VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

March 28, 2018

U.S. Nuclear Regulatory Commission
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Washington, DC 20555

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License Nos.:	NPF-4/7

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNITS 1 AND 2
RESPONSE TO MARCH 12, 2012 INFORMATION REQUEST
SEISMIC PROBABILISTIC RISK ASSESSMENT FOR
RECOMMENDATION 2.1

References:

1. NRC Letter, "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," dated March 12, 2012 [ADAMS Accession Nos. ML12056A046 and ML12053A340].
2. EPRI Report 1025287, "Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic." [ADAMS Accession No. ML 12333A170].
3. Virginia Electric and Power Company Letter to NRC, "North Anna Power Station Units 1 and 2 Response to March 12, 2012 Information Request – Seismic Hazard and Screening Report (CEUS Sites) for Recommendation 2.1," dated March 31, 2014 [ADAMS Accession No. ML14092A416].
4. NRC Letter, "North Anna Power Station, Units 1 and 2 - Staff Assessment of Information Provided Pursuant to Title 10 of the Code of Federal Regulations Part 50, Section 50.54(f), Seismic Hazard Reevaluations Relating to Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dal-Ichi Accident (TAC Nos. MF3797 and MF3798)," dated April 20, 2015 [ADAMS Accession No. ML15057A249].
5. NRC Letter, ""Final Determination of Licensee Seismic Probabilistic Risk Assessments Under the Request for Information Pursuant to Title 10 of the *Code of Federal Regulations* 50.54(f) Regarding Recommendation 2.1 "Seismic" of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," dated October 27, 2015 [ADAMS Accession No. ML15194A015].

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.

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Reference 2 contains industry guidance developed by EPRI that provides 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 North Anna Power Station (NAPS), Units 1 and 2 Seismic Hazard and Screening Report (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.

Reference 4 contains the NRC Staff Assessment of the NAPS Units 1 and 2 seismic hazard submittal and concluded that the reevaluated seismic hazard prepared for NAPS 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 1 a - Recommendation 2.1 Seismic – Information Requests), the NRC instructed that a Seismic Probabilistic Risk Assessment (SPRA) be submitted for NAPS Units 1 and 2 by March 31, 2018.

The Attachment to this letter contains the NAPS Units 1 and 2 SPRA Summary Report, which provides the information requested in Enclosure 1, Item (8)B of Reference 1.

If you have any questions regarding this information, please contact Diane E. Aitken at (804) 273-2694.

Sincerely,

Daniel G. Stoddard
Senior Vice President and Chief Nuclear Officer
Virginia Electric and Power Company

DIANE E. AITKEN
NOTARY PUBLIC
REG. #7763114
COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES MARCH 31, 2022

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Daniel G. Stoddard, who is Senior Vice President and Chief Nuclear Officer of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 28 day of March, 2018.

My Commission Expires: March 31, 2022

Diane E. Aitken
Notary Public

Commitments made in this letter: No new regulatory commitments.

Attachment: North Anna Power Station Units 1 and 2 Seismic Probabilistic Risk Assessment in Response to 10 CFR 50.54(f) Letter with Regard to NTTF 2.1 Seismic – Summary Report

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ATTACHMENT

North Anna Power Station Units 1 and 2
Seismic Probabilistic Risk Assessment in Response to 10 CFR 50.54(f) Letter with Regard to NTTF 2.1 Seismic

Summary Report

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNITS 1 AND 2

North Anna Power Station Units 1 and 2 Seismic Probabilistic Risk Assessment in Response to 10 CFR 50.54(f) Letter with Regard to NTTF 2.1 Seismic

Summary Report

March 2018

North Anna Power Station Units 1 and 2

Seismic Probabilistic Risk Assessment

Summary Report

Contents

EXECUTIVE SUMMARY

In response to the NRC 10 CFR 50.54(f) letter of March 12, 2012, a seismic probabilistic risk assessment (SPRA) was performed for North Anna Power Station (NAPS) Units 1 and 2. The SPRA effort included performing a probabilistic seismic hazard analysis (PSHA) to develop seismic hazard and response spectra at the plant using the state-of-the-art seismic source model and attenuation equations; site response analyses; dynamic analyses of structures; fragility analyses of structures, systems and components (SSCs); developing a logic model; and performing risk quantification. Each element of the SPRA effort underwent an in-process independent expert review and a final peer review by a team of experts. The comments and suggestions of the reviewers were addressed and incorporated into the SPRA as applicable.

The SPRA identified risk-significant sequences and SSCs with their risk rankings, and showed that for both North Anna units, the seismic Core Damage Frequency (SCDF) is 6.0E-5 per year and the seismic Large Early Release Frequency (SLERF) is 1.6E-5 per year.

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 are robust with respect to the modeling and assumptions used.

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 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 NAPS 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 was 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 North Anna Power Station Units 1 and 2 (NAPS) 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 NAPS 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 NAPS, 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 NAPS seismic PRA.

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 and Fussel-Vesely)
(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 NAPS Seismic Hazard and Screening Report submittal [3]. Further, 50.54(f) letter Enclosure 1 paragraph 9 requests information on the Spent Fuel Pool, which was submitted separately [15].

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 NAPS SPRA has been developed and documented in accordance with the SPID. The main elements of the SPRA performed for NAPS in response to the 50.54(f) 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, other than those already listed in Table 2-1, and provides the location in this report where the corresponding information is discussed.

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

Section 4 provides information related to the determination of seismic fragilities for NAPS 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 NAPS 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 LERF	Sections 3, 4, 5
(2)i	Methodologies used to quantify the seismic fragilities of SSCs, together with key assumptions	Section 4
(2)ii	SSC fragility values with reference to the method of seismic qualification, the dominant failure mode(s), and the source of information	Tables 5.4-2 and 5.5-2 provide fragilities (median acceleration capacity [Am] and aleatory [βr] and epistemic variability [βu]), failure mode information, and method of determining fragilities for the top risk significant SSCs based on standard importance measures such as Fussell-Vesely (FV).
(2)iii	Seismic fragility parameters	Tables 5.4-2 and 5.5-2 provide fragilities (Am, βr, βu) information for the top risk significant SSCs based on standard importance measures such as FV.
(2)iv	Important findings from plant walkdowns and any corrective actions taken	Section 4.2 addresses walkdowns and walkdown insights.
(2)v	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 5.2 provide this information.

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

50.54(f) Letter Reporting Item	Description	Location in this Report
(2)vi	Assumptions about containment performance	Sections 4.3 and 5.5 address containment and related SSC performance
(3)	Description of the process used to ensure that the SPRA is technically adequate, including the dates and findings of any peer reviews	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.

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

SPID Section 6.8 Item (1) Description	Location in this Report
A report should be submitted to the NRC summarizing the SPRA inputs, methods, and results.	Entirety of the submittal addresses this.
The level of detail needed in the submittal should be sufficient to enable NRC to understand and determine the validity of all input data and calculation models used	Entirety of the submittal addresses this. The summary report 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 the following sections: 5.7 (SPRA model sensitivities) 4.4 Fragility screening (sensitivity)
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 report 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 NAPS 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.

North Anna Power Station is a dual unit Westinghouse 3-loop pressurized water reactor plant located on a peninsula on the southern shore of Lake Anna, approximately 45 miles northwest of Richmond, Virginia. The reactor buildings are founded on competent bedrock; other principal structures are founded on weathered bedrock or on structural fill overlying bedrock. The bedrock has been weathered unevenly into saprolitic soils of varying thickness, ranging from a few feet to as much as 100 ft below original grade. Detailed studies carried out during the siting investigation for North Anna Units 1 and 2, and more recently for the proposed North Anna Unit 3, show that there are no capable faults within the site vicinity. Additional site description and composite profile development are described in the NAPS NTTF 2.1 Seismic Hazard and Screening Report 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 NAPS site hazard was provided to NRC in the seismic hazard information submitted in response to the NTTF 2.1 Seismic information request [3]. That information was used in development of the NAPS SPRA.

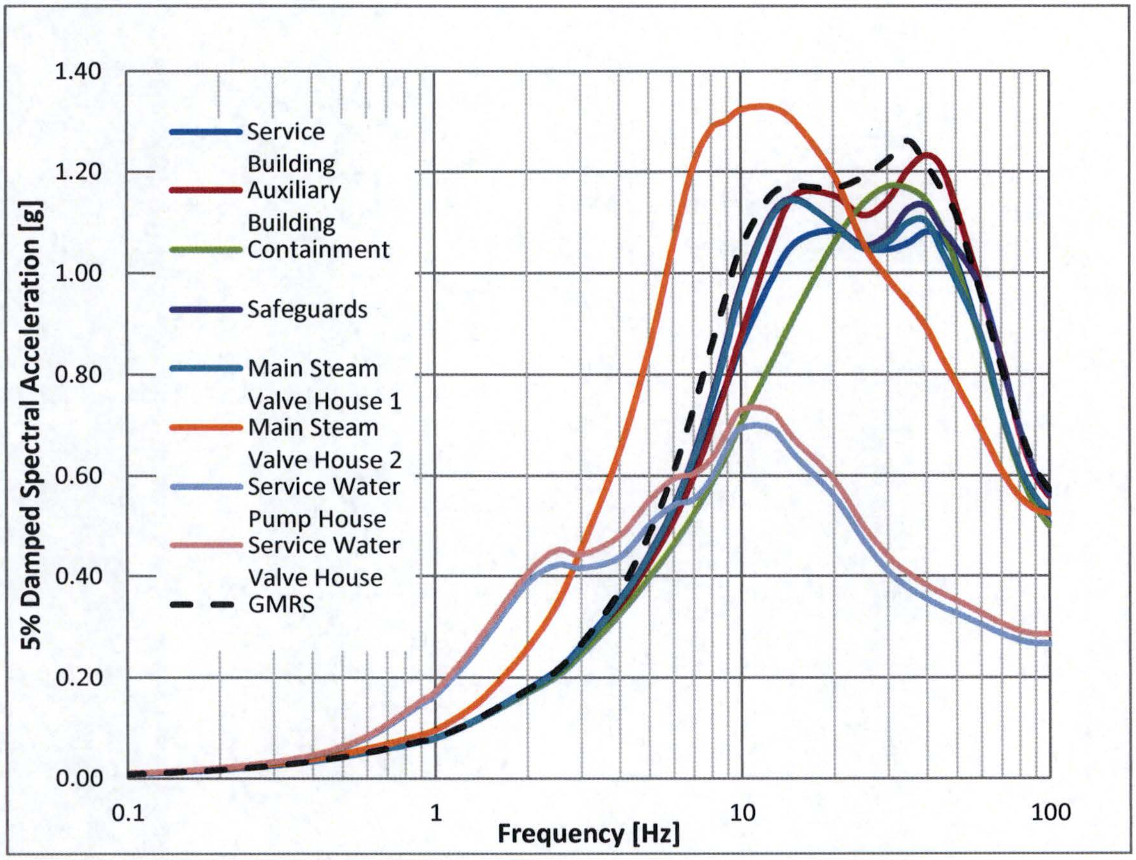
3.1.1 Seismic Hazard Analysis Methodology

The seismic hazard was developed for the NAPS SPRA as described in the NAPS NTTF 2.1 Seismic Hazard and Screening Report submittal [3]. A GMRS was developed from the uniform hazard response spectra (UHRS), which are based on hard-rock ground motions determined as part of the probabilistic seismic hazard analysis (PSHA) and the site response analysis, at the control point defined in accordance with the SPID [2]. The control point for NAPS is defined as the foundation bearing elevation of the highest rock-supported, safety-related structure, which corresponds to the Casing Cooling Tank and Pumphouse structure.

The reference earthquake used in developing building response, fragility evaluations, and risk quantification corresponds to the GMRS at the control point. The GMRS has a peak ground acceleration (PGA) of 0.572g.

Horizontal foundation input response spectra (FIRS) were developed as input to the dynamic analyses of structures that are not founded on grade (shown in Figure 3-1). The calculation of the horizontal FIRS is consistent with the methodology used to develop the GMRS, including development of soil column profiles, site amplification functions, and UHRS. As applicable, soil properties that are strain compatible with the FIRS are developed consistent with the approach suggested by SPID. The FIRS are directly used in the probabilistic SSI analysis of the Containment Building, Service Building, Auxiliary Building, and Main Steam Valve House Unit 2. The FIRS are further adjusted to generate SSI input response spectra which are suitable for deterministic analysis (per requirements of ISG-17) and used in the SSI analysis of the Service Water Pump House and Service Water Valve House and in fixed-base analyses of the Main Steam Valve House Unit 1 and Safeguards Buildings. The development of vertical FIRS is described in Section 3.1.4.

Figure 3-1: Horizontal GMRS and FIRS / SSI Input



3.1.2 Seismic Hazard Analysis Technical Adequacy

The NAPS SPRA hazard methodology and analysis associated with the horizontal GMRS were submitted to the NRC as part of the NAPS Seismic Hazard Submittal [3], and found to be technically acceptable by NRC for application to the NAPS SPRA [16].

The NAPS 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 SPRA applications [6].

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

Table 3-1 NAPS Mean and Fractile Exceedance Frequencies

	Exceedance Frequency (/yr)			
PGA (g)	**Mean**	**16th Fractile**	**50th Fractile**	**84th Fractile**
0.0530	1.07E-03	6.26E-04	1.03E-03	1.07E-03
0.0648	1.05E-03	4.73E-04	8.83E-04	1.07E-03
0.0717	9.96E-04	4.11E-04	7.82E-04	1.07E-03
0.0793	9.06E-04	3.58E-04	6.85E-04	1.07E-03
0.1019	6.51E-04	2.49E-04	4.80E-04	1.03E-03
0.1524	3.71E-04	1.39E-04	2.74E-04	6.34E-04
0.2061	2.42E-04	8.66E-05	1.77E-04	3.97E-04
0.3082	1.37E-04	4.24E-05	9.80E-05	2.13E-04
0.5097	5.70E-05	1.72E-05	3.80E-05	9.48E-05
0.7248	2.81E-05	8.64E-06	1.92E-05	4.59E-05
1.0306	1.39E-05	3.57E-06	9.37E-06	2.16E-05
1.5411	5.34E-06	1.25E-06	3.31E-06	8.88E-06
2.0840	2.39E-06	4.77E-07	1.48E-06	3.89E-06
2.5483	1.40E-06	2.31E-07	8.26E-07	2.25E-06
3.1162	7.83E-07	1.08E-07	4.25E-07	1.28E-06
3.6237	4.84E-07	5.52E-08	2.54E-07	8.11E-07
4.0071	3.48E-07	3.50E-08	1.78E-07	5.84E-07
5.1000	1.53E-07	1.00E-08	6.92E-08	2.51E-07

In the SPRA plant model, described in Section 5, the hazard data in Table 3-1 was discretized into 10 intervals, with parameters as listed in Table 3-2.

Table 3-2 Acceleration Intervals and Interval Frequencies as Used in SPRA Model

Interval Designator	Interval Lower Bound (g)	Interval Upper Bound (g)	Representative Magnitude PGA (g)	Interval Mean Frequency (/yr)
%G01	0.06	0.3	0.13	9.21E-04
%G02	0.3	0.4	0.35	5.34E-05
%G03	0.4	0.5	0.45	3.01E-05
%G04	0.5	0.6	0.55	1.79E-05
%G05	0.6	0.7	0.65	1.11E-05
%G06	0.7	0.8	0.75	7.08E-06
%G07	0.8	1.0	0.89	8.26E-06
%G08	1	1.5	1.22	9.09E-06
%G09	1.5	2.5	1.94	4.25E-06
%G10	2.5	5.1	2.75	1.48E-06

Uncertainties in the PSHA result from uncertainties in input models and parameters. These have been investigated for the NAPS SPRA. The composited seismic hazard includes Background seismic sources and individual repeated large magnitude earthquake (RLME) sources: Charleston, New Madrid Seismic Zone (NMSZ), Wabash Valley, and the northern segment of the Eastern Rift Margin fault. For 1 Hz spectral frequency at a mean annual frequency of exceedance (MAFE) of 1E-04, the Background seismic sources are dominant with Charleston and New Madrid together contributing about 15% of the total hazard. At lower MAFE levels, the Background sources become even more dominant. For 10 Hz spectral frequency a larger and almost complete contribution to the total hazard is from the Background seismic sources. The observation that high frequency is tending to be controlled by the background seismic sources and the low frequency having a larger contribution from the RLME seismic sources is commonly observed for sites in the CEUS. Sites located closer to a RLME would be expected to have a larger contribution from the RLME seismic sources, especially for the low frequency cases.

The ECC-AM seismic source, which is the background source zone in which the site is located, contributes the most individual hazard to the Background total with the combination of MESE-N and STUDY-R together contributing about the same hazard as ECC-AM. This observation is similar for both the 1 Hz and 10 Hz cases.

Finally, the last sensitivity is for the individual ground motion models used in the PSHA, which demonstrates the epistemic variation in seismic hazard among the ground motion models for the Background and RLME seismic sources for 1 Hz and 10 Hz spectral frequencies. At hazard levels of about 10-4 to 10-6, the epistemic range is about a factor of 20 to 30 for the 1 Hz spectral frequency. For the 10 Hz spectral frequency, the epistemic range is somewhat narrower, about a factor of 10.

Based on these sensitivities, the largest variation is based on the individual ground motion models implemented in the PSHA. The host background seismic source zone, ECC-AM, is the controlling seismic source for the MAFE range of interest at both the low and high frequency cases with a more significant contribution for the high frequency case relative to the low frequency case. These observations and the other sensitivity results presented in the FSAR for North Anna Unit 3 [24] are in agreement with the general observation for sites located in the CEUS that are not relatively close to a given RLME seismic source.

3.1.4 Horizontal and Vertical Response Spectra

The vertical response spectra (GMRS and FIRS) used as input to SPRA analyses were derived from the horizontal spectra by scaling using an appropriate frequency-dependent vertical-to-horizontal (V/H) ratio. The V/H ratio was developed in accordance with the guidance in Appendix J of NUREG/CR-6728 [17].

To illustrate the results of the vertical spectra development, Table 3-3 provides the frequency-specific data for the horizontal and vertical GMRS at the control point along with the corresponding V/H ratio and Figure 3-2 provides a plot of the horizontal and vertical GMRS.

Table 3-3: NAPS Control Point GMRS and V/H Ratios

Frequency (Hz)	Horizontal GMRS (g)	V/H Ratio	Vertical GMRS (g)
100.000	0.5721	1.0000	0.5721
90.000	0.6149	1.0376	0.6380
80.000	0.6965	1.0901	0.7593
70.000	0.8132	1.1275	0.9169
60.000	0.9601	1.1371	1.0918
50.000	1.1145	1.1245	1.2532
45.000	1.1652	1.1024	1.2845
40.000	1.2155	1.0423	1.2669
35.000	1.2617	0.9808	1.2374
30.000	1.2226	0.9368	1.1453
25.000	1.1889	0.8800	1.0462
20.000	1.1670	0.8256	0.9635
15.000	1.1707	0.7882	0.9227
12.500	1.1525	0.7708	0.8883
10.000	1.0508	0.7500	0.7881
9.000	0.9622	0.7500	0.7217
8.000	0.8562	0.7500	0.6421
7.000	0.7346	0.7500	0.5510
6.000	0.6068	0.7500	0.4551
5.000	0.4847	0.7500	0.3635
4.000	0.3702	0.7500	0.2777
3.000	0.2667	0.7500	0.2000
2.500	0.2159	0.7500	0.1619
2.000	0.1770	0.7500	0.1327
1.500	0.1317	0.7500	0.0988
1.250	0.1065	0.7500	0.0799

Table 3-3: NAPS Control Point GMRS and V/H Ratios

Frequency (Hz)	Horizontal GMRS (g)	V/H Ratio	Vertical GMRS (g)
1.000	0.0806	0.7500	0.0605
0.900	0.0745	0.7500	0.0559
0.800	0.0677	0.7500	0.0508
0.700	0.0602	0.7500	0.0452
0.600	0.0522	0.7500	0.0391
0.500	0.0435	0.7500	0.0326
0.400	0.0347	0.7500	0.0260
0.300	0.0260	0.7500	0.0195
0.200	0.0174	0.7500	0.0130
0.167	0.0145	0.7500	0.0109
0.125	0.0109	0.7500	0.0082
0.100	0.0087	0.7500	0.0065

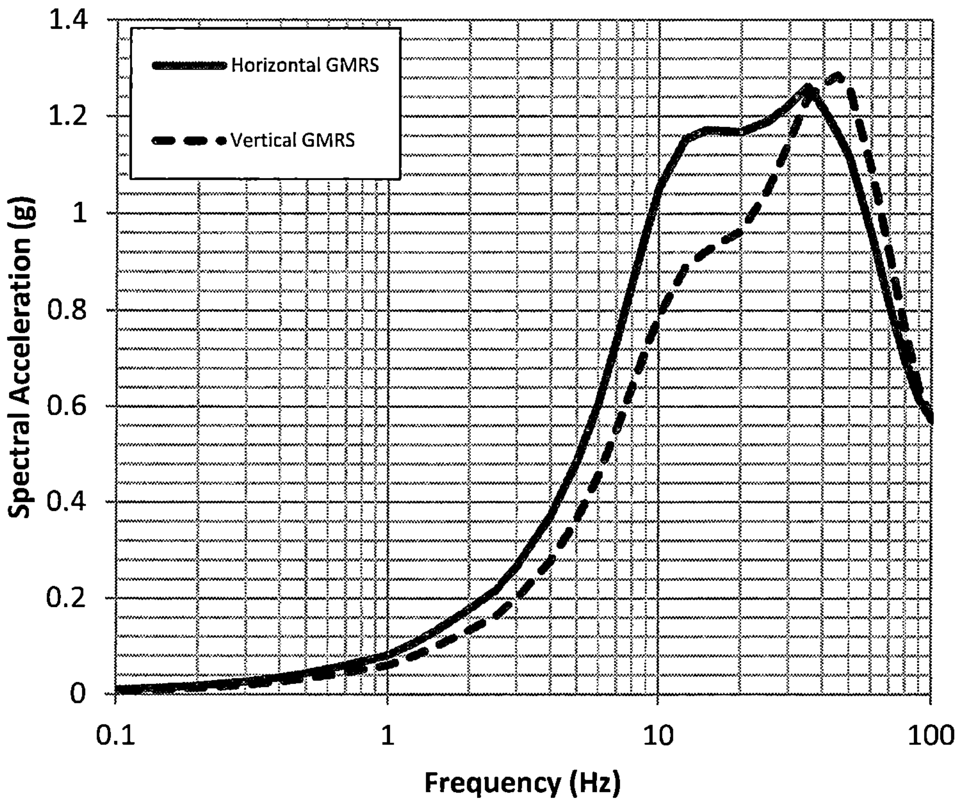


Figure 3-2: NAPS Horizontal and Vertical GMRS

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

4.1 Seismic Equipment List

For the NAPS SPRA, a seismic equipment list (SEL) was developed that includes those SSCs that are important to achieving safe shutdown following a seismic event, and for mitigating radioactivity release if core damage occurs, and that are included in the SPRA model. The methodology used to develop the SEL is consistent with the guidance provided in EPRI 3002000709, SPRA Implementation Guide [10].

