and importance measures should be assessed and the results documented, or a method to remove the non-minimal cutsets should be devised. Each subdivided correlation group should be investigated for similar effects.	The non-minimal cutsets in the peer reviewed model were identified and reviewed for impact, and determined to be non-significant to risk. The results were very slightly conservative due to these non-minimal cutsets. The issue has been addressed in the updated model, such non-minimal cutsets no longer appear. This finding has been resolved with no significant impact to the SPRA results or conclusions.

Supporting Require-ment(s)	Finding Number	Finding Description	Finding Basis	Suggested Finding Resolution	Disposition
			indicated that these non minimal cutsets were noted during the quantification review process, but were thought to not greatly impact overall results. No formal assessment was done, however, and no record of the informal assessment was included in the documentation.		
SPR-E6	17-4	No quantitative analysis of the relative contribution to LERF from Plant Damage States and Significant LERF contributors from Table 2-2.8-9 was presented in the quantification results. (This F&O originated from SR SPR-E6)	A quantitative analysis is required to meet CCII for LE-F1 & LE-G3, which are directly called from SPR-E6.	Perform the analysis and include the results in the quantification notebook.	The quantitative analysis of significant LERF plant damage states and contributors has been performed. A table and associated discussion of plant damage states and significant contributors has been added to the LERF QU documentation to resolve this finding. This finding has been resolved.