4.1.1 SEL Development

The SEL includes the equipment and systems required to provide protection for all seismically induced initiating events, including those needed to address seismic induced fires and floods and to prevent early containment failure in an earthquake. The SEL forms the basis for the seismic fragility and systems analysis tasks. The initial SEL was developed from the SSCs modeled in the internal events PRA model. The internal events PRA model is a detailed and comprehensive logic model that includes the failure of SSCs needed for mitigating the various initiating events that could occur at the site. Additional SSCs were added to this initial list of SSCs that may have been screened out of the internal events PRA such as passive failures of buildings, structures, cable trays, HVAC ducts, block walls, and tanks. SSCs important for containment performance such as containment isolation and bypass events were added to the list.

<u>SSCs Modeled in the Level 1 and 2 Internal Events PRA</u>

The SSCs modeled in the level 1 and 2 internal events PRA are modeled using basic events that model various failure modes of the SSCs. The internal events PRA model also includes other basic events that model operator actions, component alignment events, and other non-component basic events. Over 5100 basic events are contained in the model, which models both unit 1 and 2 Core Damage Frequency (CDF) and Large Early Release Frequency (LERF). Basic events from the PRA database that do not represent structures or equipment (except for post-initiator operator actions and recovery actions) were removed from the SEL list. Some examples of such basic events to remove include the following:

- Configuration events (such as percentage of time a specific train is running)
- Environmental events (such as percentage of time that a given temperature range exists and HVAC is required)
- Pre-initiator operator actions and operator actions that cause initiating events
- Maintenance events
- Common-cause failure basic events (unless the associated random failure basic events do not exist separately in the models)

Screening notes were documented to denote why SSCs or basic events were screened in or out of the SEL. After the screening, over 3800 basic events were screened out of the SEL using the screening criteria for screening out basic events.

SSCs can also be screened out of the SEL based on a number of reasons. For example, some SSCs are known to have significantly high seismic capacity such that they are considered to be inherently rugged. These were screened out of the SEL because their contribution to seismic risk would likely be very small. All SSCs reviewed and screened from the SEL, and the associated basis for screening, are documented.

Passive SSCs

While the SSCs added to the SEL from the internal events PRA include SSCs needed for mitigating initiating events, the internal events PRA may not model passive SSCs because the probability of their random failure is relatively low. However, during seismic events, the probability of failure of some passive SSCs could be high and have a significant contribution to risk, such as:

- Tanks
- Buildings
- Cable Trays and Conduit
- Ventilation Ducts
- Piping
- Soil Failures
- Pressure Boundaries
- Block Walls
- Cranes
- Passive Valves

The general approach used in identifying passive SSCs to be included in the SEL was to obtain a list of all of the SSCs for the particular type from the plant equipment database and evaluate whether their failure impacts a mitigating function, causes flooding or fire, or impacts an operator action. Passive SSCs that were evaluated as screened out of the SEL, and the associated basis for screening, are documented. Those not screened out are modeled in the SPRA.

In addition, the plant areas housing SEL SSCs or in which operators would need to perform seismic response actions were reviewed for accessibility and evaluated for potential impact. The following structures were included in the SEL:

- Auxiliary Feedwater Pump Houses
- Service Water Pump House
- Service Water Valve House
- Service Building (including Control Room, Emergency Switchgear Rooms, and Emergency Diesel Generator Rooms)
- Auxiliary Building
- Containment Buildings
- Main Steam Valve Houses (and Quench Spray Pump Houses)
- Safeguards Buildings
- Fuel Oil Pump House
- Casing Cooling Pump Houses
- Beyond Design Basis Storage Building

Cabinets and Panels

The cabinets and panels included in the SEL are those that contain the following:

1. Indications and controls that Operators use to mitigate initiating events
2. Protection and control circuits that are used in reactor protection (RPS) and engineered safety feature (ESF) systems
3. Beyond Design Basis panels used to connect the cables from the FLEX 120VAC generators to the vital AC buses.

To mitigate transients, the operators follow the guidance in the emergency operating procedures (EOPs) to ensure the unit is safely shut down and the core remains covered and cooled. They rely on various instrumentation to verify successful operation of the mitigating safety functions. As part of the development of the SEL, the instrumentation required to safely shut the unit down was reviewed to determine what panels and cabinets should be evaluated for seismic capacity. The sensors and associated cabinets and control room panels are added to the composite SEL. Seismic failure of these cabinets and panels could impact operator actions.

Reactor protection circuits and sensors are not included for the following reasons. Seismic events generally involve a loss of offsite power, which would fail power to the motor-generator sets and thus result in trip of the control rods. For seismic events where there is no loss of offsite power, the ground acceleration level is much lower than the seismic capacity of the reactor protection system sensors and cabinets that it is very unlikely that an automatic trip signal would fail due to the seismic event. In addition, the operators would manually trip the reactor if the automatic trip system failed. Note that failure of the control rods to insert is included in the SEL.

There are a number of actuation systems that automatically actuate safety systems upon detection of adverse trends in key safety parameters. Instrumentation associated with the following was reviewed and added to the SEL:

- Safety Injection
- Containment Depressurization Actuation
- Phase A and B Containment Isolation
- Main Steam Isolation
- Undervoltage/Degraded Voltage
- Recirculation Mode Transfer
- Anticipated Transient Without Scram (ATWS) Mitigation System Actuation Circuitry

The primary focus of the review of the actuation circuits was to identify the sensors that monitor the various plant parameters and the cabinets that contain the components necessary to process the signals (e.g. power supplies, comparator card, etc.). Thus, the components added to the composite SEL from this review are mainly the sensors and cabinets.

The review of cabinets and panels described above resulted in including over 150 cabinets and panels in the SEL.

Containment Performance

The main objective of the Containment Performance evaluation is to identify seismic vulnerabilities that involve early failure of containment functions. This includes consideration of Containment integrity, Containment isolation, and other Containment functions.

Section 6.5.1 of the SPID [2] provides guidance for the SSCs that should be included in the SEL that support the containment functions. Section 5.8 of the SPRA Implementation Guide [10] also includes guidance for developing a level 2 (LERF) model in seismic PRAs. Both documents provide similar guidance with respect to the SSCs that should be included in the SEL for containment analysis. SSCs associated with the following functions were added to the SEL based on this guidance:

- Containment structure including pressure boundary
- Containment pressure suppression
- Containment isolation
- Interfacing system LOCA
- Hydrogen mitigation
- Containment vacuum
- Heat exchanger (inside Containment) pressure boundary

Approximately 160 SSCs were included in the SEL for the Containment Performance functions.

Seismic-induced Fire and Flood

Additional SSCs were added to the Initial SEL based on the seismic-fire and seismic-flood evaluations, as applicable.

A review was performed to identify potential plant vulnerabilities, given the combined effects of a seismic event and consequential internal fire hazard (i.e. a fire that occurs as a direct result of the seismic event), with a focus on seismically induced internal fires that may have the potential to significantly affect the plant seismic risk. Ignition sources, fire impact to SEL SSCs, spurious actuation of fire suppression systems (CO_2 and Halon), and impact on fire mitigation actions were reviewed. The seismic-induced fire review included plant walkdowns. The walkdowns and evaluations concluded that seismic-induced fire scenarios would not have a significant impact on seismic risk.

A review was performed to identify potential plant vulnerabilities, given the combined effects of a seismic event and consequential internal flood hazard (i.e. a flood that occurs as a direct result of the seismic event), with a focus on seismically induced internal floods that may have the potential to significantly affect the plant seismic risk. This evaluation sought to identify the potentially risk-significant seismically induced flood scenarios and to screen out those that were not expected to contribute significantly to plant risk or that would be subsumed by other damage states, such as building failures. Using the internal flooding PRA, a qualitative assessment was performed to identify potential seismic-induced flood scenarios that could be significant contributors to seismic risk. In addition to reviewing the North Anna Internal Flooding PRA, the following other sources were reviewed:

- Non-seismically qualified tanks
- Fire Protection piping
- Failure of Heat Exchangers
- Expansion Joints
- Spent Fuel Pool
- Sources within the non-seismic Turbine Building

The seismic-induced flooding evaluation concluded that flooding of the Auxiliary Building due to failure of the Component Cooling Heat Exchanger service water nozzles could result in a significant contribution to risk and this flood source was added to the SPRA model. The other flood sources and scenarios screened out from unique consideration in the SPRA.

Miscellaneous Additions

Relays and contactors that are prone to chatter, as identified from the relay chatter analysis (Section 4.1.2), were added to the SEL.

Additionally, in some cases, SSCs were added if potential seismic spatial interactions were identified between non-seismic SSCs near seismic SSCs (Seismic II over I) or other spatial issues were identified during walkdowns.

Other Inputs to SEL Development

A number of other inputs were reviewed and SSCs added to the SEL. These include:

- Assumptions in the internal events systems model
- Review of plant process and instrumentation drawings
- Comparison with the Individual Plant Examination of External Events (IPEEE) Safe Shutdown Equipment List [8]

This final SEL includes approximately 800 SSCs (not including relays) for each unit and is documented in the SPRA documentation.

4.1.2 Relay Evaluation

During a seismic event, vibratory ground motion can cause electrical contacts of seismically sensitive equipment (e.g., relays) to open or close (or 'chatter') inadvertently. The chattering of device contacts can potentially result in spurious signals to equipment. Most electrical contact device (herein referred to as relays) chatter is either acceptable (i.e., 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 NAPS SPRA, in accordance with SPID, Section 6.4.2 and ASME/ANS PRA Standard, Section 5-2.2. The evaluation resulted in most relay chatter scenarios screened from further evaluation based on no impact to component function. A summary of the relay evaluation is provided in Table 4-1.

Relays that could not be screened out were modeled in the SPRA. Relay-specific fragilities were determined for relays that were modeled using the separation of variables (SOV) approach.

Table 4-1 – Relay Chatter Evaluation Summary

	Unit 1	**Unit 2**	**Total**
SSCs Evaluated	341	322	703
Devices Evaluated	2674	2332	5006
Relays/Contactors Screened In			
MCCs	15	15	30
4KV Breaker	24	24	48
EDG	17	17	34
Aux Relays	6	6	12
Total	62	62	124

4.2 Walkdown Approach

This section provides a summary of the methodology and scope of the seismic walkdowns performed for the SPRA. Walkdowns were performed by personnel with appropriate qualifications and documented in accordance with the PRA Standard. The seismic review teams (SRT) included seismic engineering experts with extensive experience in fragility assessment.

Walkdowns were performed to assess the as-installed condition of those SSCs included on the seismic equipment list for use in determining their seismic capacity and performing initial screening, to identify potential spatial interactions, and look for potential seismic-induced fire/flood interactions. The walkdowns included samples of distribution systems such as piping, cable trays, electrical conduits, and HVAC ducting.

The SSC walkdowns were performed in accordance with the criteria provided in EPRI NP 6041-SL [7] and/or Seismic Qualification Utility Group (SQUG) guidance in the Generic Implementation Procedure (GIP) [21]. Most SEL components were reasonably accessible and in areas where inspection was possible. For the limited inaccessible components or those located in areas where significant ALARA concerns existed, alternate methods were used, such as photographs and reliance on design information. Walkdown information obtained was used to refine the SEL, and provide input to the fragilities analysis (as-installed conditions, dimensions, interactions etc.) and SPRA modeling (e.g., regarding correlation and rule-of-the-box considerations). In some cases, information from previously performed walkdowns, such as the IPEEE / USI A-46 Program [8] walkdown results, was used. In these cases, a walk-by of the applicable SSCs was performed to confirm the installed condition of the SSC was consistent with the previously performed walkdown and that the results remained applicable. The walk-by included verifying that the current material conditions and configurations were consistent with the conclusions, and to identify potential spatial interaction concerns. If applicable, recent walkdowns performed for the NTTF Recommendation 2.3: Seismic effort [22], post-Mineral earthquake plant inspections performed to support NAPS restart, and ESEP [20] were used provided these walkdowns furnished the appropriate level of detail needed for the SPRA.

Seismic-induced fire and flood and operator pathways walkdowns were also performed. The walkdown team included PRA Systems Analysts and plant Operations personnel as well as SRT members. The results of these walkdowns were used to refine the SEL as discussed in Section 4.1.

Walkdown procedures and results of walkdowns and walkbys (observations and conclusions) were documented as required per the PRA standard.

4.2.1 Significant Walkdown Results and Insights

Components on the SEL were evaluated for seismic anchorage and interaction effects, effects of component degradation, such as corrosion and concrete cracking, for consideration in the development of SEL fragilities. In addition, walkdowns were performed on operator pathways, and the potential for seismic-induced fire and flooding scenarios was assessed. The information gathered during walkdown inspections was adequate for use in developing the SSC fragilities for the SPRA.

No significant findings were noted during the NAPS seismic walkdowns. In a few instances, potential seismic spatial interaction concerns related to the higher seismic demand to be evaluated for the SPRA were identified. For example:

- Space heaters in the Emergency Diesel Generator (EDG) Rooms were identified as potential seismic spatial interaction concerns for nearby electrical cabinets and modifications to the heater supports have been developed to resolve the concern.
- Fire extinguishers and mobile firefighting carts were identified as potential seismic spatial interaction concerns for sensitive equipment in nearby cabinets and the firefighting equipment is being evaluated for restraint or relocation to resolve the concern.

No conditions that could challenge the NAPS seismic design basis were identified.

4.2.2 Seismic Equipment List and Seismic Walkdowns Technical Adequacy

Initial SEL Independent Technical Review

The Initial SEL is the result of screening SSCs from, or adding SSCs to, the final SEL using the general approach discussed above.

An independent in-process technical review of the initial draft SEL was performed by industry experts. The reviewers' overall assessment was that the SEL development was comprehensive and thorough. Comments from the review were resolved and documented in the SPRA documentation and the SEL was updated accordingly.

Walkdown Methodology Independent Technical Review

The methodology used to perform SSC walkdowns was reviewed by industry experts. Comments from the review were resolved and documented in the SPRA documentation.

The NAPS 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 NAPS SPRA SEL and seismic walkdowns are suitable for this SPRA application.

4.3 Dynamic Analysis of Structures

New dynamic analyses of structures that contain systems and components important to achieve safe shutdown were performed to develop structural responses and in-structure response spectra (ISRS). Scaling of responses from previous analyses (design basis, IPEEE etc.) was not performed for any structure because the shapes of the GMRS-based spectra at the foundations of structures in the SPRA are completely different when compared to the spectral shapes used in the past design-basis or other seismic analyses performed for NAPS.

NAPS is designated as a rock site since key safety-related structures (e.g., the reactor containments) are rock-founded. However, some auxiliary structures on the site are founded on soil, or partially on soil and partially on rock. Based on the founding condition, importance of the structure/components within it to the SPRA, and fidelity of the previous design-basis lumped-mass stick models (LMSM), various fixed-base and SSI analyses using either the previous LMSMs (with modifications where necessary to meet SPID requirements) or new finite element method (FEM) models for the key structures, were performed using deterministic and probabilistic methods, as appropriate. Table 4-2 shows the foundation condition, the type of model used, whether deterministic or probabilistic analysis was performed, and other relevant information for each structure that was analyzed for the SPRA.

Table 4-2: Description of Structures and Dynamic Analysis Methods for North Anna SPRA

Structure	Foundation Condition	Type of Model	Analysis Method	Comments/Other Information
Reactor Containment Buildings (Units 1 and 2)	Rock	LMSM*	Probabilistic SSI	Shear Wave velocity > 5000 ft/sec; SSI analysis performed with incoherence, 30 SSI input profiles used
Service Water Pump House	Soil	LMSM*	Deterministic SSI	LB, BE, UB cases, 5 sets of T-H
Service Building	Rock / Soil	FEM	Probabilistic SSI	Structure is partially on soil, partially on rock. SSI Analysis with 30 SSI input profiles used
Service Water Valve House	Soil	LMSM*	Deterministic SSI	LB, BE, UB cases, 5 sets of T-H used
Auxiliary Building	Rock / Soil	FEM	Probabilistic SSI	Structure is partially on soil, partially on rock. SSI analysis with 30 SSI input profiles used
Safeguards Building	Rock	FEM	Fixed Base	LB, BE, UB cases, 5 sets of T-H used
Auxiliary Feedwater Pump Houses	Rock	LMSM*	Fixed Base	One set of T-H (simple structure)
Unit 1 Main Steam Valve House	Rock	FEM	Fixed base	LB, BE, UB cases, 5 sets of T-H used
Unit 2 Main Steam Valve House	Soil	FEM	Probabilistic SSI	SSI Analysis, 30 SSI input profiles used

* LMSM models were reviewed based on the criteria in EPRI SPID and found to be acceptable for use in the SPRA.

4.3.1 Input Motions for Structural Analyses

The foundation input response spectra (FIRS) and SSI input response spectra, as applicable, were developed for each structure. These spectra were derived from site response analyses and correspond to the GMRS. Time-histories (T-H) corresponding to these spectra were developed and used as input motions in the probabilistic and deterministic analyses of structures.

4.3.2 Damping Values

In the structural dynamic analyses, 5% median damping value for concrete and 3% for steel were used per Table 3-4 of EPRI TR-103959. The 5% concrete damping is based on demands at approximately ½ the yield strength for reinforced concrete with cracking. This value is also consistent with Table 4-1 of EPRI NP-6041-SL, Rev. 1, which recommends 5% damping for reinforced concrete with moderate cracking. An exception was the Auxiliary Feedwater Pump House, which is a simple structure, and therefore, 4% damping value for concrete was used to develop 84% responses using one set of time-history input. Median and 84 percentile ISRS were developed at various locations and elevations of structures at various damping ratios (e.g., at 1%, 2%, 3%, 4%, 5%, 7%, and 10%).

4.3.3 Fixed-base Dynamic Analyses

As indicated in Table 4-2, three rock founded structures were analyzed as fixed-base because they are considered relatively low-mass buildings and the shear wave velocities at the foundation of each of these structures exceed 5000 ft/sec. Given the small footprint of these structures, their dynamic analyses were performed using coherent input motions.

Detailed dynamic analyses were performed for two structures - the Safeguards buildings (both units are similar, Unit 2 was used to develop responses for both buildings) and the Unit 1 Main Steam Valve House. New finite element models were developed for both these structures. Lower bound (LB), best estimate (BE) and upper bound (UB) cases were established by varying the structural stiffness by one-standard deviation (through concrete Young's modulus E_c using logarithmic standard deviation of 0.3) from the BE values; this corresponds to approximately ±15% variation of natural frequencies. A lower bound damping (LB-D) case was also analyzed with a lower damping of 3.7% (log standard deviation of 0.3) for the BE stiffness case. The input ground motions are applied using five sets of input time-histories which are spectrally matched to the SSI input response spectra. The ISRS were calculated at 301 frequencies at equal intervals in the logarithmic space between 0.1 Hz and 100 Hz (100 frequencies per decade). For each of the 20 seismic analysis cases, each node, and each damping ratio, the ISRS in the X direction are obtained by combining the acceleration response spectra (ARS) designated as XX (X response due to input in the X direction), XY (X response due to input in the Y direction), and XZ (X response due to input in the Z direction) using the square root of sum of squares (SRSS) method. The Y and Z direction ISRS are calculated similarly. For each node, each damping ratio, and each direction (X, Y, and Z), the logarithmic mean of the ISRS due to the five sets of input time-histories for the BE case is calculated and used as the BE ISRS. Similarly, the LB, UB and LB-D ISRS are calculated from their respective five time-history cases. The variation obtained in the ISRS results from the five BE seismic analysis cases reflects the variation due to phase differences between the five sets of input time-histories. The aleatory variation of the response due to the time-history phase variation is estimated as the logarithmic standard deviation of the ISRS obtained from the five analysis cases. The variation of the response due to damping is estimated on a frequency-by-frequency basis as the natural logarithm of the ratio of the BE and LB-D ISRS. The median ISRS (SA_{50}) is estimated frequency-by-frequency as the logarithmic mean of the BE, LB, and UB ISRS results. The variation of the response due to stiffness effects is estimated on a frequency-by-frequency basis as the natural logarithm of the ratio of the envelope of the BE, LB, and UB ISRS to the median ISRS results. The broadening of the envelope ISRS is done to fill in potential gaps between the LB, BE, and UB results by connecting the ISRS peaks using straight lines. Other sources of uncertainty, such as modelling, ground motion directivity effects and V/H ratio uncertainties are estimated separately. All uncertainties (aleatory and epistemic) are combined on a frequency-by-frequency basis to obtain the total composite uncertainty (β_c) for the ISRS. The 84th percentile ISRS are calculated as $SA_{50} \times e^{\beta_c}$. For the calculation of functional fragilities of equipment, peak clipped median and 84th percentile of ISRS are developed based on the methodology in Reference 9 (EPRI TR-103959).

The third structure analyzed as fixed base is the Auxiliary Feedwater Pump House. This is a simple structure and was analyzed using a lumped mass stick model (LMSM) with one set of 3-directional time-history input and 4% concrete damping to estimate 84% non-exceedance probability (NEP) responses. ISRS were calculated at several elevations of the structure at various damping values.

4.3.4 Soil Structure Interaction (SSI) Dynamic Analyses

As listed in Table 4-2, detailed probabilistic SSI analyses were performed for four key structures - Reactor Containment buildings (RCB - identical for both units), Service building (SB), Auxiliary building (AB), and the Unit 2 Main Steam Valve House (MSVH-2). Deterministic SSI analyses were performed for the Service Water Pump House (SWPH) and Service Water Valve House (SWVH) structures.

For the RCB, the best estimate of the shear wave velocity of the supporting media below it is approximately 5200 fps. This is higher than the threshold provided by SPID for fixed base analysis. However, an SSI analysis was performed for this building because the RC structure is tall and heavy and its response is expected to be affected by potential foundation rocking. Given the large building footprint and high-frequency-rich nature of the input motions, the ground motion incoherency was included in the development of ISRS from the SSI analysis. The existing LMSM of the RCB was considered adequate to capture the structural response and satisfied the SPID requirements for model adequacy. The horizontal and vertical FIRS for the RCB were calculated at the bottom of the mat foundation of RCB at elevation of 203 ft. These FIRS are calculated as geologic outcrop motion and are appropriate for use in the SSI analysis of the RCB as a surface structure. From the site response analysis, 30 sets of strain compatible soil properties consistent with FIRS and reflecting the rock property variations for the SSI analysis of the RCB were calculated. The SSI analysis used 30 sets of spectrally-matched time-histories which are tightly matched to the building FIRS Best estimate concrete strength of 5400 psi was used. The RCB structure was considered uncracked. Five engineering variables are identified for uncertainty modeling in the probabilistic SSI analyses: (1) Young's modulus for concrete, (2) Structural damping ratio, (3) Dynamic soil profile properties (4) Ground motion directional variability, and (5) ground motion V/H variability. The best estimate and logarithmic standard deviation (log-SD) of all random variables were explicitly included in the analysis. Using Latin Hypercube Sampling (LHS), 30 sets of SSI input parameters were developed by combining the above variables in an unbiased fashion. Other sources of uncertainty, namely, modelling uncertainty and coherency uncertainty, are explicitly estimated and included in the calculation of the total composite uncertainty. The median and 84th percentile of the probabilistic ISRS with and without ground motion incoherency were calculated at various damping values and median values of displacements relative to the top of the RCB foundation, with and without rigid body rotations were calculated. Peak clipped ISRS were generated using the methodology in Reference 9 (EPRI TR-103959). From the probabilistic analysis, frequency dependent aleatory and epistemic variabilities due to SSI and structural response were calculated in each direction in addition to the median and 84% ISRS responses. Structure-soil-structure interaction (SSSI) effects from the RCB were evaluated on nearby structures. These effects were found only to be significant for the vertical ISRS of SG and MSVH structures by causing slightly more than 10% increase in certain frequency bands; the ISRS within these structures were adjusted to include the SSSI effects.

The probabilistic SSI analyses of SB, AB and MSVH-2 were performed in a similar manner as described above for the RCB. However, for these three structures, new finite element models were developed instead of using the previous LMSMs. Similar to the RCB, ground motion incoherency was included for the AB and SB evaluation. Note that ground motion incoherency effects were not included in evaluation of the MSVH-2, due to its small footprint. The SSSI effect of the AB on nearby structures were also evaluated and found to be negligible.

For the SWPH and SWVH structures, deterministic SSI analyses were performed using updated LMSMs based on those used in the design basis calculations. SSI analyses were performed for the LB, BE and UB soil cases, each with 5 sets of time-histories, which yields 15 SSI analysis cases. Note that the variation of the structural properties (e.g., stiffness and damping) were not considered significant for these buildings because their SSI response were found to be entirely dominated by the soil impedance. The same approach for combining spatial components and uncertainties as discussed for the fixed-base analysis of Safeguards building and MSVH-1 was used and the median and 84th percentile ISRS were developed at various elevations and damping values.

4.3.5 Structure Response Analysis Technical Adequacy

The structural dynamic analyses were subjected to an in-process independent technical review by industry experts. Comments from the review were resolved and documented in the SPRA documentation.

The NAPS structural dynamic analyses 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 NAPS structural dynamic analyses are suitable for this SPRA application.

4.4 Fragility Analyses of SSCs

Seismic fragilities representing the conditional probabilities that a component would fail for a specified seismic ground motion or response as a function of that value were developed for SSCs in the SPRA seismic equipment list (SEL). The high-confidence-of-low-probability-of-failure (HCLPF) and median capacities were expressed as a fraction of the peak ground acceleration (PGA) of the control point GMRS. This PGA is 0.572g. With the exception of loss-of-offsite-power and LOCA events, which were based on the guidance in the SPRA Implementation Guide [10], seismic fragilities were plant-specific and were calculated in a realistic manner based on the actual conditions of the SSCs in the plant, as confirmed through detailed walkdowns.

This section summarizes the fragility analysis methodology, presents a tabulation of the fragilities with median capacity A_m and randomness and uncertainty variabilities β_r and β_u, and the calculation method and failure modes for those SSCs determined to be sufficiently risk important, based on the final SPRA quantification. Important assumptions and important sources of uncertainty, and any particular fragility-related insights identified, are also discussed.

4.4.1 SSC Screening Approach

Screening of SSCs primarily followed the guidance in Section 5.2 of SPRA Implementation Guide [10], and the guidance for screening in the Screening Prioritization and Implementation Details (SPID) – EPRI-1025287 [2]. The following methods were established for the screening of SSCs:

1. Screen inherently rugged SSCs. Inherently rugged SSCs were typically not retained in the logic model.

2. Develop a screening HCLPF using the SPID capacity-based screening criterion. This criterion can be applied during the walkdowns, and also via inspection of margins in the previous design basis, USI A-46 and/or IPEEE calculations.
3. Starting from the initial SPRA quantification, use a graded approach to screen SSCs and prioritize them for calculation of fragilities based on their risk significance. Review the validity of screening out SSCs that are not risk-significant via a surrogate screening event in the final logic model.

Using the North Anna control point hazard curve, the capacity-based screening HCLPF was calculated to be 1.8g. This was judged to be conservative; therefore, a 1.0g HCLPF screening threshold was used. Even though SSCs had capacities greater than this 1.0g screening HCLPF, they were retained in the SPRA logic model. A surrogate event, with a HCLPF of 0.6g, was included in the logic model which provided confirmation that the contribution to SCDF and SLERF from SSCs that could have been screened out was very low.

4.4.2 SSC Fragility Analysis Methodology

Detailed fragility analyses were performed for those SSCs that were not screened. The conservative deterministic failure margin (CDFM) approach per the guidance of EPRI NP-6041-SL, Revision 1 [7], supplemented by EPRI-1019200 [19] was initially used for most SSCs in the SEL, with the exception of relays. Using the CDFM approach, the HCLPF capacities were calculated using the 84% in-structure response spectra (ISRS). Detailed and more refined fragility analyses using the separation of variables (SOV) approach were performed for the top risk-important SSCs where the 50% confidence level ISRS were directly used to calculate their median capacities. The epistemic and aleatory variabilities for the fragilities calculated using the CDFM method were developed using one of the following two approaches: (a) Use the variabilities from the SPID, as appropriate, or (b) Use the detailed North Anna specific structural response variabilities (calculated frequency-by-frequency for each orthogonal direction), develop the equipment response variabilities per the guidance in EPRI TR-103959, and combine using SRSS. For the SOV method, variabilities were always calculated using approach (b) above.

In calculating the fragilities of SSCs, both structural and functional failure modes were considered. The seismic demand consisted of spectral accelerations up to a frequency of 20 Hz for structural failures such as bolted cabinets and also for functional failure modes with the exception of potentially high frequency sensitive SSCs, where a cut-off frequency limit of 40 Hz was used. For functional evaluation, peak clipped ISRS were used per the guidance in EPRI TR-103959. In many instances, functional capacities were based on Table 2-4 of EPRI NP-6041 with a modification that ISRS peaks, rather than ground peak spectral values, were used as recommended in EPRI-1019200. For SSCs covered by EPRI NP-6041 Table 2-4, 5% damped spectral peaks of only the horizontal ISRS (both directions) were compared to the modified peak spectral acceleration capacities of EPRI NP-6041 Table 2-4. An exception was the functional assessments of batteries and racks where vertical spectral peaks were also considered. EPRI NP-6041 Table 2-4 capacities were increased by a factor of 1.5 to obtain the HCLPF capacity (1% non-exceedance probability) and/or by a factor of 4.0 to obtain the median capacity, as recommended in EPRI-1019200. For some SSCs such as relays, functional capacities were based on the available shake table test data. When a static analysis was used to determine the capacity of a beam or frame type equipment item (e.g., anchorage evaluation of a cabinet), and if the natural frequency of the item was not known, peak spectral accelerations (slightly reduced as discussed below) were used with no multi-mode factor (i.e., a multi-mode factor of unity). Where it was judged that SSCs are not significantly sensitive to seismic accelerations in one horizontal direction more than the other, calculations of HCLPF capacities based on Table 2-4 of EPRI NP-6041 were refined by using the geometric average of the spectral accelerations (i.e., clipped ISRS spectral peaks up to the 20 Hz cut-off) in the two horizontal directions rather than using the maximum of two horizontal directions. The use of the geometric averaging is consistent with EPRI NP-6041, which notes that the screening guidance provided in Tables 2-3 and 2-4 are "in terms of five percent-damped peak spectral ground acceleration (average of two orthogonal horizontal components)." Where applicable, similar SSCs in close proximity were grouped together to perform a single fragility calculation. For fragility analyses of SSCs in structures analyzed using SSI, frequency (peak) shifting or broadening was limited to ±10% to address uncertainties in equipment natural frequencies because uncertainties in the soil and structural stiffnesses were already accounted for in the SSI analyses. However, for SSCs in structures analyzed using fixed-based dynamic analyses, the EPRI-recommended ±20% peak shifting or peak broadening was used. When the natural frequency of an equipment item was not available or unknown, peak of the ISRS was used but with a slight modification. It was reasonably assumed that the component frequency has equal probability of lying within ±15% of the frequency at which the peak spectral acceleration occurs and the spectral acceleration values within this ±15% window centered on the peak were averaged to obtain the seismic demand. In limited cases, small reductions in the ISRS were obtained based on the coupled analyses of structures and equipment.

The fragilities of structures were initially based on Table 2-3 of EPRI NP-6041; however, detailed fragility analyses were subsequently performed for several structures because either (a) the caveats of EPRI NP-6041 Table 2-3 could not be satisfied, or (b) the use of EPRI NP-6041 Table 2-3 was conservative and more realistic fragilities were needed because the structure was high in the risk significance list of SSCs for CDF or LERF. Fragilities of block walls in areas near the SEL items were developed by grouping the walls and analyzing the bounding cases.

The fragilities of reactor internals and other NSSS components were calculated using a scaling approach; these components have typically been demonstrated to have high capacities based on past SPRAs. Evaluations of representative distributed systems (piping, HVAC ducts, cable trays, and conduits) and associated components were performed; these components also have been shown to be generally rugged or have high capacities.

Correlation of components (or common cause failure) was considered in accordance with the ASME/ANS PRA Standard [4]. For the NAPS SPRA, if the equipment items were similar in design and physical orientation, with similar anchorage, and located in the same building on the same elevation, then these equipment items were assumed to be fully-correlated. In some cases, separate ISRS were used to develop location-specific fragilities for similar components located on the same floor. From the detailed finite element models of the structures, the seismic demand at different locations of the buildings was available. Since the seismic fragility of a component is a function of its seismic capacity and the seismic demand at the component location, similar components at different locations could have different demand, thus different fragilities. If the difference between the capacities of such components was small, then the components were considered correlated using the lower capacity value. However, if there was a significant difference in the fragilities of two similar components, then both detailed individual fragilities were entered in the logic model.

The impact of two (or multiple) failure modes, e.g., the functional and structural failure modes of a component, may cause the combined probability of failure to be slightly higher than the probability of either of the two failure modes, thus impacting the component's fragility. This occurs if the two failure modes are independent but not mutually exclusive (i.e., both could happen). The probability that at least one failure will occur is expressed by the union of two events (failures) A and B or $P(A \cup B)$, where $P(A \cup B) = P(A) + P(B) - P(A) \times P(B)$. This consideration is more pronounced when the HCLPF capacities of the two failure modes of an item are within about 20% of each other. Thus the fragilities for two failure modes, if within 20% of each other, were combined for the top risk significant SSCs to obtain a more accurate estimate of the fragility.

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 5.4-2 (for SCDF) and Table 5.5-2 (for SLERF). Refined fragility calculations were performed for the highest risk significant SSCs, as well as for selected other components.

4.4.4 SSC Fragility Analysis Technical Adequacy

A sampling of NAPS fragility analyses were subjected to an in-process independent technical review by industry experts. Comments from the review were resolved and documented in the SPRA documentation.

The NAPS fragility analyses 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 NAPS fragility analyses are suitable for this SPRA application.

5.0 Plant Seismic Logic Model

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

5.1 Development of the SPRA Plant Seismic Logic Model

The NAPS seismic response model was developed by starting with the NAPS internal events at power PRA model of record as of March 30, 2017, 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 credits FLEX equipment in the SBO sequences as well as 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. The following sections provide additional description in the development of the SPRA.

Seismic Equipment List

A seismic equipment list (SEL) was developed to define the scope of SSCs to include in the SPRA. Guidance in the SPRA Implementation Guide [10] was used in the development of the SEL. The SSCs modeled in the internal events PRA was used as a start for the SEL. Plant drawings, procedures and other design and configuration resources were reviewed and SSCs are added to the SEL to capture specific failures that can occur during seismic events and are not modeled in the FPIE PRA. The SSCs on the SEL were also reviewed to identify relays that could impact the SSC function if the relay contacts chattered during a seismic event. The circuits for the SSCs were reviewed to determine which relay contacts could impact the SSC function. Over 120 relays screened in and are modeled in the SPRA. Section 4.1 contains additional details of the SEL.

Initiating Events and Accident Sequences

The seismic hazard was modeled using 10 discrete hazard intervals (or bins) based on increasing peak ground acceleration. The seismic hazard bins are as listed in Table 3-2. 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)), without onsite AC power, and with response reflecting impact of the seismic event on mitigating systems. The event trees that model the seismic accident sequences are essentially the same as the event trees for the internal events core damage event trees. The following seismic-induced initiating events are modeled:

- LOOP
- ATWS
- Small-small LOCA
- Small LOCA
- Medium LOCA
- Large LOCA
- SBO
- Damage – includes excessive LOCA, building failures, distributed systems, etc

Modeling of Correlated Components

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. The seismic capacity for the group was assigned the capacity of the weakest component in the group. If the components are located in different areas or there are significant differences in the capacities of the components in the group due to differences in in-structure response spectra, the components were modeled as un-correlated. Section 4.4.2 contains additional information on correlation.

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 new HEPs for mitigating seismic failures of mitigating functions. HEPs were then adjusted as a function of seismic magnitude using a performance shaping factor approach consistent with the EPRI seismic HRA methodology [18]. Each Operator action is modeled by four HEP basic events that model the probability of failure for four different seismic hazard intervals. The ten hazard intervals are binned into the four HRA bins, which allow adjusting the HEP probabilities to account for increased stress and other shaping factors due to higher ground motion. The importance of the four HEPs is combined to obtain the overall importance of the Operator action.

The HEPs in the SPRA contains logic for failing the HEPs if SSCs needed by the Operators to complete the actions are failed. For example, failure of the main control panels or the process cabinets fails the HEPs.

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 dependency module in the HRA Calculator was used to determine the level of dependencies and the probability of the dependent HEPs. The dependent HEPs are added to the cutsets using a recovery file.

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. The seismic core damage accident sequences were mapped to the appropriate SLERF damage states based on the mapping in the internal events level 2 PRA. Most core damage sequences went to several SLERF damage states depending on failures in the Level 2 event trees from the internal events PRA. Some of the new core damage sequences, such as failure of the buildings and containment isolation, were directly mapped to SLERF. Others, such as a SBO sequences, were mapped based on similar core damage sequence mapping, using the level 2 event trees in the internal events PRA.

Additional SSC Failures Modeled in the SPRA

Certain 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:

- Distributed Systems – Cable Trays/Conduit
- Distributed Systems – Piping
- Building Failures - Reactor Containment Building, Auxiliary Building, Service Building
- Excessive LOCA caused by failure of Reactor Vessel, Steam Generators, Reactor Coolant Pumps

As part of the seismically-induced internal floods evaluation, seismic failure of the Component Cooling heat exchangers resulting in failure of the Service Water supply piping to the heat exchangers was included in the SPRA logic for failing SSCs in the Auxiliary building.

Table 5.1-1 Summary of Modifications to Internal Events CDF Fault Tree to Create Seismic CDF Fault Tree
Recovery of offsite power is not credited in the SBO sequences.
SBO event tree modified to not credit recovery of offsite power and added credit for selected FLEX actions: • load shed batteries to extend vital 125VDC battery life • repower 120VAC vital buses using FLEX generators • installing FLEX RCS Injection Pump to makeup to RCS (SSLOCA assumed)
RCP Seal LOCA model revised to use the Flowserve N9000 low leakage seal failure probabilities. The seals for all RCPs at North Anna have been replaced with Flowserve seals.
Added spurious opening of the pressurizer PORVs due to seismic failure of reactor pressure signals.
Revised HEPs in the seismic accident sequences to model four HEPs. The four seismic HEPs model the probability of failure at four different seismic ground motion bins.
Added seismic failures that impact Operator actions to fail HEPs. For example, seismic failure of the MCR panels, process cabinets, or instrumentation are modeled as failing HEPs.
Added over 160 fragility groups to the PRA fault trees that model seismic failure of the various SSCs that are used for mitigating seismic-induced accidents.
Various miscellaneous changes were made to the fault trees to accommodate new logic for the seismic model.

5.2 SPRA Plant Seismic Logic Model Technical Adequacy

The initial NAPS SPRA seismic plant response logic model was reviewed by industry experts. Comments from the review were resolved and documented in the SPRA documentation.

The NAPS 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 NAPS 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 NAPS 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 10 seismic initiators. FRANX was also used to generate the fault tree gates that model seismic failure of the SSC fragility groups modeled in the systems fault trees. The Unit 1 and 2 seismic CDF and seismic LERF top gates were quantified using the EPRI PRAQuant code to obtain cutset files that were then processed using the EPRI Code ACUBE. ACUBE was used to obtain a more accurate CDF/LERF by calculating the exact probability on the 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 were made as part of the seismic PRA quantification:

1. Due to the relatively low fragility of the insulators on the switchyard transformers, a loss of offsite power (LOOP) is likely to occur during most seismic events. The model includes SEIS-LOOP in all sequences in the Seismic Event Tree.
2. The seismic capacity for small-small LOCA is assumed to be 0.12g, which is the Safe Shutdown Earthquake (SSE) for North Anna. Guidance in SPRA Implementation Guide [10] includes several options, but generally recommends using the SSE as the capacity if detailed fragility calculations and walkdowns of the RCS piping are not performed.
3. Chatter of multiple relays in series where the contacts of the relays have to chatter in unison is considered to have a very low likelihood and therefore is not considered in the relay chatter evaluation.
4. Some SSCs that are part of alternate or backup mitigating functions were not credited in the SPRA either due to their low seismic capacities or to reduce the scope of the fragility analyses. For example, the alternate AC diesel generator is not credited because the seismic capacity of the building and support SSCs is likely to be low. Likewise, the Condensate Storage Tanks (CSTs), which are used

to supply the AFW pumps when the Emergency Condensate Storage Tank (normal AFW supply) is depleted, are not credited because the CSTs are unanchored, flat bottom tanks that typically have low capacity.

5. Seismic failure of the Component Cooling heat exchangers was assumed to result in flooding of the Auxiliary building SSCs from the failure of the Service Water supply piping to the heat exchangers. Isolation of the flood was not credited given the uncertainty in the size of the pipe breaks and the resulting flood flow rate.
6. Seismic failure of the Steam Generator (SG) tubes is not considered to be controlling and is subsumed by failure of the SG supports, which is assumed to result in an excessive LOCA.
7. Mission time is assumed to be 24 hours. A sensitivity using a mission time of 72 hours showed little impact on the SPRA results.

5.4 SCDF Results

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

The seismic PRA performed for NAPS shows that the point estimate seismic CDF is 6.0×10^{-5} for both Unit 1 and Unit 2. A discussion of the mean SCDF with uncertainty distribution reflecting the uncertainties in the hazard, fragilities, and model data is presented in Section 5.6. Important contributors are discussed in the following paragraphs.

The top SCDF accident sequences based on Fussell-Vesely (FV) importance of the sequence flags are documented in Table 5.4-1. These sequences contribute over 90% of the SCDF. Note that these sequences have been combined across all the hazard bin intervals. Three of the top seven sequences are seismic events with a loss of offsite power and failure of the EDGs due to relay chatter resulting in a Station Blackout (SBO).

SSCs with the most significant seismic failure contributions to SCDF are listed in Table 5.4-2, sorted by FV importance. The seismic fragilities for each of the significant contributors are also provided in Table 5.4-2, along with the corresponding limiting seismic failure mode and method of fragility calculation. Importance analyses were performed for both SCDF and SLERF, using the ACUBE code. From the ACUBE output, FV values for the seismic failures (i.e. fragility groups) is the sum of the FV values for each hazard interval.

The FV listing shows the top individual contributor to SCDF as seismically induced Loss of Offsite Power (LOOP), due to the low median seismic capacity assumed for offsite power failure following a seismic event. The fragility for LOOP is a value from the SPRA Implementation Guide [10] and considered reasonably representative for NAPS.

The next highest contributor is seismically induced small-small LOCA (SSLOCA), which similar to LOOP, has a low median capacity. The capacity is based on the SPRA Implementation Guide, which provides guidance for modeling SSLOCA and recommends the capacity (i.e. HCLPF) be set to the Safe Shutdown Earthquake (SSE), which for NAPS is 0.12g.

Most of the top seismic failures involve chatter of relays that result in failure of emergency power, or key safety system pumps due to chatter of the 4kv breaker lockout relays. The capacity of these relays is relatively low and the seismic failures for each are assumed to be correlated (e.g., both trains of LHSI pumps fail due to lockout). The model does not currently credit Operator action to reset the relays and restore the mitigating functions.

Other top seismic failures involve failure of the 120vac vital buses and the vital bus inverters, which not only fail the actuation systems and power to some SSCs, but also fails critical instrumentation relied on for Operator actions (i.e. fails Human Error Probability basic events in the model). Failure of the vital 125v DC buses and batteries also have significant FV importances, which have similar impacts as the vital buses.

Table 5.4-1 Summary of Top SCDF Accident Sequences

FV Importance	Accident Sequence Description
U1 = 8.3E-02 (29%) U2 = 7.7E-02 (27%)	Station Blackout (SBO) with successful Auxiliary Feedwater (i.e. Turbine-driven AFW pump) but either Long Term Cooling fails, Cooldown and Depressurization fails, or the SI Accumulators fail. The dominant failures are: • SBO caused mainly by relay chatter of EDG output breaker or 4kv breaker supply to the 480V buses and MCCs; no credit for Operators recovery of the relay chatter. • Seismic failure of the Steam Generator (SG) Power Operated Relief Valves (PORVs) • Seismic failure of the 120VAC vital buses, DC buses and inverters that power critical instrument transmitters required for Operator actions • Seismic failure of Main Control Room panels Sequence Ux-SBO-SEIS-02

Table 5.4-1 Summary of Top SCDF Accident Sequences

FV Importance	Accident Sequence Description
U1 = 6.8E-02 (24%) U2 = 7.2E-02 (25%)	Loss of Offsite Power with a Small-small LOCA and successful AFW and long term cooling and failure of RCS makeup using the Charging pumps. The dominant failures are: • Seismic failure of the RWST • Seismic failure of SW pumps due to chatter of the lockout relays which fails cooling to the Charging pumps • Seismic failure of the Charging pumps due to chatter of the lockout relays • Seismic failure of the Low Head Safety Injection (LHSI) pumps due to relay chatter or due to failure of the Safeguards area ventilation where the pumps are located • Seismic failure of the SG PORVs Sequence Ux-LOOP-SEIS-01
U1 = 4.2E-02 (15%) U2 = 4.0E-02 (14%)	SBO with successful AFW and Long Term Cooling, but FLEX mitigation fails due to the following: • Seismic failure of the RWST which fails RCS makeup from the FLEX RCS Injection Pump • Seismic failure of the vital 125vdc batteries resulting in loss of critical instrumentation • Seismic failure of the FLEX electrical distribution panel Sequence Ux-SBO-SEIS-01
U1 = 3.1E-02 (11%) U2 = 3.0E-02 (11%)	SBO with failures that go directly to core damage due to insufficient time to mitigate (large, medium, small LOCAs, ATWS). Dominant failures that result in this SBO direct core damage sequence are: • Small LOCA • Control Rods • Medium LOCA Sequence Ux-LOOP-SEIS-04

Table 5.4-1 Summary of Top SCDF Accident Sequences

FV Importance	Accident Sequence Description
U1 = 2.1E-02 (7%) U2 = 2.1E-02 (7%)	Loss of Offsite Power with successful AFW but long term cooling fails (i.e. align Service Water or Fire Protection to AFW after Emergency Condensate Storage Tank depletes) and Bleed & Feed fails. The dominant failures are: • Seismic failure of the 120VAC vital buses that power critical instrumentation (which fails HEPs for long term cooling and Bleed & Feed) • Failure of the MCR panels or process cabinets, which also fails HEPs • Seismic failure of MCCs that power the MOVs for High Head SI and pressurizer PORVs • Seismic failure of SW pumphouse or SW reservoir, which fails SW • Seismic failure of process cabinets, which fails actuation signals and critical instrumentation Sequence Ux-LOOP-SEIS-03
U1 = 1.3E-02 (4%) U2 = 1.3E-02 (5%)	Small LOCA (2" break) with successful AFW but with failure of the High Head SI injection. The dominant failures are: • Chatter of the HHSI pump lockout relays results in failure of high head safety injection • Chatter of the Service Water lockout relays results in failure of cooling to the HHSI pumps • Seismic failure of the RWST • Seismic failure of the Component Cooling heat exchangers results in a flood that fails the HHSI pumps Sequence Ux-SLOCA-SEIS-04

Table 5.4-1 Summary of Top SCDF Accident Sequences	
FV Importance	**Accident Sequence Description**
U1 = 1.1E-02 (4%) U2 = 1.2E-02 (4%)	Small LOCA (2" break) with successful AFW and High Head SI but with failure of the High Head SI recirculation when the RWST is depleted. The dominant failures are: • Chatter of the Service Water lockout relays results in failure of containment sump cooling • Chatter of the Low Head SI pump lockout relays results in failure of the LHSI pumps • Failure of the Safeguards area ventilation due to seismic failure of the upper levels of the Auxiliary building, which fails the Safeguards area fans; Failure of the Safeguards are ventilation fails the LHSI pumps • Chatter of relays in the Recirculation Spray (RS) pumps causing them to pre-maturely start before the containment sump contains water. Failure of the RS results in failure of containment sump recirculation since the pumps are required for sump cooling. Sequence Ux-SLOCA-SEIS-01

Table 5.4-2 SCDF Importance Measures Ranked by FV								
Fragility Groups	**Fragility Group Description**	**U1 CDF FV**	**U2 CDF FV**	**Am**	**Br**	**Bu**	**Failure Mode**	**Fragility Method**
SEIS-LOOP	SEISMIC-INDUCED LOSS OF OFFSITE POWER	6.91E-01	6.90E-01	0.30	0.27	0.40	Generic	EPRI Report [10]
SEIS-SSLOCA	SEISMIC-INDUCED SMALL-SMALL LOCA	9.51E-02	1.02E-01	0.30	0.28	0.28	Generic	HCLPF is set to SSE [10]
SEIS-EE-BKR-HJ8-RLY	4KV to 480V BUS BREAKERS - RELAY CHATTER	6.76E-02	6.90E-02	0.52	0.24	0.52	Functional	SOV
SEIS-SW-P-1AB-RLY	SERVICE WATER PUMPS - RELAY CHATTER	3.84E-02	3.96E-02	0.77	0.24	0.49	Functional	SOV
SEIS-CH-P-1ABC-RLY	CHARGING PUMPS - RELAY CHATTER	3.63E-02	3.75E-02	0.77	0.24	0.49	Functional	SOV
SEIS-SLOCA	SEISMIC-INDUCED SMALL LOCA	3.33E-02	3.37E-02	1.00	0.30	0.40	Generic	EPRI Report [10]
SEIS-VB-INV-1234	120 VAC VITAL BUS INVERTERS	3.26E-02	3.23E-02	1.10	0.19	0.58	Functional	SOV
SEIS-SI-P-1AB-RLY	LOW HEAD SI PUMP - RELAY CHATTER	2.83E-02	2.83E-02	0.77	0.24	0.49	Functional	SOV
SEIS-FW-P-3AB-RLY	MOTOR-DRIVEN AFW PUMPS - RELAY CHATTER	2.65E-02	2.65E-02	0.77	0.24	0.49	Functional	SOV
SEIS-EE-BKR-HJ2-RLY	EDG OUTPUT BREAKERS - RELAY	1.90E-02	1.94E-02	0.77	0.24	0.49	Functional	SOV
SEIS-EP-CB-12ABCD	125 VDC DISTRIBUTION PANELS	1.46E-02	1.48E-02	1.15	0.24	0.38	Functional	CDFM Hybrid
SEIS-EP-CB-4ABCD	120 VAC VITAL BUS DISTRIBUTION PANELS	1.40E-02	1.41E-02	1.16	0.24	0.38	Anchorage	CDFM Hybrid
SEIS-EDG-HJ-RLY	EMERGENCY DIESEL GENERATORS - RELAY CHATTER	1.09E-02	1.09E-02	0.70	0.24	0.83	Functional	SOV
SEIS-BY-B-1-24	STATION BATTERIES 1-II AND 1-IV	8.53E-03	8.38E-03	1.14	0.24	0.38	Structural failure of rack	CDFM Hybrid
SEIS-EI-CB-MCR-PNL	SEISMIC FAILURE OF MCR BOARDS AND PANELS	7.55E-03	7.61E-03	1.30	0.24	0.38	Functional	CDFM Hybrid

The most significant non-seismic SSC failures (e.g., random failures of modeled components during the SPRA mission time) are listed in Table 5.4-3. The unavailability of the diesel-driven fire pump and FLEX equipment (pumps and generators) constitutes the highest FV importance for SCDF. These SSCs support mitigation of a SBO.

Table 5.4-3 SCDF Importance Measures Ranked by FV for Non-Seismic Failures

Unit 1 Model Basic Events	Prob	SCDF FV	Description
1FP-DDP--TM-2	3.16E-02	1.33E-02	DIESEL-DRIVEN FIRE PUMP 1-FP-P-2 OUT OF SERVICE FOR TEST OR MAINTENANCE
0BDBEDG--FR-1A-FLEX	2.04E-02	7.05E-03	FLEX DIESEL GENERATOR FAILS TO RUN
0BDBDDP--FS-3A-FLEX	5.46E-03	1.84E-03	FLEX RCS INJECTION PUMP (00-BDB-P-3A) FAILS TO START
0BDBEDG--FS-1A-FLEX	4.53E-03	1.53E-03	FLEX DIESEL GENERATOR FAILS TO START
1FW-TRB--TM-2	2.81E-03	1.43E-03	U1 TURBINE-DRIVEN AFW PUMP OUT OF SERVICE FOR TEST OR MAINTENANCE
1FW-TRB--FS-2	1.92E-03	9.63E-04	U1 TURBINE-DRIVEN AFW PUMP FAILS TO START
0BDBEDG--FL-1A-FLEX	2.90E-03	9.59E-04	FLEX DIESEL GENERATOR FAILS TO LOAD
1FW-TRB--FR-2	1.71E-03	8.55E-04	U1 TURBINE-DRIVEN AFW PUMP FAILS TO RUN
1FP-DDP--FR-2	2.13E-03	8.33E-04	DIESEL-DRIVEN FIRE PUMP 1-FP-P-2 FAILS TO RUN
0BDBDDP--FR-3A-FLEX	2.28E-03	7.37E-04	FLEX RCS INJECTION PUMP (00-BDB-P-3A) FAILS TO RUN
Unit 2 Model Basic Events and FV Importance			
1FP-DDP--TM-2	3.16E-02	1.32E-02	DIESEL-DRIVEN FIRE PUMP 1-FP-P-2 OUT OF SERVICE FOR TEST OR MAINTENANCE
0BDBEDG--FR-1A-FLEX	2.04E-02	6.88E-03	FLEX DIESEL GENERATOR FAILS TO RUN
0BDBDDP--FS-3A-FLEX	5.46E-03	1.79E-03	FLEX RCS INJECTION PUMP (00-BDB-P-3A) FAILS TO START
0BDBEDG--FS-1A-FLEX	4.53E-03	1.49E-03	FLEX DIESEL GENERATOR FAILS TO START
2FW-TRB--TM-2	2.81E-03	1.43E-03	U2 TURBINE-DRIVEN AFW PUMP OUT OF SERVICE FOR TEST OR MAINTENANCE
2FW-TRB--FS-2	1.92E-03	9.61E-04	U2 TURBINE-DRIVEN AFW PUMP FAILS TO START
0BDBEDG--FL-1A-FLEX	2.90E-03	9.41E-04	FLEX DIESEL GENERATOR FAILS TO LOAD
2FW-TRB--FR-2	1.71E-03	8.53E-04	U2 TURBINE-DRIVEN AFW PUMP FAILS TO RUN
1FP-DDP--FR-2	2.13E-03	8.24E-04	DIESEL-DRIVEN FIRE PUMP 1-FP-P-2 FAILS TO RUN
0BDBDDP--FR-3A-FLEX	2.28E-03	7.24E-04	FLEX RCS INJECTION PUMP (00-BDB-P-3A) FAILS TO RUN

A summary of the SCDF results for each seismic hazard interval is presented in Table 5.4-4. Figure 5.4-1 shows a bar chart of the Unit 1 SCDF as a function of PGA (Unit 2 results are the same as Unit 1). The seismic ground motions that contribute the most to SCDF are in the 0.5g to 1.0g range (%G04 - %G07). The small increase in SCDF contribution for the %G07 and %G08 intervals is due to the width of the intervals being larger than the lower intervals.

5.4-4 Contribution to SCDF by Acceleration Interval

	PGA	Initiator Frequency	U1 CDF	% Total U1 CDF	U1 CCDP	U2 CDF	% Total U2 CDF	U2 CCDP
%G01	0.06g to <0.3g	9.21E-04	6.87E-08	0.1%	0.00	6.87E-08	0.1%	0.00
%G02	0.3g to <0.4g	5.34E-05	3.16E-06	5.3%	0.06	3.12E-06	5.2%	0.06
%G03	0.4g to <0.5g	3.01E-05	7.10E-06	11.8%	0.24	6.99E-06	11.7%	0.23
%G04	0.5g to <0.6g	1.79E-05	1.06E-05	17.7%	0.59	1.06E-05	17.7%	0.59
%G05	0.6g to <0.7g	1.11E-05	9.30E-06	15.5%	0.84	9.30E-06	15.5%	0.84
%G06	0.7g to <0.8g	7.08E-06	6.74E-06	11.2%	0.95	6.74E-06	11.3%	0.95
%G07	0.8g to <1g	8.26E-06	8.19E-06	13.7%	0.99	8.19E-06	13.7%	0.99
%G08	1g to <1.5g	9.09E-06	9.09E-06	15.2%	1.00	9.09E-06	15.2%	1.00
%G09	1.5g to <2.5g	4.25E-06	4.25E-06	7.1%	1.00	4.25E-06	7.1%	1.00
%G10	>2.5g	1.48E-06	1.48E-06	2.5%	1.00	1.48E-06	2.5%	1.00

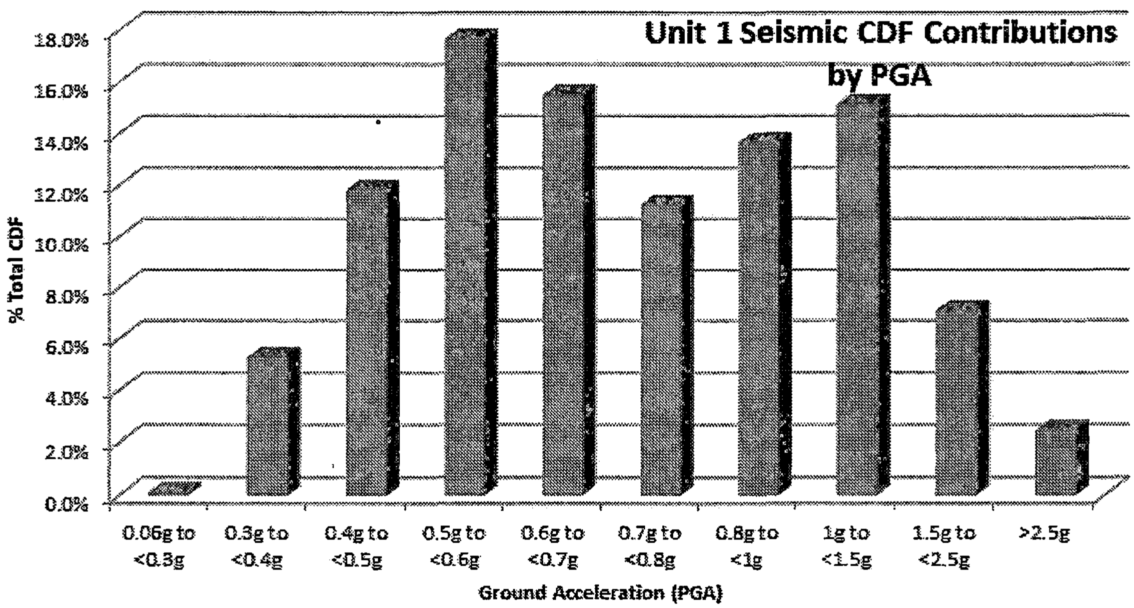


Figure 5.4-1 Unit 1 SCDF Contributions by PGA

The most significant Operator actions modeled as Human Error Probability (HEPs) in the model are listed in Table 5.4-5. As discussed in Section 5.1, the seismic PRA models each Operator action using four HEP basic events per action, which model different failure probabilities for the higher ground motions. The FV importance of the Operator action is the sum of the FV importance for each of the four HEP basic events. The important actions involve restoring alternate cooling to the Charging pumps upon loss of normal SW cooling, aligning the turbine-driven AFW pump to all three SGs during a SBO, and installing and starting the FLEX RCS injection pump during a SBO.

5.4-5 - SCDF Importance Measures Ranked by FV for Operator Actions

HEP Basic Event	SCDF FV	Description
HEP-C-0SW-CHP-ALT	6.61E-02	Restore Cooling to the Charging Pumps from Fire Protection or Primary Grade Water systems
HEP-C-ALIGN-TDAFW	2.66E-02	Align turbine-driven AFW Pump to the B and C SGs
HEP-C-FLEX-RIP	1.55E-02	Install and Start FLEX RCS Injection Pump
HEP-C-FLEX-LOADSHED	7.87E-03	Load shed the vital 125vdc batteries during SBO
HEP-C-1SI-OPN1836	6.71E-03	Open 1-SI-MOV-1836 to Align Alternate Flow Path for HHSI
HEP-C-1FW-AFWSPLY	5.50E-03	Align SW OR Fire Protection Water to AFW Pumps When ECST Depletes
HEP-C-FLEX-VAC	5.36E-03	Install FLEX Generator to Power Vital Buses
REC-SEIS-FLD-CCHX	5.11E-03	Isolate SW Flood in Auxiliary Building Caused by Seismic Failure of the CCW Heat Exchangers
HEP SCDF FV Importance in Unit 2 Model		
HEP-C-0SW-CHP-ALT	6.90E-02	Restore Cooling to the Charging Pumps from Fire Protection or Primary Grade Water systems
HEP-C-ALIGN-TDAFW	2.59E-02	Align TDAFW Pump to the B and C SGs
HEP-C-FLEX-RIP	1.51E-02	Install and Start FLEX RCS Injection Pump
HEP-C-FLEX-LOADSHED	7.67E-03	Load shed the vital 125vdc batteries during SBO
HEP-C-2SI-OPN2836	6.82E-03	Open 2-SI-MOV-2836 to Align Alternate Flow Path for HHSI
HEP-C-2FW-AFWSPLY	5.57E-03	Align SW or Fire Protection Water to AFW Pumps When ECST Depletes
HEP-C-FLEX-VAC	5.19E-03	Install FLEX Generator to Power Vital Buses
REC-SEIS-FLD-CCHX	5.19E-03	Isolate SW Flood in Auxiliary Building Caused by Seismic Failure of the Component Cooling Heat Exchangers

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 and their relative SLERF contributions.

The seismic PRA performed for NAPS shows that the point estimate seismic LERF is 1.6×10^{-5} for both Unit 1 and Unit 2. A discussion of the mean SLERF with uncertainty distribution reflecting the uncertainties in the hazard, fragilities, and model data is presented in Section 5.6. Important contributors are discussed in the following paragraphs.

The top SLERF accident sequences based on FV importance of the sequence flags are documented in Table 5.5-1. These sequences contribute over 80% of the SLERF. Note that these sequences have been combined across all the hazard bin intervals. Three of the top seven sequences are seismic events with a loss of offsite power and failure of the EDGs due to relay chatter resulting in a Station Blackout (SBO).

These core damage sequences progress to a release generally due to temperature-induced steam generator tube rupture caused by a loss of AFW which results in dry out of the SGs. Some of the sequences where there is a loss of containment sump cooling, a release occurs due to containment failure caused by containment overpressurization upon loss of heat removal from the sump.

Table 5.5-1 Summary of Top SLERF Accident Sequences

FV Importance	Accident Sequence Description
U1 = 7.35E-02 (26.8%) U2 = 7.24E-02 (25.9%)	SBO with successful AFW (i.e. Turbine-driven AFW pump) but either Long Term Cooling fails, Cooldown and Depressurization fails, or the SI Accumulators fail. The dominant failures are: • SBO caused mainly by relay chatter of EDG output breaker or 4kv breaker supply to the 480V buses and MCCs; no credit for Operators recovery of the relay chatter. • Seismic failure of the SG PORVs • Seismic failure of the 120VAC vital buses, DC buses and inverters that power critical instrument transmitters required for Operator actions • Seismic failure of Main Control Room panels Sequence Ux-SBO-SEIS-02

Table 5.5-1 Summary of Top SLERF Accident Sequences

FV Importance	Accident Sequence Description
U1 = 4.37E-02 (15.9%) U2 = 4.73E-02 (16.9%)	SBO with successful AFW and Long Term Cooling, but FLEX mitigation fails due to the following: • Seismic failure of the RWST which fails RCS makeup from the FLEX RCS Injection Pump • Seismic failure of the vital 125vdc batteries resulting in loss of critical instrumentation • Seismic failure of the FLEX electrical distribution panel Sequence Ux-SBO-SEIS-01
U1 = 3.90E-02 (14.2%) U2 = 4.16E-02 (14.9%)	SBO with failures that go directly to core damage due to insufficient time to mitigate (large, medium, small LOCAs, ATWS). Dominant failures that result in direct core damage are: • Small LOCA • Control Rods Sequence Ux-SBO-SEIS-04
U1 = 2.09E-02 (7.6%) U2 = 2.27E-02 (8.1%)	Small LOCA (2" break) with successful AFW and High Head SI but with failure of the High Head SI recirculation when the RWST is depleted. The dominant failures are: • Failure of the Service Water MOVs in the Quench Spray Pumphouse that need to open to provide cooling to the Recirculation Spray Heat Exchangers for containment sump cooling • Chatter of relays in the Recirculation Spray (RS) pumps causing them to pre-maturely start before the containment sump contains water. Failure of the RS pumps result in failure of containment sump recirculation since the pumps are required for sump cooling. • Sump recirculation fails due to failure of containment heat removal resulting in containment failure prior to core damage. Contributes to LERF with conditional probability of 1.0 since containment is open at the time of core damage. Sequence Ux-SLOCA-SEIS-01

Table 5.5-1 Summary of Top SLERF Accident Sequences	
FV Importance	**Accident Sequence Description**
U1 = 1.86E-02 (6.8%) U2 = 2.05E-02 (7.3%)	Loss of Offsite Power with successful AFW but long term cooling fails (i.e. align Service Water or Fire Protection to AFW after Emergency Condensate Storage Tank depletes) and Bleed & Feed fails. The dominant failures are: • Seismic failure of the 120VAC vital buses that power critical instrumentation (which fails HEPs for long term cooling and Bleed & Feed) • Failure of the MCR panels or process cabinets, which also fails HEPs • Operator actions to align an alternate source of water to AFW • Seismic failure of the RWST • Failure of the Safeguards area ventilation due to seismic failure of the upper levels of the Auxiliary building, which fails the Safeguards area fans; Failure of the Safeguards are ventilation fails the LHSI pumps Sequence Ux-LOOP-SEIS-03
U1 = 1.61E-02 (5.9%) U2 = 1.31E-02 (4.7%)	Loss of Offsite Power with failure of AFW and failure of Bleed & Feed. The dominant failures are: • Seismic failure of the turbine-driven AFW pump and relay chatter of the motor-driven AFW pumps • Failure of the Safeguards area ventilation due to seismic failure of the upper levels of the Auxiliary building, which fails the Safeguards area fans; Failure of the Safeguards are ventilation fails the LHSI pumps • Chatter of the HHSI pump lockout relays • Failure of the Operator action to establish Bleed and Feed. Sequence Ux-LOOP-SEIS-05
U1 = 1.42E-02 (5.2%) U2 = 1.49E-02 (5.3%)	Seismic event causes damage to the reactor containment building resulting in failure of the RCS, core damage and large early release. Sequence Ux-DMG-SEIS-05

SSCs with the most significant seismic failure contributions to SLERF are listed in Table 5.5-2, sorted by FV importance. The seismic fragilities for each of the significant contributors are also provided in Table 5.5-2, along with the corresponding limiting seismic failure mode and method of fragility calculation. Importance analyses were performed for SLERF using the ACUBE code. From the ACUBE output, FV values for the seismic failures (i.e. fragility groups) is the sum of the FV values for each hazard interval.

The FV listing shows the top individual contributor to SLERF as seismically induced Loss of Offsite Power (LOOP), due to the low median seismic capacity assumed for offsite power failure following a seismic event. The fragility for LOOP is a value from SPRA Implementation Guide [10] and considered reasonably representative for NAPS.

The next highest contributor is seismically induced small LOCA (SLOCA), which has a relatively low median capacity, and is based on the SPRA Implementation Guide [10]. The relay chatter failures of the Recirculation Spray pumps (needed for sump cooling) and the AFW pumps show up in these SLOCA cutsets, where these relays have relatively low capacities.

The next highest contributor to SLERF is seismic failure of the containment building, which is assumed to result in direct core damage as well as direct LERF.

Other top contributors to SLERF are failures that fail containment sump cooling such as relay chatter of the RS pumps, seismic failure of the four RS heat exchangers as well as seismic failure of the Service Water MOVs in the Quench Spray Pumphouse basement that have to open to provide cooling to the RS heat exchangers. There are also a number of other seismic failures that have SLERF FV values greater than 0.005 that are in SBO, LOOP and SLOCA sequences.

Table 5.5-2 SLERF Importance Measures Ranked by FV

Fragility Groups	Fragility Group Description	U1 LERF FV	U2 LERF FV	Am	Br	Bu	Failure Mode	Fragility Method
SEIS-LOOP	SEISMIC-INDUCED LOSS OF OFFSITE POWER	5.01E-01	5.05E-01	0.30	0.27	0.40	Generic	EPRI Report [10]
SEIS-SLOCA	SEISMIC-INDUCED SMALL LOCA	9.07E-02	9.19E-02	1.00	0.30	0.40	Generic	EPRI Report [10]
SEIS-RS-P-1AB-RLY	INSIDE RS PUMP - RELAY CHATTER	5.46E-02	5.25E-02	1.37	0.23	0.48	Functional	SOV
SEIS-BLDG-RC	REACTOR CONTAINMENT BUILDING	4.40E-02	4.31E-02	1.71	0.24	0.26	Structural	CDFM
SEIS-RS-P-2AB-RLYSS	Outside RS Pumps Spuriously Start due to Relay Chatter	2.94E-02	2.81E-02	1.37	0.23	0.48	Functional	SOV
SEIS-FW-P-3AB-RLY	MOTOR-DRIVEN AFW PUMPS - RELAY CHATTER	2.43E-02	1.80E-02	0.77	0.24	0.49	Functional	SOV
SEIS-RS-P-2AB	OUTSIDE RECIRC SPRAY PUMPS	2.29E-02	2.18E-02	1.38	0.24	0.32	Anchorage	CDFM
SEIS-FW-P-2	TURBINE-DRIVEN AUXILIARY FEEDWATER PUMP	2.26E-02	2.36E-02	1.60	0.24	0.32	Functional	CDFM
SEIS-EE-BKR-HJ8-RLY	4KV TO 480V BUS BREAKERS - RELAY CHATTER	2.18E-02	2.20E-02	0.52	0.24	0.52	Functional	SOV
SEIS-RS-E-1ABCD	RECIRC SPRAY HEAT EXCHANGERS	1.58E-02	1.50E-02	2.01	0.24	0.32	Structural	CDFM
SEIS-EI-CB-MCR-PNL	SEISMIC FAILURE OF MCR BOARDS AND PANELS	1.42E-02	1.62E-02	1.30	0.24	0.38	Functional	CDFM
SEIS-BLDG-AB-LOWER	AUX BLDG LOWER FLOORS FAIL	1.42E-02	1.39E-02	2.05	0.24	0.26	Shear Wall Failure	CDFM
SEIS-MS-TV-111AB SEIS-MS-TV-211AB	MAIN STEAM TRIP VALVE TO TURBINE DRIVEN AFW PUMP	1.39E-02	3.18E-03	1.81 2.51	0.24	0.32	Functional	CDFM
SEIS-SSLOCA	SEISMIC-INDUCED SMALL-SMALL LOCA	1.37E-02	1.46E-02	0.30	0.28	0.28	Generic	EPRI Report [10]

Table 5.5-2 SLERF Importance Measures Ranked by FV

Fragility Groups	Fragility Group Description	U1 LERF FV	U2 LERF FV	Am	Br	Bu	Failure Mode	Fragility Method
SEIS-LLOCA	LARGE LOCA	1.34E-02	1.27E-02	2.50	0.30	0.40	Generic	EPRI Report [10]
SEIS-EG-B-3	EDG 1J Battery	1.32E-02	4.07E-03	1.15	0.24	0.38	Functional	CDFM
SEIS-EG-P-1J	EDG 1J Fuel Oil Transfer Pumps	1.29E-02	3.99E-03	1.16	0.24	0.38	Functional	CDFM
SEIS-MOV-QSPH-RSHX	MOVs in QUENCH SPRAY PUMP HOUSE - SW Cooling to RS HXs	1.19E-02	1.13E-02	2.13	0.24	0.32	Functional	CDFM
SEIS-VB-INV-1234	120 VAC VITAL BUS INVERTERS	1.19E-02	1.35E-02	1.10	0.19	0.58	Functional	SOV
SEIS-EP-CB-4ABCD	120 VAC VITAL BUS DISTRIBUTION PANELS	1.14E-02	1.28E-02	1.16	0.24	0.38	Anchorage	CDFM
SEIS-MLOCA	MEDIUM LOCA	1.13E-02	1.22E-02	2.00	0.35	0.45	Generic	EPRI Report [10]
SEIS-EE-BKR-HJ2-RLY	EDG OUTPUT BREAKERS - RELAY	1.05E-02	1.02E-02	0.77	0.24	0.49	Functional	SOV
SEIS-QS-TK-1	REFUELING WATER STORAGE TANK (RWST)	9.77E-03	1.10E-02	1.07	0.15	0.29	Tank Overturning	SOV
SEIS-RC-CNTRL-RODS	REACTOR CONTROL RODS	9.68E-03	1.09E-02	1.26	0.24	0.32	Failure of Fuel Hold Down Spring	CDFM
SEIS-EI-CB-202	EMERGENCY DIESEL GENERATOR CONTROL PANELS IN ESGR - Fails EDGs	8.83E-03	1.13E-02	1.40	0.24	0.38	Functional	CDFM
SEIS-EP-SS-1H1-1J1	480V LOAD CONTROL CENTERS 1H1 AND 1J1	8.16E-03	1.20E-02	1.22	0.24	0.38	Functional	CDFM
SEIS-EI-CB-201	EMERGENCY DIESEL GENERATOR CONTROL PANELS IN EDG ROOM - Fails EDGs	7.98E-03	9.96E-03	1.45	0.24	0.38	Anchorage	CDFM
SEIS-CH-P-1ABC-RLY	CHARGING PUMPS - RELAY CHATTER	7.47E-03	7.30E-03	0.77	0.24	0.49	Functional	SOV
SEIS-CV-TV-150ABCD	Containment Vacuum Isolation Trip Valves	7.11E-03	6.96E-03	2.51	0.24	0.32	Functional	CDFM

Table 5.5-2 SLERF Importance Measures Ranked by FV

Fragility Groups	Fragility Group Description	U1 LERF FV	U2 LERF FV	Am	Br	Bu	Failure Mode	Fragility Method
SEIS-EDG-HJ-RLY	EMERGENCY DIESEL GENERATORS - RELAY CHATTER	7.08E-03	7.13E-03	0.70	0.24	0.83	Functional	SOV
SEIS-EP-SS-1H-1J	480V LOAD CONTROL CENTERS 1H AND 1J	6.87E-03	1.75E-02	0.99	0.24	0.38	Functional	CDFM
SEIS-BY-B-1-24	STATION BATTERIES 1-II AND 1-IV	6.70E-03	8.21E-03	1.14	0.24	0.38	Structural Failure of Rack	CDFM
SEIS-SW-P-1AB-RLY	SERVICE WATER PUMPS - RELAY CHATTER	6.21E-03	5.97E-03	0.77	0.24	0.49	Functional	SOV
SEIS-EP-SW-1H-1J	4160V EMERGENCY BUSES	6.19E-03	5.85E-03	1.13	0.24	0.33	Combined Structural / Function	CDFM
SEIS-EG-B-4	EDG 2J Battery	6.09E-03	1.71E-02	0.97	0.24	0.38	Functional	CDFM
SEIS-EG-P-2J	EDG 2J Fuel Oil Transfer Pumps	5.78E-03	1.64E-02	1.00	0.24	0.38	Functional	CDFM
SEIS-EG-B-1	EDG 1H Battery	5.69E-03	2.24E-03	1.49	0.24	0.33	Combined Structural / Function	CDFM
SEIS-FW-P-3AB	MOTOR-DRIVEN AUXILIARY FEEDWATER PUMPS	5.67E-03	4.06E-03	1.62	0.24	0.32	Functional	CDFM
SEIS-BLDG-AB-UPPER	AUX BLDG UPPER FLOORS FAIL	5.40E-03	5.04E-03	1.02	0.24	0.26	Failure of Steel Superstructure	CDFM
SEIS-RS-P-2AB-RLYLO	Outside RS Pumps Fail to Start due to Lockout Relay	5.09E-03	4.64E-03	0.77	0.24	0.49	Functional	SOV
SEIS-EI-CB-PROCESS	PLANT PROCESS CABINETS	4.78E-03	5.19E-03	1.91	0.19	0.55	Functional	SOV
SEIS-BDB-DB-123	BEYOND DESIGN BASIS (FLEX) DISTRIBUTION PANELS	4.10E-03	5.71E-03	1.10	0.24	0.26	Seismic Interaction	CDFM
SEIS-EG-B-2	EDG 2H Battery	3.73E-03	1.21E-02	1.22	0.24	0.33	Combined Structural / Function	CDFM
SEIS-EG-P-2H	EDG 2H Fuel Oil Transfer Pumps	3.03E-03	1.05E-02	1.40	0.24	0.38	Functional	CDFM

The most significant non-seismic SSC failures (e.g., random failures of modeled components during the SPRA mission time) are listed in Table 5.5-3. The unavailability of the diesel-driven fire pump has the highest FV for both Units 1 and 2. As noted in the important non-seismic failures for SCDF, the diesel-driven fire pump is important for long term supply to the turbine-driven AFW when the ECST depletes during a SBO. The other non-seismic failures that are important for SLERF are SW supply headers and SW pumps that fail cooling to the RS heat exchangers and thus fails containment heat removal. The EDGs and FLEX equipment are also important for mitigating SBO sequences.

Table 5.5-3 SLERF Importance Measures Ranked by FV for Non-Seismic Failures

Unit 1 Model Basic Events	Prob	SLERF FV	Description
1FP-DDP--TM-2	3.16E-02	5.14E-03	DIESEL-DRIVEN FIRE PUMP 1-FP-P-2 OUT OF SERVICE FOR TEST OR MAINTENANCE
0SW-HDR--TM-B	1.52E-02	2.95E-03	B SW HEADER IN OUT OF SERVICE FOR TEST OR MAINTENANCE
0BDBEDG--FR-1A-FLEX	2.04E-02	2.73E-03	FLEX DIESEL GENERATOR FAILS TO RUN
0SW-HDR--TM-A	1.52E-02	1.99E-03	A SW HEADER IN OUT OF SERVICE FOR TEST OR MAINTENANCE
1SW-PAT--TM-1B	8.55E-03	1.43E-03	U1 1B SW PUMP OUT OF SERVICE FOR TEST OR MAINTENANCE
1EE-EDG--FR-1H	2.79E-02	1.29E-03	U1 H DIESEL GENERATOR FAILS TO RUN
1EE-EDG--FR-1J	2.79E-02	1.14E-03	U1 J DIESEL GENERATOR FAILS TO RUN
1SW-PAT--TM-1A	8.55E-03	1.01E-03	U1 1A SW PUMP OUT OF SERVICE FOR TEST OR MAINTENANCE
1EE-EDG--TM-1H	2.25E-02	9.13E-04	U1 H DIESEL GENERATOR OUT OF SERVICE FOR TEST OR MAINTENANCE
1EE-EDG--TM-1J	2.25E-02	8.65E-04	U1 J DIESEL GENERATOR OUT OF SERVICE FOR TEST OR MAINTENANCE
Unit 2 Model Basic Events and FV Importance			
1FP-DDP--TM-2	3.16E-02	5.25E-03	DIESEL-DRIVEN FIRE PUMP 1-FP-P-2 OUT OF SERVICE FOR TEST OR MAINTENANCE
0BDBEDG--FR-1A-FLEX	2.04E-02	2.67E-03	FLEX DIESEL GENERATOR FAILS TO RUN
0SW-HDR--TM-A	1.52E-02	2.62E-03	A SW HEADER IN OUT OF SERVICE FOR TEST OR MAINTENANCE
2EE-EDG--FR-2H	2.79E-02	2.31E-03	U2 H DIESEL GENERATOR FAILS TO RUN
2EE-EDG--TM-2H	2.25E-02	1.79E-03	U2 H DIESEL GENERATOR OUT OF SERVICE FOR TEST OR MAINTENANCE
0SW-HDR--TM-B	1.52E-02	1.55E-03	B SW HEADER IN OUT OF SERVICE FOR TEST OR

Table 5.5-3 SLERF Importance Measures Ranked by FV for Non-Seismic Failures			
Unit 1 Model Basic Events	**Prob**	**SLERF FV**	**Description**
			MAINTENANCE
2EE-EDG--FR-2J	2.79E-02	1.27E-03	U2 J DIESEL GENERATOR FAILS TO RUN
2SW-PAT--TM-1B	8.55E-03	1.12E-03	U2 1B SW PUMP OUT OF SERVICE FOR TEST OR MAINTENANCE
2EE-EDG--TM-2J	2.25E-02	9.71E-04	U2 J DIESEL GENERATOR OUT OF SERVICE FOR TEST OR MAINTENANCE
2QS-PSB--FS-1A	5.59E-03	8.63E-04	U2 1A QS PUMP FAILS TO START

A summary of the SLERF results for each seismic hazard interval is presented in Table 5.5-4. Figure 5.5-1 shows a bar chart of the unit 1 SLERF as a function of PGA (Unit 2 results are the same as Unit 1). The seismic ground motions that contribute the most to SLERF are in the 1.0 to 2.5g range (%G08 and %G09) which is generally the case in SPRA LERF results.

5.5-4 Contribution to SLERF by Acceleration Interval								
	PGA	**Initiator Frequency**	**U1 LERF**	**% Total U1 LERF**	**U1 CLERP**	**U2 LERF**	**% Total U2 LERF**	**U2 CLERP**
%G01	0.06g to <0.3g	9.21E-04	7.58E-10	0.00%	0.00	7.58E-10	0.00%	0.00
%G02	0.3g to <0.4g	5.34E-05	5.77E-08	0.37%	0.00	5.69E-08	0.36%	0.00
%G03	0.4g to <0.5g	3.01E-05	1.35E-07	0.87%	0.00	1.31E-07	0.84%	0.00
%G04	0.5g to <0.6g	1.79E-05	2.75E-07	1.77%	0.02	2.68E-07	1.72%	0.01
%G05	0.6g to <0.7g	1.11E-05	3.71E-07	2.39%	0.03	3.65E-07	2.34%	0.03
%G06	0.7g to <0.8g	7.08E-06	4.96E-07	3.19%	0.07	5.03E-07	3.23%	0.07
%G07	0.8g to <1g	8.26E-06	1.60E-06	10.29%	0.19	1.63E-06	10.45%	0.20
%G08	1g to <1.5g	9.09E-06	6.89E-06	44.29%	0.76	6.91E-06	44.31%	0.76
%G09	1.5g to <2.5g	4.25E-06	4.25E-06	27.32%	1.00	4.25E-06	27.25%	1.00
%G10	>2.5g	1.48E-06	1.48E-06	9.51%	1.00	1.48E-06	9.49%	1.00

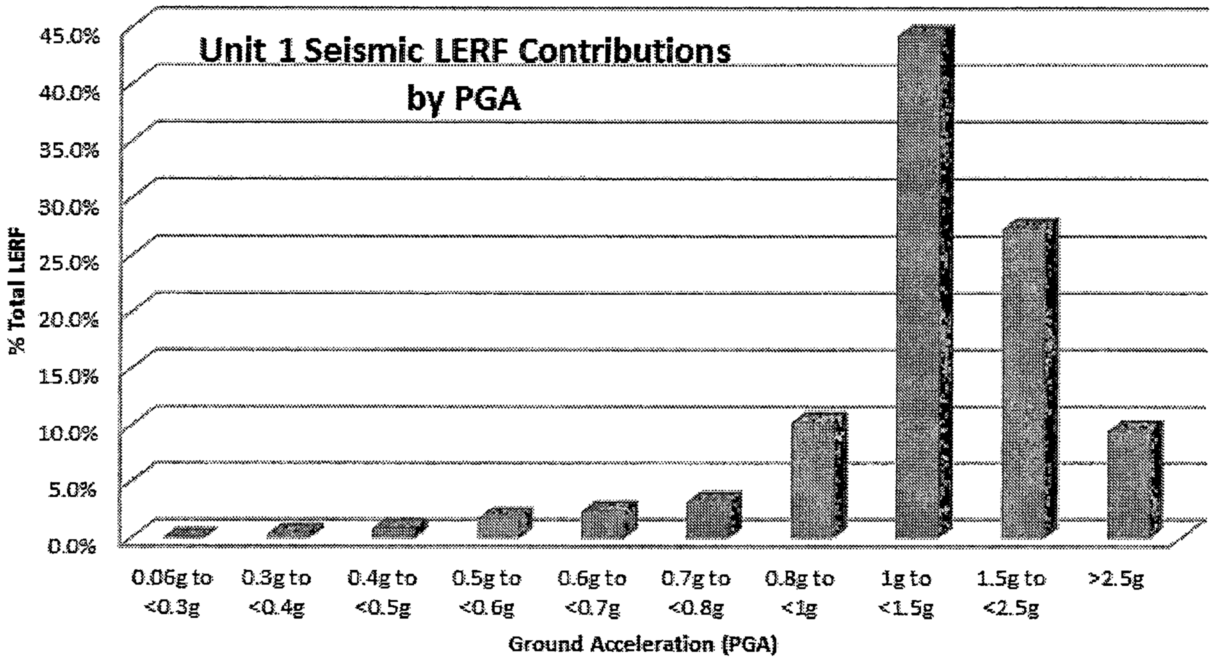


Figure 5.5-1 Unit 1 SLERF Contributions by PGA

The most significant Operator actions for SLERF are listed in Table 5.5-5. As discussed in Section 5.1, the seismic PRA models each Operator action using four Human Error Probability (HEP) basic events per action, which model different failure probabilities for the four damage states. The FV importance of the Operator action is the sum of the FV importance for each of the four HEP basic events. The important actions involve depressurizing the RCS after core damage per the SAMGs. Other important actions are mainly important for mitigating core damage, such as aligning the turbine-driven AFW pump to the other SGs, initiating Bleed and Feed, and performing FLEX mitigating actions (battery load shed and installing RCS injection pump).

5.5-5 - SLERF Importance Measures Ranked by FV for Operator Actions

HEP Basic Event	SLERF FV	Description
HEP-C-RCSDEP	2.71E-02	Depressurize the RCS Per SAMGs
HEP-C-ALIGN-TDAFW	2.26E-02	Align turbine-driven AFW Pump to B and C SGs
HEP-C-1BAFE	1.17E-02	Initiate Bleed and Feed After AFW Fails
HEP-C-FLEX-RIP	8.28E-03	Install and Start FLEX RCS Injection Pump

5.5-5 - SLERF Importance Measures Ranked by FV for Operator Actions		
HEP Basic Event	**SLERF FV**	**Description**
HEP-C-1HV-SFGD-VENT	6.77E-03	Restore Safeguards Area Ventilation
REC-SEIS-FLD-CCHX	5.92E-03	Isolate SW Flood in Auxiliary Building Caused by Failure of the Component Cooling Heat Exchangers
HEP-C-FLEX-LOADSHED	5.23E-03	Load shed the vital 125vdc batteries during SBO
HEP SLERF FV Importance in Unit 2 Model		
HEP-C-RCSDEP	2.56E-02	Depressurize the RCS Per SAMGs
HEP-C-ALIGN-TDAFW	2.34E-02	Align turbine-driven AFW Pump to B and C SGs
HEP-C-2BAFE	9.27E-03	Initiate Bleed and Feed After AFW Fails
HEP-C-FLEX-RIP	8.31E-03	Install and Start FLEX RCS Injection Pump
HEP-C-2HV-SFGD-VENT	6.73E-03	Restore Safeguards Area Ventilation
REC-SEIS-FLD-CCHX	5.72E-03	Isolate SW Flood in Auxiliary Building Caused by Failure of the Component Cooling Heat Exchangers
HEP-C-FLEX-LOADSHED	5.52E-03	Load shed the vital 125vdc batteries during SBO

5.6 SPRA Quantification Uncertainty Analysis

This section documents the parametric uncertainty analysis and the approach used to identify sources of model uncertainty.

Parametric Uncertainty

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 through 5.6-4 show the curves of cumulative probability and probability density function.

Table 5.6-1 – Seismic CDF and LERF Uncertainty Distributions

	Unit 1 CDF	Unit 2 CDF	Unit 1 LERF	Unit 2 LERF
Mean	6.32E-05	6.34E-05	1.93E-05	1.94E-05
5th Percentile	1.04E-05	1.03E-05	3.00E-06	2.91E-06
Median	4.32E-05	4.30E-05	1.30E-05	1.28E-05
95th Percentile	1.81E-04	1.84E-04	5.58E-05	5.71E-05
StdDev	6.77E-05	6.83E-05	2.08E-05	2.24E-05
Skewness	4.4	4.7	3.9	5.5

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 1,000 cutsets using ACUBE. The distribution for both SCDF and SLERF appears generally uniform. The distribution (i.e. spread between 5th and 95th) for SCDF and SLERF is larger than that of the internal events uncertainty distribution, which is expected and reasonable given relatively large uncertainties in the seismic hazard curves and SSC fragility curves.

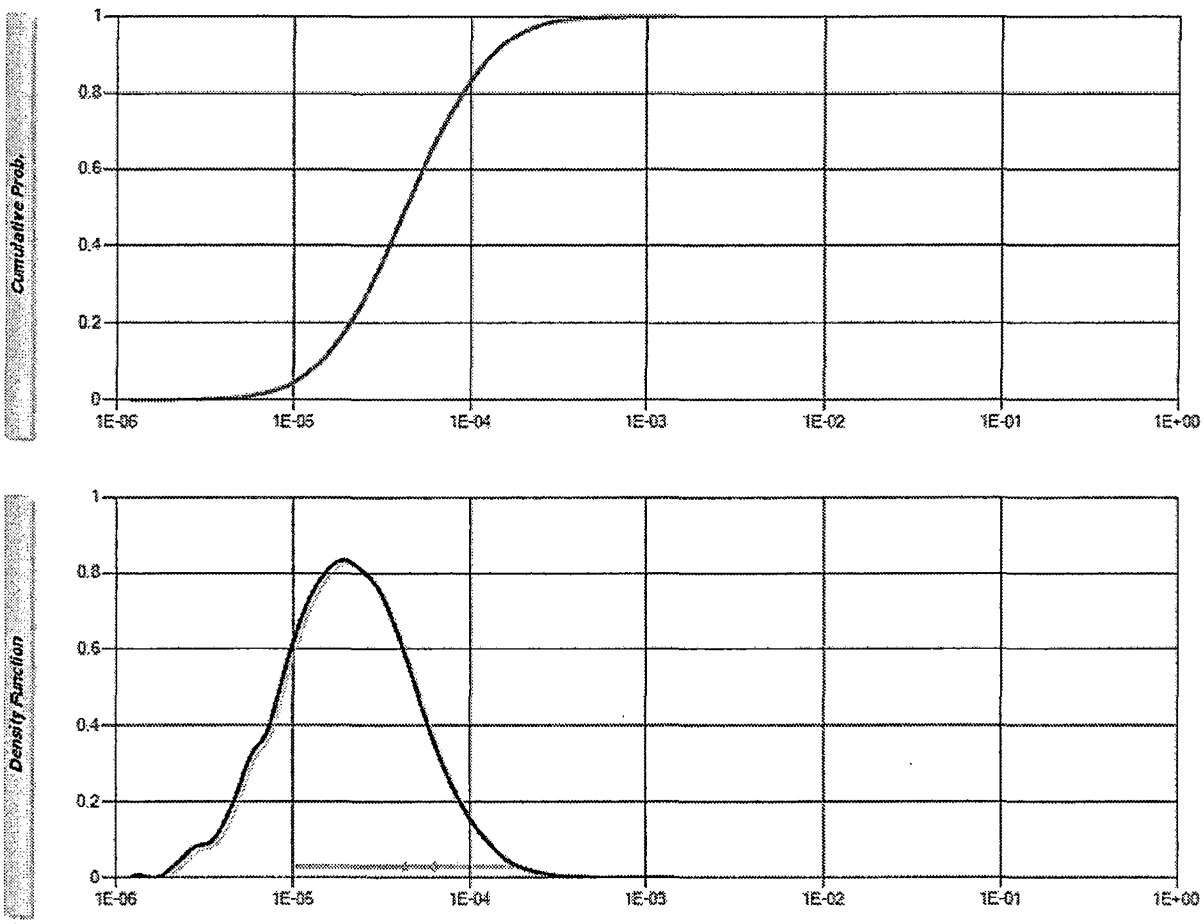


Figure 5.6-1 – Unit 1 Seismic CDF Cumulative and Density Distribution Functions

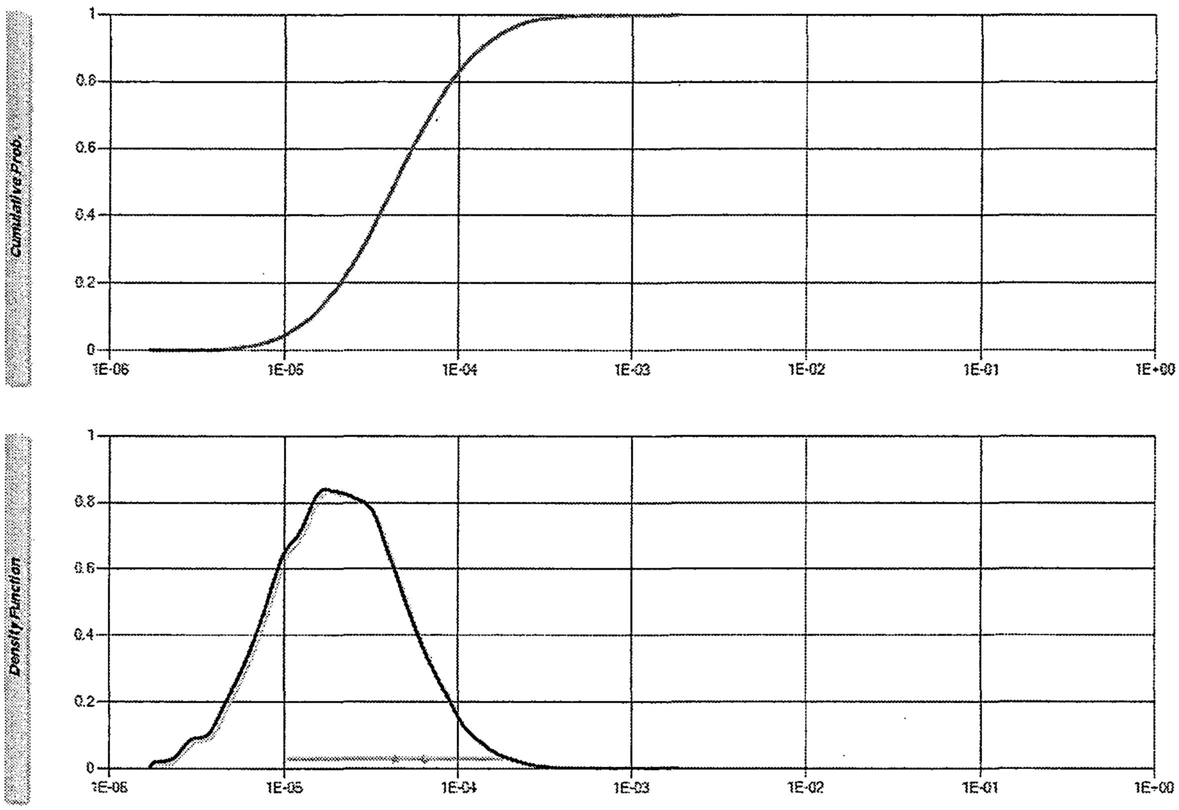


Figure 5.6-2 – Unit 2 Seismic CDF Cumulative and Density Distribution Functions

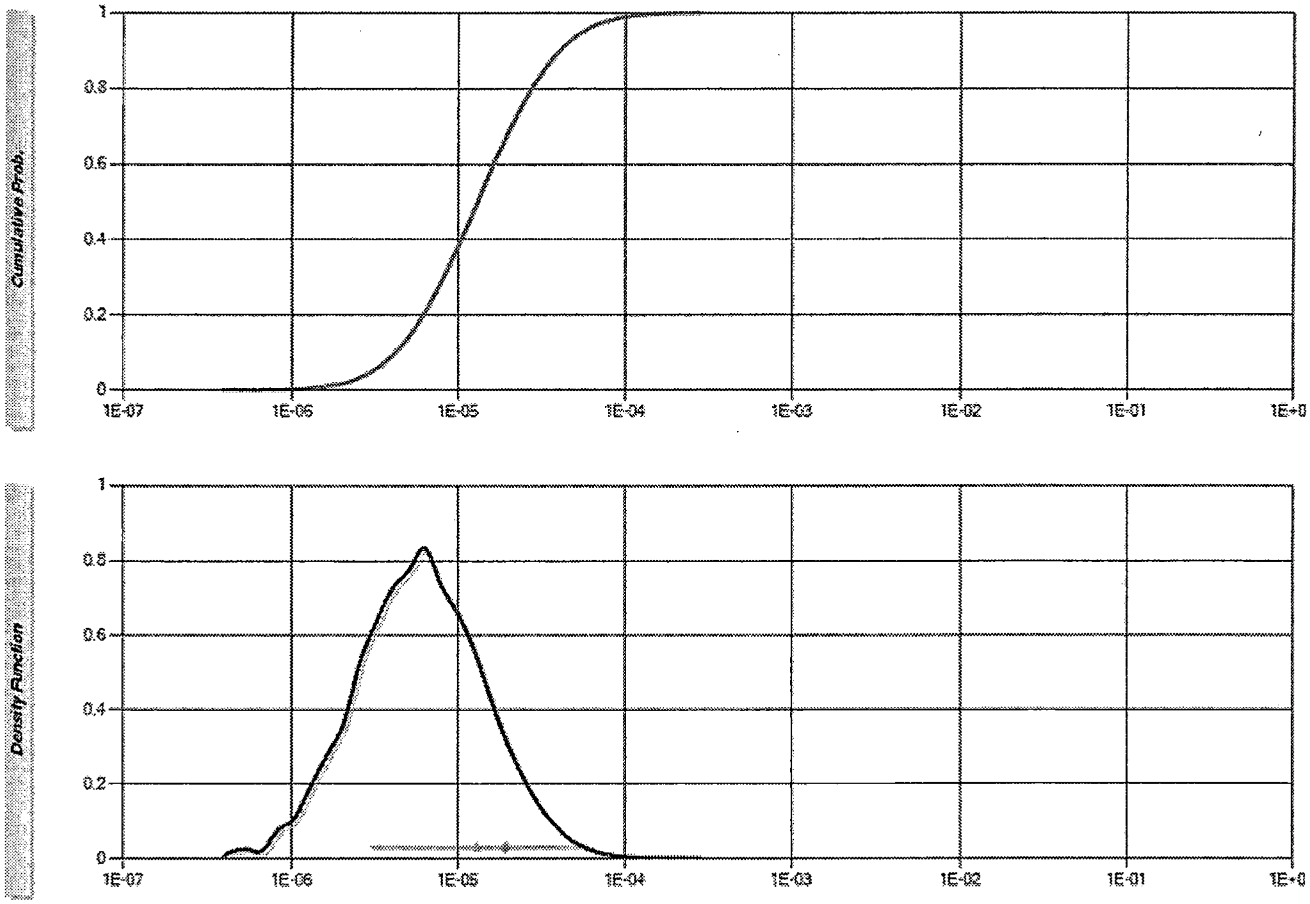


Figure 5.6-3 – Unit 1 Seismic LERF Cumulative and Density Distribution Functions

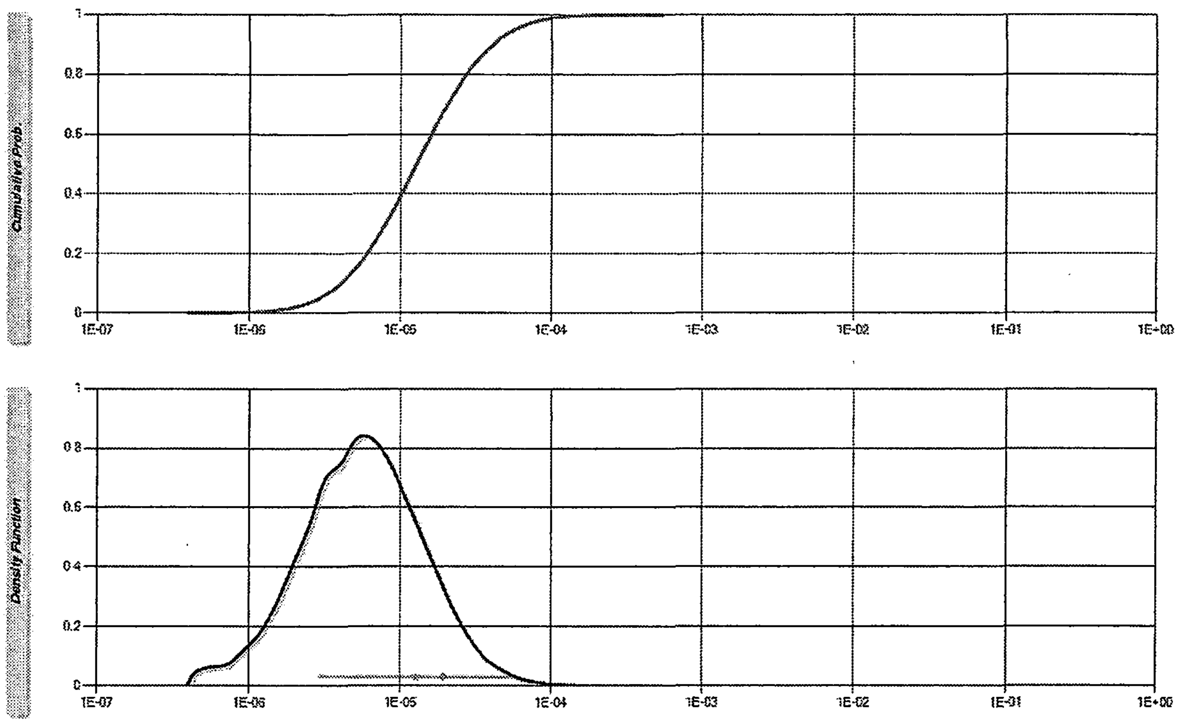


Figure 5.6-4 – Unit 2 Seismic LERF Cumulative and Density Distribution Functions

Model Uncertainty

Model uncertainty relates to the uncertainty associated with some aspect of a PRA model that can be represented by any one of several different modeling approaches. Consequently, uncertainty is introduced into the PRA results since there may not be consensus about which model approach most appropriately represents the particular aspect of the plant being modeled. The uncertainty associated with a model and its constituent parts is typically addressed by making assumptions.

The guidance provided in EPRI reports in references [10] and [19] were used in the identification and characterization of sources of model uncertainty and related assumptions. A generic list of uncertainty sources for seismic PRAs is contained in Appendix C of reference [19]. In addition to the generic sources of uncertainty, the assumptions made in the NAPS SPRA were also evaluated for sources of uncertainty. Since many of the assumptions are considered reasonable and consistent with standard industry practices, only the assumptions that may involve a significant source of uncertainty were identified for potential sensitivities studies. The sources of uncertainty that were identified for further evaluation are discussed in the next section (5.7) for sensitivity studies.

5.7 SPRA Quantification Sensitivity Analysis

As discussed in Section 5.6, various sources of model uncertainties were reviewed and examined to identify sources that may have a significant impact on the SCDF and SLERF. The following sensitivity studies were performed to evaluate how the SCDF and SLERF are impacted by model assumptions, simplifications and uncertainties:

- Model Truncation and Convergence
- Relay Chatter
- Small-Small LOCA
- FLEX Credited in SBO
- Mission Time
- Building HCLPFs
- HEP to Isolate Flood from CC HX Failure
- HEPs at 5^{th} and 95^{th} Percentile
- LERF

5.7.1 Model Truncation and Convergence

The baseline SPRA was quantified at 1E-09 for SCDF and 1E-10 for SLERF. Model convergence per the criteria in the PRA Standard was achieved at these levels.

5.7.2 Relay Chatter

As part of the development of the North Anna SPRA, detailed circuit analyses were performed to identify relays that could impact SSC function if chatter occurs. The North Anna SPRA includes over 20 fragility groups that model relay chatter. The HCLPF capacity of many of the relays is relatively low, which results in loss of some key mitigating functions. These relay fragility groups show up as significant contributors to SCDF and SLERF (based on FV importance) partly due to not crediting Operator action to reset the relays to restore the mitigating functions. This sensitivity was performed to determine the reduction in SCDF and SLERF if all relays were assumed to not be vulnerable to chatter. A reduction of approximately 28% was realized in SCDF and 15% reduction for SLERF for both Unit 1 and 2.

The sensitivity shows the SCDF and SLERF results are impacted by the modeling of relay chatter. The HCLPF capacities of the relays are considered reasonable since they were developed using the current industry methods (separation of variables fragility method) for relay fragilities. The SPRA does not credit Operator action to reset the relays mainly due to the time required to investigate the lockout condition before restoring the breakers. Therefore, no further refinement to the SPRA was made.

5.7.3 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 (SSLOCA). For NAPS, detailed walkdowns of the RCS were not performed to develop a fragility for the small-bore piping and instrument tubing that connects to the RCS. Therefore, SSLOCA is included in the SPRA with a HCLPF equal to the Safe Shutdown Earthquake of 0.12g per the guidance in the EPRI SPRA Implementation Guide [10]. In this sensitivity, the HCLPF was increased to that of the small LOCA (SLOCA) HCLPF, which is 0.32g. The results showed the SCDF decreased by approximately 8% and slightly less than 1% for SLERF for both Units 1 and 2. Even though the SCDF may be reduced by 8%, this is not considered significant enough reduction to warrant changing the model.

5.7.4 FLEX Credited in SBO Sequences

The SPRA does not credit recovery of offsite power given the possible damage to the offsite power sources and the likely long repair times to restore power. The SPRA does credit FLEX mitigating strategies in the SBO sequences to restore and maintain safety functions to prevent core damage as long as there is sufficient time available to install and start the FLEX equipment. FLEX is not credited for sequences where there is insufficient time to implement the FLEX equipment before core damage. For example, FLEX is not credited if there is a large, medium, or small LOCA, or a large RCP seal LOCA coincident with the SBO since core damage would occur before the FLEX strategies could be implemented. Also, if the TDAFW pump fails to start and run, FLEX is not credited as there would be insufficient time before SG dryout. The probabilities for the FLEX HEPs and the FLEX equipment failures were identified as sources of uncertainty. Two sensitivities were performed to assess the impact of these uncertainties.

<u>FLEX HEP Probabilities</u>

The SPRA credits FLEX for mitigating seismic-induced SBO. Given the unique nature of the FLEX mitigating strategies as compared with standard actions in the EOPs, the uncertainties associated with the FLEX actions were evaluated. The FLEX mitigating actions modeled are:

- Load shedding the vital 125vdc station batteries to extend battery life
- Installing generators to power vital buses before batteries deplete
- Installing RCS Injection pump to makeup to RCS
- Refuel FLEX engine-driven SSCs

The Human Reliability Analysis (HRA) for these actions followed the guidance in EPRI report 3002008093 [18]. The HRA for these actions did include some judgements with respect to using surrogates for estimating the failure probabilities of actions unique to FLEX strategies, such as transporting the FLEX equipment from the FLEX storage building. This sensitivity evaluates the impact on the SCDF and SLERF if these HEPs were increased by a factor of 5 to account for the selection of different commission errors. The results show the SCDF increased by approximately 7% and SLERF increased by approximately 1% for both units. The use of surrogates for the execution error actually only increases the HEP probability by less than 15% if different (higher) probabilities are used in the HRA. So using a factor of 5 for this sensitivity is considered conservative for assessing the impact of using surrogates. This sensitivity also provides insight that the SPRA risk is not significantly impacted by changes in the FLEX HEP probabilities. No further refinement to the SPRA is considered necessary.

FLEX Equipment Reliability

This sensitivity evaluates the impact if the reliability of the FLEX equipment is less than assumed in the SPRA. The following FLEX equipment is credited for maintaining power to the critical instrumentation and for RCS makeup:

- FLEX 120VAC Portable Generator; Used to repower the vital buses to maintain critical instrumentation (0-BDB-GEN-1A)
- Portable RCS Injection Pump; Used to makeup to the RCS (0-BDB-P-3A)

The failure probabilities used for the FLEX equipment are based on similar installed equipment (e.g. EDGs) for now until sufficient reliability data is available for the FLEX equipment. These failure probabilities for the FLEX equipment are not expected to be significantly different than the failure probabilities of the actual portable equipment. However, since there is uncertainty in these failure probabilities, this sensitivity evaluates the impact if the failure probabilities are increased by a factor of 5. The results show the SCDF increased by less than 5% and SLERF increased by approximately 1% for both units. The FLEX equipment failure probabilities are not expected to increase by a factor of 5. The failure probabilities used in the SPRA are considered reasonable and the SCDF and SLERF are not significantly impacted by changes in the equipment probabilities. Therefore, no further refinement to the SPRA is considered necessary.

5.7.5 Mission Time

The mission time assumed in the SPRA is 24 hours. In this sensitivity, the mission time is changed to 72 hours. The results show the SCDF and SLERF increased by approximately 2%. Extending the mission time to longer than 24 hours does not have much of an impact on the SCDF and SLERF. The majority of the SSC failures are due to seismic damage and not random failures of the SSCs during the mission time. No further refinement to the SPRA is considered necessary.

5.7.6 Building HCLPFs

The SPRA assumes the HCLPF capacity of buildings represents gross failure of the building such that all SSCs in the building are failed. This is a very conservative assumption, as the reported HCLPF value often corresponds to a local failure, for which the majority of SSCs in the building will survive and not result in a complete or "gross" failure condition. Given the uncertainty in how the buildings fail, this sensitivity evaluates the impact on SCDF and SLERF if the HCLPF capacity of the buildings is increased, which would represent a higher HCLPF capacity that would result in gross failure of the building. The buildings evaluated in this sensitivity are all reinforced concrete, missile protected structures that will be assumed to have a HCLPF capacity of 3.0g in this sensitivity. A HCLPF of 3.0g is selected since it provides a reasonable estimate for the gross failure of the buildings without being overly optimistic. The results show less than 1% decrease in SCDF and approximately 10% decrease in SLERF. The SCDF is not particularly sensitive to these building failures. SLERF decreased mainly due to the reactor containment, whose failure is direct LERF, and due to failure of the Service Water Pump House and Service Water Valve House, which results in loss of containment heat removal. No further refinement to the SPRA is considered necessary.

5.7.7 Isolating Service Water Flood

Seismic failure of the Component Cooling (CC) heat exchangers was determined to result in a Service Water flood that could impact the Charging pumps if the flood was not mitigated in time. This flood scenario is modeled in the SPRA and the HEP for mitigating the flood is set to 1 because of the uncertainty of the size of the flood given that the four heat exchangers are assumed to be 100% correlated.

The SPRA models failure of the four CC heat exchangers in the Auxiliary building as a major flood due to failure of the SW piping that connects to the heat exchangers. There is uncertainty on the size of the flood and the flow rate from the pipe breaks. The model assumes the flood flow rate is large enough such that there is little time available to diagnose and isolate the flood before it damages the Charging pumps given failure of all four heat exchangers (assumed correlated). Therefore, the HEP for isolating the flood is set to 1. In this sensitivity, it was assumed that the flood rate from the SW lines of the four CC heat exchangers is low enough such that there is time available for Operators to isolate the breaks (i.e. the breaks are not complete guillotine breaks, but are splits in the pipe nozzles). The HEP probabilities for the four HRA bins were assumed to vary from 5E-03 to 1E-01, which are considered reasonable estimates for isolating lower flow rate floods. The results show very little reduction in SCDF and SLERF, less than 1%, if the seismic failure of the CC heat exchangers is assumed to result in a low enough SW flood flow rate to allow crediting isolation of the flood. No further refinement to the SPRA is considered necessary.

5.7.8 HEP Probabilities

This sensitivity evaluates the impact of the HEPs credited in the SPRA. There is uncertainty in the development of the adjustments to the HEPs in the model to account for the various impacts on the Operators taking mitigating actions after a seismic event. This sensitivity quantifies the SPRA with the HEPs set to their 95^{th} and to their 5^{th} percentile probabilities. The results show that increasing the HEPs to their 95^{th} percentile results in less than a 10% increase in SCDF and approximately 2.5% increase in SLERF. If the HEPs were reduced to their 5^{th} percentile, the SCDF decreases approximately 5% and the SLERF decreases less than 2%. The results indicate that the model is not overly sensitive to the HEP probabilities. Therefore, no further refinement to the SPRA is considered necessary.

5.7.9 Delay Evacuation Impact on LERF

This sensitivity evaluates the impact of delayed evacuations caused by damage to surrounding infrastructure (e.g. bridges, communication towers). The delayed evacuations results in LERF sequences that were previously screened out because they are not early releases that should be included in the LERF as releases before evacuations take place. A simplified approach was used in this sensitivity where all seismic events with magnitude >0.5g result in sufficient delay in the evacuation time such that they are modeled as leading directly to the LERF end state. The results show that SLERF increases by a factor of 3.2. Two other cases were evaluated where all seismic events >0.6g and >1g were assumed to result in SLERF. The results showed increases in SLERF by a factor of 2.5 and 1.1 for >0.6g and >1.0g, respectively.

This sensitivity used a very simplified approach for estimating the impact on SLERF due to delays in evacuations since not all SCDF sequences at the elevated ground motions would result in direct LERF. There is uncertainty in what size seismic events could significantly impact the surrounding infrastructure and thus cause delays in evacuations. The sensitivity shows there may be some impact on SLERF if the infrastructure is impacted at lower ground motions. However, due to the conservative approach used in the sensitivity, no further refinements to the SPRA are considered necessary.

5.7.10 SPRA Logic Model and Quantification Technical Adequacy

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

6.0 Conclusions

A seismic PRA has been performed for North Anna Power Station Units 1 and 2 in accordance with the guidance in the PRA Standard [4] and the SPID [2]. The SPRA shows that the point estimate seismic CDF is 6.0×10^{-5}/yr and the seismic LERF is 1.6×10^{-5}/yr for both units. The PRA model provides insights and identifies the most important equipment relied upon for responding to a seismic event. No seismic hazard vulnerabilities were identified.

The SPRA as described in this submittal reflects the as-built/as-operated North Anna Power Station Units 1 and 2 as of the SPRA freeze date - January, 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.

7.0 References

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[7] EPRI NP 6041-SL, *A Methodology for Assessment of Nuclear Power Plant Seismic Margin*, Rev. 1. , Electric Power Research Institute, Palo Alto, CA, August 1991.

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[15] Virginia Electric and Power Company Letter, "North Anna Power Station Units 1 and 2 Response to March 12, 2012 Information Request – Spent Fuel Pool Seismic Evaluation for Recommendation 2.1," dated December 14, 2017.

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8.0 Acronyms

AB	Auxiliary Building
AC	Alternating Current
AFW	Auxiliary Feedwater
ANS	American Nuclear Society
AOD	Air Operated Damper
AOV	Air Operated Valve
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient without Scram
BE	Best Estimate
CCDP	Conditional Core Damage Probability
CDF	Core Damage Frequency
CDFM	Conservative Deterministic Failure Margin

CEUS	Central and Eastern United States
CLERP	Conditional Large Early Release Probability
CO_2	Carbon Dioxide
CST	Condensate Storage Tank
DC	Direct Current
ECC-AM	Extended Continental Crust—Atlantic Margin
ECST	Emergency Condensate Storage Tank
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
ESEP	Expedited Seismic Evaluation Program
ESF	Engineered Safeguards Features
ESGR	Emergency Switchgear Room
FEM	Finite Element Model
FIRS	Foundation Input Response Spectra
FLEX	Diverse and Flexible Mitigation Strategies
FPIE	Full Power Internal Events
FV	Fussell-Vesely
GMPE	Ground Motion Prediction Equation
GMRS	Ground Motion Response Spectra
IPEEE	Individual Plant Examination for External Events
HCLPF	High Confidence of a Low Probability of Failure
HEP	Human Error Probability
HF	High Frequency
HHSI	High Head Safety Injection
HRA	Human Reliability Analysis
HVAC	Heating, Ventilation, and Air Conditioning
ISRS	In-Structure Response Spectrum
LB	Lower Bound
LCC	Load Control Center

LERF Large Early Release Frequency

LHSI Low-Head Safety Injection

LMSM Lumped Mass Stick Model

LOCA Loss of Coolant Accident

LOOP Loss of Offsite Power

MAFE Mean Annual Frequency of Exceedance

MCC Motor Control Center

MCR Main Control Room

MESE-N Mesozoic and younger extended prior – narrow

MOD Motor Operated Damper

MOV Motor Operated Valve

MSVH Main Steam Valve House

NAPS North Anna Power Station

NEI Nuclear Energy Institute

NMSZ New Madrid Seismic Zone

NRC Nuclear Regulatory Commission

NSSS Nuclear Steam Supply System

NTTF Near Term Task Force

PGA Peak Ground Acceleration

PORV Power Operated Relief Valve

PRA Probabilistic Risk Assessment

PSHA Probabilistic Seismic Hazard Analysis

QS Quench Spray

RCB Reactor Containment Building

RCP Reactor Coolant Pump

RCS Reactor Coolant System

RLME Repeated Large Magnitude Earthquake

RPS Reactor Protection System

RS Recirculation Spray

RWST Refueling Water Storage Tank

SAMG	Severe Accident Management Guidelines
SB	Service Building
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	Safeguards Building, Steam Generator
SHA	Seismic Hazard Analysis Element Within ASME/ANS PRA Standard
SHS	Seismic Hazard Submittal
SHSR	Seismic Hazard and Screening Report
SI	Safety Injection
SLERF	Seismic Large Early Release Frequency
SMA	Seismic Margin Assessment
SOV	Solenoid Operated Valve, 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
SRSS	Square Root Sum of the Squares
SRT	Seismic Review Team
SSC	Structure, System or Component
SSE	Safe Shutdown Earthquake
SSEL	Safe Shutdown Equipment List
SSI	Soil Structure Interaction
SSLOCA	Small-small LOCA
SSSI	Structure-Soil-Structure Interaction
SW	Service Water
SWPH	Service Water Pump House
SWVH	Service Water Valve House

T-H	Time History
UB	Upper Bound
UHS	Ultimate Heat Sink
USI	Unresolved Safety Issue
V/H	Vertical to Horizontal

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 why the SPRA is technically adequate for the 50.54(f) response.

The NAPS SPRA was subjected to an independent peer review against the pertinent requirements in Part 5 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, 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 plant response) elements for seismic CDF and LERF. The peer review therefore addressed the set of SRs identified in Tables 6-4 through 6-6 of the SPID [2].

The NAPS SPRA peer review was conducted during the week of July 17, 2017 at the Dominion Energy Innsbrook Technical Center offices in Glen Allen, Virginia. As part of the peer review, a walk-down of portions of NAPS Units 1 & 2 was performed on July 18, 2017 by selected members of the peer review team.

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 fifth day.

The SPRA peer review process defined in [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 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 [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 review was conducted by Dr. Andrea Maioli of Westinghouse, Dr. Martin McCann of Jack Benjamin & Associates, Dr. Glenn Rix of Geosyntec Consultants, Dr. James J. Johnson of James J. Johnson and Associates, Mr. Frederic Grant of Simpson Gumpertz & Heger, Mr. Benny Ratnagaran of Southern Nuclear Operating Company, Dr. Jonathan Lucero of Arizona Public Services and Mr. Edmond Wiegert of Duke Energy. Appendix D contains the resumes for the reviewers. The team was assembled by the peer review team lead. The lead and reviewer qualifications have been reviewed by Dominion and have been confirmed to be consistent with requirements in the ANS/ASME PRA Standard and the guidelines of NEI-12-13.

Consistent with the requirement in Section 1-6.2.2 of the ASME/ANS PRA Standard [4], the members of the peer review team were independent of the North Anna Units 1 & 2 PRA. They were not involved in performing or directly supervising work on any element evaluated in the overall North Anna Units 1 & 2 seismic PRA.

Dr. Andrea Maioli, the team lead, has over 10 years of experience at Westinghouse in the nuclear safety area generally and PRA specifically for both existing and new nuclear power plants. He is the technical lead for all seismic PRA activities with Westinghouse. He has supported and led peer reviews for internal events, internal flooding, fire PRAs, high winds and other external hazards as well as seismic PRAs and is a member of the ASME/ANS JCNRM and of the JCNRM Subcommittee on Standard Maintenance, which is maintaining the ASME/ANS PRA Standard.

Dr. Martin McCann was the lead for the review of the Seismic Hazard Analysis (SHA) technical element. He has over 35 years of experience in engineering seismology including site response analysis and specification of ground motion. Dr. McCann has served as SHA lead reviewer for a number of recent SPRAs. He was assisted in the hazard review by Dr. Glenn Rix, who has more than 25 years of experience in the areas of geotechnical earthquake engineering and engineering seismology (particularly for the eastern and central U.S.), seismic hazard assessment and risk mitigation for civil infrastructure including dams and power plants, and advanced near-surface geophysics investigations and interpretations across a range of applications. Dr. Rix also served as reviewer for multiple recent SPRAs peer reviews.

Dr. James Johnson, the lead reviewer for the SFR technical element, is an independent contractor with more than 40 years of experience mainly in the area of structural and engineering mechanics. He has been involved SPRAs for 35 nuclear power plants as well as in numerous peer reviews. He was assisted in fragility review by Mr. Frederic Grant and Mr. Benny Ratnagaran. Mr. Grant has 11 years of structural mechanics engineering experience, the majority of which has been in the commercial and government nuclear industries. His work in the nuclear industries involves seismic probabilistic risk assessments, seismic fragility analysis, seismic margin assessments, experience-based seismic qualification methods, walkdown of existing facilities, probabilistic seismic response analysis of structures, and analysis of damage indicating ground motion parameters. Most recently he served as reviewer for the Watts Bar SPRA peer review and he has defended the Indian Point SPRA peer review. He is a member of the ASME/ANS JCNRM Working Group maintaining Part 5 of the ASME/ANS PRA Standard. Mr. Ratnagaran has 5 years of experience and supported the Vogtle and Hatch Seismic PRA. He has defended the Vogtle Units 1 & 2 and 3 & 4 as well as Hatch SPRA peer reviews.

Mr. Edmond Wiegert was the lead reviewer for the SPR technical element. Mr. Wiegert has 25 years of experience in the nuclear industry and 17 years' experience in the areas of probabilistic risk assessment (PRA), now at Duke Energy as lead engineer in the PSA applications and models group. Mr. Wiegert has been supporting the SPRA modeling task for the Duke plants and supported numerous peer reviews. He was assisted in the SPR technical element review by Dr. Jonathan Lucero. Dr. Lucero has eight years of PRA and nuclear power experience in various aspects of PRA such as model maintenance, online and shutdown risk assessment, and regulatory oversight process. Dr. Lucero is the lead for the SPRA at Palo Verde and has supported numerous peer reviews for fire PRA and external hazards PRAs. Dr. Lucero was also the lead reviewer for the PRA configuration control element of the review.

Three working observers (Robert Keiser, Rusty Childs and Winston Stewart from Duke) supported the review of the SFR technical element, while David Gerlits, from Westinghouse supported as working observer the review of the SPR technical element. Any observations and findings these working observers generated were given to the peer review team for their review and "ownership." As such, Mr. Keiser, Mr. Childs, Mr. Stewart and Mr. Gerlits assisted with the review but were not formal members of the peer review team.

Finally, Mr. Gerald Dowdy (AEP) supported the review as a process observer. In this role Mr. Dowdy was not a formal reviewer.

A.4. Summary of the Peer Review Conclusions

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

Seismic Hazard (SHA)

The Standard requires the seismic hazard input to the SPRA be determined on the basis of a site-specific probabilistic seismic hazard analysis (PSHA). The PSHA performed for the North Anna site is a site-specific analysis that was performed by:

1. Using existing regional seismic source characterization (SSC) and ground motion characterization (GMC) models;
2. Assessing whether conditions local to the plant site and/or the availability of new data since the SSC and GMC models were developed require a revision of the regional-scale models to define a site-specific PSHA for the North Anna site;
3. Evaluating the effects of local site conditions on the ground motions; and
4. Considering potential ground failures caused by soil liquefaction, landslides, fault displacement, and other secondary hazards.

The regional-scale SSC model is the recently completed Central and Eastern U.S. (CEUS) seismic source model (NRC, EPRI, and DOE, 2012). The existing GMC model is based on the recent EPRI (2013) CEUS ground motion update project. Both models were the result of SSHAC Level 3 studies and, in the case of the GMC model, a SSHAC Level 2 update of a prior Level 3 study. The SSHAC process provides a structured approach to the use of experts and the evaluation and integration of available information, and provides minimum technical requirements to complete a PSHA. Using an appropriate

"SSHAC level" when conducting a seismic hazard study ensures that data, methods, and models supporting the PSHA are appropriately assessed and incorporated and that uncertainties are fully considered in the process at a sufficient depth and level of detail necessary to satisfy scientific and regulatory requirements. Although the Standard does not define a minimum required SSHAC level of analysis, the available Level 3 studies satisfy SHA High Level Requirement A (SHA-A).

As part of the SSC SSHAC study, comprehensive datasets were compiled to support the evaluations of the Technical Integration Teams, including regional geological, seismological, and geophysical data for the CEUS. As part of the CEUS SSC project, an earthquake catalog of relevant historical, instrumental, and paleoseismic information was gathered and processed. These aspects of the SSC study satisfy SHA-B.

The CEUS SSC model defines seismic sources for the entire central and eastern U.S. For purposes of the North Anna PSHA, background seismic sources in the CEUS SSC model within 320 km of the site were included in the PSHA. To expedite the calculations, cells in the gridded seismicity model that are more than 1,000 km from the North Anna site were not included in the calculation. This simplification is reasonable and does not impact the estimate of the seismic hazard at the site. In addition, repeated large-magnitude earthquake (RLME) seismic sources were included in the North Anna PSHA. The inclusion of RLME sources was based on the criterion that sources that contributed to 99 percent of the mean hazard for spectral acceleration of 1.0 Hz were included in the PSHA (sources contributing less than 1 percent were excluded). For the North Anna analysis, the Charleston, Reelfoot, New Madrid, and Wabash Valley RLME sources were included in the PSHA. Based on this selection process, "near-field" and "far-field" earthquake sources that are contributors to ground motions at North Anna were considered in the analysis. These aspects of the SSC study satisfy SHA-C.

As part of the GMC SSHAC studies in 2004 and 2013, available ground motion datasets and models were compiled and evaluated. The resulting ground motion prediction equations (GMPEs) account for epistemic and aleatory uncertainties. Accordingly, SHA-D is satisfied.

The effects of local site conditions are included (SHA-E) via amplification factors derived from site response analyses, which incorporate site-specific information on site topography, surficial geologic deposits, and site geotechnical properties. Epistemic uncertainty and aleatory variability in shear wave velocity, layer thickness, and nonlinear properties are considered in the site response analyses. However, inadequate justification is provided for the assumption that epistemic uncertainty is negligible (and thus not considered) compared to the aleatory variability.

Requirement SHA-F addresses the quantification of the seismic hazard; propagation of uncertainties and the results that are generated. Both the aleatory and epistemic uncertainties were addressed in the SSC and GMC parts of the analysis and were propagated through the hazard quantification. The PSHA results that were generated include fractile and mean hazard curves, uniform hazard response spectra, and magnitude-distance deaggregation plots. Sensitivity analysis results are presented in the report that document the contribution of seismic sources and alternative GMPEs to the site hazard. However, no sensitivity analyses were included to evaluate uncertainties in site response parameters.

The spectral shape used in the seismic PRA is based on the results of the site-specific PSHA (SHA-G). A GMRS was generated from the site-specific 1E-4 and 1E-5 UHRS and the associated design factors from Regulatory Guide 1.208. The UHRS developed in the PSHA was extended to a spectral frequency of 0.1 Hz. The process used in the extrapolation was based on the EPRI (2013) GMPEs for large-magnitude earthquakes (M 7 to 7.5) which predict constant spectral velocity in the 0.5 and 0.2 Hz and transition to constant spectral displacement at lower frequencies.

Vertical response spectra were developed for input to the seismic response analysis. The vertical spectra were derived from the horizontal response spectra using vertical-to-horizontal (V/H) spectral ratios. The McGuire et al. (2001) hard rock V/H ratios were used to derive V/H spectral ratios appropriate for CEUS soil sites.

The CEUS SSC and the GMC models are existing, regional-scale models, and in principle are not site-specific. The requirements of SHA-H state that if an existing PSHA is used, 'it shall be confirmed that the basic data and interpretations are still valid in light of current information. In the context of an existing, regional-scale study, SHA-H requires that steps be taken to develop a North Anna site-specific PSHA model. Somewhat unique to the North Anna site was the occurrence of the 2011 Mineral, VA earthquake near the plant. To satisfy SHA-H, the PSHA analysts conducted a systematic data collection and evaluation of geological, seismological, and geophysical data. This included a SSHAC Level 2 evaluation focused on the implications of the Mineral, VA earthquake, an update to the earthquake catalog that is the basis for the estimate of earthquake recurrence rates, and the evaluation of new information available in the literature to determine if there was a basis for making revisions to the SSC model or the addition of new, local seismic sources that would contribute to the ground motion hazard at the North Anna site. The evaluation of the Mineral, VA earthquake which included discussions/input from experts in the field, a literature review concluded there was no basis to revise or amend the SSC model for the North Anna PSHA.

In the case of the GMC model, a systematic assessment was not performed to assess whether an update was required. The reason for not conducting such an evaluation is the pending completion of the NGA East modeling effort that will develop new GMPEs for the CEUS.

The potential for induced earthquakes associated with hydraulic fracturing or waste fluid injection was not evaluated as part of the PSHA.

The Standard requires that a screening analysis be performed to assess whether in addition to vibratory ground motion, other seismic hazards, such as fault displacement, landslide, soil liquefaction, or soil settlement, need to be included in the seismic PRA. As part of the PSHA a systematic evaluation should be carried out to identify if there are other seismic hazards that may impact the site. Analyses were performed that considered the (i) seismic stability of the dike for the service water reservoir, and (ii) potential for soil liquefaction. A screening assessment for other potential seismic hazards, such as fault displacement, ground settlement, seiche in the reservoir, flooding due to dike breach and uncontrolled release of the reservoir, etc. was not performed. Furthermore, the evaluation of the potential in the power block area and instability of the service water reservoir dike lacked rigor and therefore a basis to confidently screen them out from the seismic PRA.

SHA-J defines the requirements for documentation of the PSHA. The documentation of the North Anna PSHA is a collection of documents and analyses for Units1/2 and 3. SHA-J sets a high bar with regard to the documentation that should be prepared for the PSHA and the needs (applications) it must satisfy (e.g., PRA applications, peer review, future updates). For the North Anna PSHA, a PSHA Summary Report that fully describes the methodology that was implemented, the rock PSHA results, the site response analysis, sensitivity studies, the control point motions, etc. was not prepared. The lack of a PSHA Summary report is further complicated by the fact that part of the Unit 1/2 seismic hazard story is based on the analyses and results that were performed for the Unit 3 combined license (COL) which is documented in the Unit 3 Final Safety Analysis Report (FSAR) and various supporting calculations.

Seismic Fragility (SFR)

Seismic fragility analyses were performed for the North Anna Power Station Units 1 and 2 structures, systems, and components (SSCs) that were included in the seismic equipment list (SEL). The SEL was developed from the internal event PRA basic events with additions and subtractions resulting from recognition that the seismic hazard applies loading conditions to passive as well as active equipment and components and recognition that some systems, components, and equipment are robust when subjected to seismic loading conditions. The focus was on SEL items that were significant contributors to seismic core damage frequency (SCDF) and seismic large early release frequency (SLERF). This was possible through efficient computational tools that permitted risk quantification analyses to be performed quickly for sensitivity study purposes.

The seismic fragility analyses were significantly enhanced through the NAPS systematic approach to solve complex technical problems associated with the SPRA. Position Papers were developed on fourteen (14) topics that represented either complex technical issues or issues that required coordination between various disciplines to ensure that rigorous treatment and consistency were maintained. The Position Papers identified industry standards, methodologies, and best practices to be implemented. An external expert technical panel reviewed and commented on draft versions of the Position Papers; these comments were addressed and final versions of the Position Papers were issued. The Position Papers were consistently used as guidance for the work performed throughout the SPRA effort. The PWROG Peer Review considers this approach a "best practice."

The SSCs that are judged to be of high-seismic capacity were not screened out and are retained in the Seismic PRA model. Only the SSCs that are considered inherently rugged were screened out from the Seismic PRA model. The high-seismic capacity SSCs are assigned a screening level HCLPF of 1.0g. A sensitivity study was conducted to show that the risk contribution (SCDF) of SSC failure (modeled as direct core damage) with HCLPF capacity of 0.6g is very low.

Seismic input motions were based on the ground motion response spectra (GMRS) characterized by high frequency motion (greater than 10 Hz) and anchored to a horizontal peak ground acceleration (PGA) of 0.572g. Foundation input response spectra (FIRS) were developed from the probabilistic site response analyses for eight different locations in the site profile, each corresponding to structure foundation locations of one or more (grouped) structures.

Median-centered seismic response analyses were performed for all the structures that were included in the SPRA. Depending on the foundation (soil/rock properties) and embedment conditions either fixed-base, deterministic soil structure interaction (D-SSI), or probabilistic SSI (P-SSI) analyses were performed. New finite element models were developed for structures if their existing lumped mass stick models (LMSM) were judged to be inadequate for use in the seismic response analyses. The preferred method of generating seismic responses for fragility development is the P-SSI approach. The NAPS SPRA Team implemented P-SSI analyses for soil/rock supported structures based on evaluations of their importance to risk, their physical attributes (supporting media, foundations, structure configuration) and the Team's intent to provide the best estimate of structure specific seismic responses (median and variabilities) for fragility development. The peer review team considered the response analysis results to be reasonable, and upon close inspection of the calculations, found that the methods and approaches were generally technically rigorous. For these reasons, the PWROG Peer Review considers the response analysis to represent a "best practice."

The seismic fragility analyses followed industry guidelines as described in Position Papers 3, 4, and 6. Generally, plant-specific data was used, including seismic qualification data, 2011 Mineral earthquake performance data and post-earthquake evaluation results, IPEEE data and analyses, and other available information. Plant specific data was supplemented by seismic experience data, based on EPRI NP-6041, Rev. 1 and generic test data such as GERS and relay tests. Various levels of screening were progressively implemented based on robust behavior of SEL items as recognized by EPRI NP-6041, Rev. 1, and experience of the NAPS SPRA Team and consultants; bounding calculations of NAPS site specific hazard data and assumed fragility curve values to identify and confirm fragility functions (HCLPF and variability) that had minimal effects on risk metrics; and individual SEL items (or groups) that have minimal effects on the risk metrics as calculated by the sensitivity studies of risk quantification. Generally, plant-specific fragility functions for the remaining unscreened items were generated, including consideration of anchorage capacity and seismic systems interaction (II/I). Seismically induced fire and flood initiators were walked down and evaluated. These approaches of screening, calculation of preliminary fragilities (generally conservative), and performing sensitivity studies with risk quantification is acceptable. Some exceptions are noted below.

For future work (including resolution of findings of the PWROG Peer Review), the NAPS SPRA Team should focus on the following:

- Based on the ASME ANS "Addendum B," for Capability Categories II and III, realistic fragilities based on site/plant specific data are required. Accurate risk insights require realism to avoid misinterpretation of risk important SSCs and phenomena (SR SFR A2). Examples are:
    - Turbine Building (TB) and its contents were not credited in the quantification of the SCDF and SLERF. However, consequences of the assumed TB failure and its contents were not extensively evaluated. Assumed structure failure likely causes failure of its contents, which includes large diameter piping systems (Circulating Water System) (4-96" lines) that could be a flooding source to the adjacent Emergency Switchgear Rooms. Other consequences of TB failure, such as potential interaction or impact with, or load redistribution onto neighboring structures, should also be evaluated. The evaluation should be documented.
    - Structure fragilities based on EPRI NP-6041, Rev. 1 Table 2-3 provides HCLPF values for overall behavior when caveats are met. In addition, local sources of failure should be evaluated in a structure's focused walkdown, e.g., penetrations, relative displacement effects on systems running from structure-to-structure and supported therein. The review and evaluations of local sources of failure should be documented.
- Revisit all risk significant contributors to SCDF and SLERF, and verify their fragilities are site/plant specific and realistic, including MOVs, PORVs, reactor containment building, auxiliary feed water pump house, service water valve house, and emergency condensate storage tank.
- Fragilities calculated for other purposes, such as IPEEE, should be revisited to verify that all failure modes have been considered, e.g., the Emergency Condensate Storage Tank (ECST) fragility was previously based on tank failure – currently, and its fragility is based on EPRI NP-6041 Rev. 1 Table 2-3 failure of the

concrete missile shield (MS). This should be revisited, including penetrations in the MS.

- Facts and observations associated with the seismic walkdown including the following:
    - Revise and improve the summary walkdown report to facilitate peer review and future applications of the SPRA.
    - Review walkdown documentation for consistency between teams, equipment types, and locations. For example, focus on documentation of issues such as seismic systems interaction (II/I), systems extending from structure-to-structure and supporting structure-to-supporting structure.
    - Document the presence of all walkdown personnel and their credentials including technical support personnel.
- Respond to facts and observations associated with the structure response analysis, such as structure damping values, concrete compressive strength for stiffness calculations, SASSI Modified Subtraction Method (MSM) verification, etc.

Seismic Plant Response (SPR)

The NAPS SPRA was developed starting from the internal events PRA and captures seismically induced failures along with random failures, unavailabilities and operator errors. The SPRA was determined to adequately model seismically induced initiating events: the process was systematic to identify, screen, and model the events.

The review team identified some scenarios were not identified or not addressed. No seismic fire scenarios were modeled. This is possibly the result of an aggressive screening approach. It is noted that North Anna does not have a fire PRA to support fully defending the screening of all seismically induced fire scenarios. Furthermore, some of the scenarios evaluated in the IPEEE fire evaluation, with a realistic potential for a seismic-induced equivalent, were not included or addressed in the model. The rationale used for the screening of the seismically induced fire scenario relies upon the SPRA Implementation Guide (SPRAIG) [10]. The SPRAIG has been recently recognized to underestimate the possibility of seismically induced fire scenarios for example, by not addressing the possibility of seismically induced fires generated by high energy cabinets. Similarly, an important flood scenario discussed in the internal flooding PRA was not addressed completely for the seismic-related equivalent scenario.

The SPRA appropriately models the seismic failures in the system model. The effect of relay chatter is also addressed adequately. Operator actions are adequately addressed for the seismic related performance shaping factor. A progressive approach is used where important operator actions are addressed more in details. One exception was observed where a potentially significant operator action was not credited in the SPRA, possibly resulting in a slightly conservative estimation of LERF. As discussed above, the Flowserve RCP seal package is credited in the SPRA but was not peer reviewed as part of either this or previous peer reviews. The Flowserve RCP seal modeling in considered a model upgrade based on Reference 25, and as such will need to be peer reviewed.

The Seismic Equipment List appears to be comprehensively assembled, with the notable exception of the travelling screens that have been screened functionally before including them in the SEL for fragility considerations. This resulted in overlooking the structural failure mode of these components, which could prevent water flow. This appears to be an isolated issue in the process, that otherwise seems to be capturing all relevant components for fragility considerations.

The quantification of the SPRA was determined to be performed in accordance with standard practice. Meaningful insights can be retrieved from the quantification results and are effectively summarized and discussed. It was observed that no investigation was made on the sensitivity of the SPRA model to the number and size of the seismic acceleration bins used in the quantification of the SPRA. The eight seismic acceleration bins generated by default in FRANX have been retained. In other SPRAs across the industry, it was observed that some SPRA models are significantly sensitive to the number and especially size of the bins: SCDF can be overestimated up to 40% by a non-optimal binning size. It is recommended to perform a more extensive study of the stability of the model that could result in an appreciable modification of CDF and LERF.

It was also observed that the uncertainty assessment was limited to a few standard sensitivities. Most notably, no sensitivities were made to test the model sensitivity to grouping and correlation of fragilities. The grouping and correlation of fragilities is known to have the potential to mask or bias results and is recognized to have an appreciable degree of epistemic uncertainties. There is no evidence that alternative grouping of components in fragility groups have been considered or even discussed with the fragility team. Furthermore, there was no investigation of the epistemic/model uncertainties associated with modeling of FLEX equipment. Modeling FLEX equipment can be impacted due to the limited knowledge and experience associated with current human reliability analysis (HRA) methods and component reliability data. Because of the very high importance of FLEX equipment in the NAPS SPRA, some investigation of these aspects appears to be necessary. There are other assumptions documented in the development of the SPRA, for which a disposition of the potential associated epistemic uncertainties was not performed. These quantification and uncertainty issues are more important on LERF than on CDF, given the higher absolute LERF value.

PRA Configuration Control

The primary NAPS PRA Configuration Control procedure reviewed was NF-AA-PRA-410, Revision 8. This provided a good overview of the NAPS configuration control process. This document also listed specific guidance documents to assist the PRA engineer on how to evaluate and assess PRA model impacts. Overall the NAPS PRA Configuration

Control process meets the requirements of SMU. However, the following observations are made:

- Industry guidance documents are not listed.
- There's no specific step to evaluate a change impact as an "update" or "upgrade."
- There's no specific guidance on when to have a peer review performed.

It was also observed that all of the tools that are used to track design changes, errors or similar impacts to the internal events PRA model seem well organized and easy to use. However, the tools are not set up for managing model impacts specifically against the seismic PRA model per se. This has been flagged as a possible limitation of the program, especially given that the SPRA is the first additional hazard (i.e., non-internal events) that has been added to the NAPS PRA.

The review team concluded that the North Anna SPRA realistically reflects the seismic risk profile of the plant, with no evident bias or conservatism. As a result of the above the North Anna SPRA is judged by the review team to be technically adequate for supporting risk-informed applications and risk-informed decision making.

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 associated Finding F&Os and disposition for each. Table A-2 presents summary of the Finding F&Os and the disposition for each (included at the end of this Appendix due to size).

Table A-1: Summary of SRs Graded as Not Met or Capability Category I for Supporting Requirements Covered by the NAPS SPRA Peer Review

SR	Assessed Capability Category	Associated Finding F&Os	Disposition to Achieve Met or Capability Category II
SHA			
SHA-E2	CCI	20-3	Associated F&Os have been resolved. SR is judged to be Met for Capability Category II.
SHA-I2	Not Met	20-5, 20-8	Associated F&Os have been resolved. SR is judged to be Met.
SHA-J1	Not Met	20-1	Associated F&Os have been resolved. SR is judged to be Met.
SFR			
SFR-A2	CCI	23-8, 23-10, 23-11, 23-12, 24-2	Associated F&Os have been resolved. SR is judged to be Met for Capability Category II.
SFR-F2	Not Met	23-8, 24-2	Associated F&Os have been resolved. SR is judged to be Met.
SPR			
None	N/A	-	-
(S)MU			
(S)MU-B4	Not Met	25-4	Associated F&Os have been resolved. 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 NTTF Recommendation 2.1: Seismic of the 10 CFR 50.54(f) letter [1]. The conclusions of the peer review discussed above and summarized in this submittal demonstrates that the NAPS 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 CDF and LERF (Section 5)
- Summary of adaptations made in the internal events PRA model to produce the seismic PRA model and bases for the adaptations (Section 5)

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

The NAPS SPRA reflects the as-built and as-operated plant as of the cutoff date for the SPRA, January 2015. There are no permanent plant changes that have not been reflected in the SPRA model except for those discussed further in section A.9.

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 NAPS 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 2018. No changes were made in updating the model that would require a subsequent focused peer review except for the Flowserve RCP seal upgrade as discussed in F&O 25-9.

A.7. Summary of SPRA Capability Relative to SPID Tables 6-4 through 6-6

The PWR Owners Group performed a full scope peer review of the NAPS internal events PRA and internal flooding PRA that forms the basis for the SPRA to determine compliance with ASME PRA Standard, RA-S-2009 [23] and RG 1.200 Rev. 2 [11] in in November 2013. The ASME/ANS PRA standard contains a total of 316 numbered supporting requirements for internal events and internal flooding in nine technical elements and 10 configuration control supporting requirements. Eleven of the SRs were determined to be not applicable to the North Anna PRA. Of the 315 remaining SRs, 292 SRs, or 92%, were rated as SR Met, Capability Category I/II, or greater. Three SRs were rated as Category I and 20 SRs were Not Met. A total of 72 F&Os were issued by the peer review team with 35 being findings, 35 suggestions, and 2 were best practices. Since the peer review, the internal events PRA model has been revised to address 13 finding F&Os that were found to impact the PRA model logic and results. The remaining finding F&Os were considered to be documentation improvements and other changes that were not considered to impact the PRA model results. As part of the SPRA development, these F&Os were reviewed again and verified to not impact the SPRA model.

The PWR Owners Group performed a peer review of the NAPS SPRA in July, 2017 The results of this peer review are discussed above, including resolution of SRs assessed by the peer review as not meeting Capability Category II, and resolution of peer review findings pertinent to this submittal. The peer review team expressed the opinion that the NAPS 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 NAPS 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 (located at the end of this Appendix due to size) provides a summary of the disposition of the open SPRA peer review findings.
- Table A-3 provides an assessment of the expected impact on the results of the NAPS SPRA of the peer review Findings that have not been fully addressed.

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.

Table A-3 Summary of Impact of Not Met SRs and Unresolved Peer Review Findings

SR	Finding	Summary of Issue Not Fully Resolved	Impact on SPRA Results
SPR-B1	25-9	New Flowserve RCP seal model is considered an upgrade that has not been peer reviewed.	Flowserve seal model in NAPS SPRA is the same as the Flowserve seal model in the Surry PRA model, which has been peer reviewed. F&Os from the Surry Flowserve peer review were reviewed and verified not to impact the SPRA results.

As this list indicates, there is only one finding F&O that has not been resolved, and it is not expected to impact the SPRA results as noted in the table. All of the other finding F&Os have been resolved and therefore, the SPRA is considered to be technically adequate to provide risk insights for the NTTF 2.1 submittal.

The SPID [2] defines the principal parts of an SPRA, and the NAPS SPRA has been developed and documented in accordance with the SPID. The information in the tables identified above demonstrates that the NAPS 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, 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 NAPS 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 NAPS SPRA technical elements are noted in the SPRA documentation that was subject to peer review, and a summary of important modeling assumptions is included in Section 5.

- Completeness uncertainty addresses scope and level of detail. Uncertainties associated with scope and level of detail are documented in the PRA but are only considered for their impact on a specific application. No specific issues of PRA completeness were identified in the SPRA peer review.

A summary of potentially important sources of uncertainty in the NAPS 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 NAPS SPRA peer review team noted that both the aleatory and epistemic uncertainties have been addressed in characterizing the seismic sources. Uncertainties in each step of the hazard analysis were propagated and displayed in the final quantification of hazard estimates for the NAPS site. The peer review team noted that inadequate justification was provided in the site response analysis for the assumption that epistemic uncertainty is negligible (and thus not considered) compared to the aleatory variability.	The seismic hazard reasonably reflects sources of uncertainty. The conclusion in the site response analysis that epistemic uncertainty is negligible due to the extensive site investigations has been further justified in response to peer review team F&O 20-3 and is not considered a significant source of uncertainty.
Seismic Fragilities	The fragility of some SSCs were identified by the peer review team as being overly conservative resulting in low HCLPF capacities.	The fragilities of the SSCs noted by the peer review team were revised to more appropriately model their seismic capacity. Also, a sensitivity was performed to assess the impact of building capacity on the SCDF and SLERF, which showed no significant impact on the results that warrant changes to the fragilities.

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 PRA Model	Assumptions and sources of uncertainties in the SPR development were reviewed to identify sources that may have an important impact on the SPRA results. Sensitivities were performed as documented in Section 5.7 to verify the sources of uncertainty do not have a significant impact on the SPRA results. The peer review team assessed supporting requirement SPR-F3 as met for documenting sources of uncertainty. It did issue some F&Os recommending that additional sensitivities be performed. Additional sensitivities were performed to confirm the adequacy of the SPRA model.	Additional sensitivities were performed to address the peer review team F&Os. The results showed the SPR modeling as appropriate with no significant impact on the results.

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

The NAPS SPRA reflects the plant as of the cutoff date for the SPRA, which was January 2015. Table A-5 lists significant plant changes subsequent to this date and provides a qualitative assessment of the likely impact of those changes on the SPRA results and insights.

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

Description of Plant Change	Impact on SPRA Results
Motor-Control Center (MCC) Bucket Replacement – MCC breaker assemblies being replaced in various MCCs.	The new MCC buckets have contactors that have higher HCLPF capacity for relay chatter of the Motor-operated valves (MOVs) they power. The bucket replacement is an ongoing effort that will continue for the next couple of years. The SPRA reflects the latest configuration as of January 2018. As buckets are replaced, the HCLPF capacity of the corresponding MOVs will be improved and therefore the SCDF and SLERF will be reduced. The reduction in SCDF and SLERF is not expected to be significant given the relatively low FV importance of the fragility groups that model these MOV chatter failures.

Table A-5 Summary of Significant Plant Changes Since SPRA Cutoff Date	
Description of Plant Change	**Impact on SPRA Results**
EDG Room Heaters	During the walkdowns of the Emergency Diesel Generators (EDGs), the supports for the unit heaters in the EDG rooms were identified as potential seismic spatial interaction concerns. The heaters are located such that if the supports failed and the heater unit displaced significantly, the attached steam supply and drain piping could breach and leak into the room, or the heater unit could fall and impact sensitive EDG equipment. Modifications have been developed to upgrade the heater supports to resolve the seismic spatial interaction concern. The SPRA model assumes these modifications have been completed for all of the heaters. The modification for all but two of the heaters have been completed. The modifications are scheduled to be completed for the final two heaters during the Spring 2018 unit 1 refueling outage.
Flowserve Seals	North Anna has replaced the RCP seals with Flowserve low leakage seals for all RCPs expect for the Unit 1 'C' RCP, which still has the Westinghouse seal. This seal will be replaced during the Spring 2018 unit 1 refueling outage. The SPRA assumes all RCP seals have been replaced with the Flowserve seals.

Table A-5 Summary of Significant Plant Changes Since SPRA Cutoff Date	
Description of Plant Change	**Impact on SPRA Results**
Fire Extinguishers	During the seismic walkdowns, portable fire extinguishers were identified that were stored on a bracket configuration that could allow the fire extinguisher to fall during a severe seismic event. For the fire extinguishers located in areas where there are mitigating SSCs, the fire extinguisher could potentially impact sensitive equipment upon falling. Additionally, mobile CO2 firefighting carts were identified that could displace in the event of a severe seismic event and potentially impact sensitive equipment cabinets containing mitigating instruments. The SPRA does not include these damage scenarios because of the uncertainty in whether the fire extinguishers could fall from the support bracket or the firefighting carts could displace sufficiently to impact sensitive cabinets. The potential for seismic spatial interactions from this firefighting equipment is being addressed through engineering review to resolve the concern.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
SHA-J1, J3	20-1	The documentation of the North Anna PSHA is a collection of documents and analyses for Units 1/2 and 3. Documentation should be prepared for the PSHA that meets the needs of PRA applications, peer review, and future updates. For the North Anna PSHA a single volume, a PSHA Summary Report, that fully describes the methodology that was implemented, the rock PSHA results, the site response analysis, sensitivity studies, etc. was not prepared. The lack of a PSHA Summary report makes peer review difficult and could compromise future	As part of the PSHA documentation, a summary report of the seismic hazard methodology and results was not prepared. The lack of a summary report makes peer review difficult, fails to document certain basic PSHA results, and lacks reference (i.e., a roadmap) to supporting documents for more detailed explanation of each element of the analysis. While some elements of the overall PSHA process are extremely well-documented (e.g., CEUS SSC model), there is no summary document that: 1. Summarizes the	Prepare a Summary PSHA Report that provides a complete description of the PSHA, an overview and summary of the overall PSHA process, model uncertainties and assumptions, and fully documents intermediate and final seismic hazard products supporting the North Anna SPRA.	Documentation of the North Anna PSHA has been enhanced to provide a complete description of the PSHA, including an overview and summary of the overall PSHA process, model uncertainties and assumptions, and reference to intermediate and final seismic hazard documents supporting the North Anna SPRA. This finding is considered resolved and there is no affect on the SPRA results or conclusions.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
		efforts to understand what was done and implement future model updates. (This F&O originated from SR SHA-J1)	methodology that was used in the PSHA, including the implementation of the PSHA methodology, the estimate of GMRS and FIRS, the propagation of uncertainties in the analysis, etc. 2. Reports PSHA results for reference rock site conditions and control point motions, 3. Examines the potential for seismic hazards other than earthquake ground motion, and 4. Reports sensitivity analyses. A summary document, as typically prepared, describes the methodology that is		

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			implemented in the PSHA, provides an overview of the elements of the PSHA/GMRS/FIRS process, explains how the elements of the analysis fit together, provides a summary of work performed for each element, provides reference to supporting documents for more detailed explanation, describes the outputs from each element (and, if appropriate, where more complete outputs can be found), and describes, as appropriate, any independent external peer review to which each element was subjected.		

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
SHA-E2, F3	20-3	Epistemic uncertainty has not been included in the site response analysis. (This F&O originated from SR SHA-E2)	The analysis of epistemic uncertainties in the site response analysis involved a limited assessment that concluded these uncertainties were negligible. There are two issues that contribute to this requirement not being met: 1. Information on the site shear-wave velocity profile does not necessarily support the conclusion there is negligible epistemic uncertainty in the site response analysis. 2. Available documentation does not present an analysis of possible epistemic uncertainties. As a consequence there is no evidence to support the conclusion that these uncertainties are neglible.	The site response analysis should include epistemic uncertainty in shear wave velocity that reflects the potential for different interpretations of the available data. Alternatively, sensitivity analyses may be performed that demonstrate the insensitivity of the calculated surface hazard curves to the assumed combination of epistemic and aleatory uncertainties.	The geologic conditions at the North Anna site consist of saprolitic soils near the surface at the site that transition into more intact rock at depth. The weathering across the site is uneven, and the thicknesses of the various material layers within the subsurface soil profile vary widely and randomly throughout the site. The rock layers (zones) are defined by both rock quality designation (RQD) and shear wave velocity. Considering the original site investigations at Units 1 and 2 site and the more recent site investigations at the proposed site for Unit 3 (which shares the same geologic characteristics), the North Anna site is well characterized and extensively investigated with abundant high-quality data (>200 borings, including five deep borings with P-S suspension logging velocity measurements),

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
			In addition to the foregoing, supporting requirement SHA-E2 requires that aleatory and epistemic uncertainties be included in the site response analysis to satisfy Capability Category II/III. Evaluation of Shear-Wave Velocity Profiles - The PSHA analysts reviewed the measured shear wave velocity profiles and concluded that: "two shear wave velocity profiles are defined in the power block area using B-901, B907 and B-909. One is for mostly unfractured rock throughout the profile, and the other is for partially fractured rock down to around El. 184 ft, underlain by the same mostly		which reduces epistemic uncertainty in the site properties. These data also provide information to characterize the aleatory variation in layer thickness and shear wave velocity across the site. These variations were included in considerations of aleatory uncertainties for the base-case profile. No alternate profiles were considered because of the significant amount of recent site specific data and the relative insignificance of epistemic uncertainty with respect to the aleatory variability for this site. In addition, the seismic hazard results from the North Anna Units 1 and 2 PSHA are consistent with (1) the results of the PSHA independently performed for North Anna Unit 3 [24] and (2) the reuslts of the NRC confirmatory analysis PSHA [16] performed for the review of

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			unfractured profile" (Calculation 25161-G-011). Based on the nature of the surficial geology at the site, the PSHA analysts subsequently interpreted these two profiles (Profiles 1 and 2) to represent the lower and upper bounds of the aleatory variability in shear wave velocity and calculated the corresponding depth-dependent mean and standard deviation for a single base-case shear wave velocity profile. In arriving at this interpretation, the PSHA analysts did not include epistemic uncertainty in the shear wave velocity profile "because of the significant amount of recent site specific data and the relative insignificance of		the NAPS SHSR [3]. This finding is considered resolved and there is no effect on the SPRA results or conclusions.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			epistemic uncertainty with respect to the aleatory variability for this site." (25784-000-K0C-0000-00006). The review team believes that it is possible, that other experts may have interpreted the available data differently and arrived at the conclusion that Profiles 1 and 2 represent the epistemic uncertainty in shear wave velocity. Documentation of the Analysis of Epistemic Uncertainties - The current documentation of the site response analysis does not present an analysis and quantitative estimate of potential epistemic uncertainties that supports a determination and conclusion that they are negligible (e.g., effectively		

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			zero, or small enough to be of no engineering significance).		

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
SHA-F2	20-4	An important part of the North Anna PSHA is the analysis of site response and its impact on the ground motion hazard. Sensitivity analyses have not been performed that illustrate influence of site response on site motions, including variations in site velocity profiles, interpretations of site velocity data, etc. (This F&O originated from SR SHA-F2)	Per this requirement, analyses should be carried out to show the influence of factors that are important to the site hazard. With respect to the site response analysis, sensitivities have not been performed.	Sensitivity calculations should be performed to illustrate the effect of the site response on the plant ground motions, effect of alternative site Vs profiles, etc. on the site response and the ground motion hazard.	Sensitivity calculations have been performed for the North Anna Unit 3 (NA3) PSHA and site response analysis as documented in the NA3 FSAR [24], Section 2.5.2. A study was performed that demonstrated applicability of the North Anna Unit 3 PSHA to the North Anna Units 1 and 2 PSHA based on similarity of subsurface conditions / soil profiles, common site location of the units, and similarity of the hard rock seismic hazard. As a result, sensitivity calculations performed for NA3 are considered applicable for the North Anna Units 1 and 2 site response and ground motion hazard. This finding is considered resolved and there is no effect on the SPRA results or conclusions.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
SHA-I1	20-5	As part of the seismic hazard analysis, an evaluation must be performed to assess whether the other hazards that may be initiated by a seismic event can be screened out from the seismic PRA, not inlcuded in the assessment of seismic risk, or whether they should be quantitatively evaluated and included in the analysis.There are a number of 'other' seismic hazards that should be considered in the screening analysis. (This F&O originated from SR SHA-I1)	Analyses have been performed that considers the 1) seismic stability of the dike for the service water reservoir and 2) potential for soil liquefaction. A screening assessment for other potential seismic hazards, such as fault displacement, ground settlement, seiche in the reservoir, flooding due to dike breach and uncontrolled release of the reservoir, etc. have not been performed.	A systematic evaluation that identifies potential other seismic hazards, and performs a screening evaluation for each.	A review for other potential seismic hazards has been performed. Potential seismic hazards of seiche and fault displacement are evaluated in the UFSAR and determined not to be credible based on geographical parameters, which are not changed by consideration of the GMRS. Ground settlement near important SSCs with respect to the GMRS seismic hazard has been evaluated and determined to be insignificant. Therefore, these other seismic hazards were screened out. An analysis of the slope stability of the Service Water Reservoir dike was performed (as indicated in the finding basis) and the results are included in the SPRA model. Therefore, Service Water Reservoir dike breach as a result of seismic activity has been evaluated.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
					This finding is considered resolved and there is no effect on the SPRA results or conclusions.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
SHA-I1	20-8	The analysis used to screen out liquefaction triggering for structures within the power block and for buried piping lacks rigor. (This F&O originated from SR SHA-I1)	Regarding the potential for liquefaction triggering based on SPT data for soils within the power block and for buried piping, Calculation 25784-000-KOC-0000-00017 indicates that an SPT-based approach gives minimum FS values less than 1.0 and that a (lower-bound) Vs-based approach yields minimum FS values below 1.1. An ad-hoc argument is constructed that these values are likely not accurate because of the nature of the soils at the site. This approach to does not result in a compelling argument that liquefaction may be screened out.	As an alternative to an ad-hoc approach, one possible approach to address this issue is to begin with a review of liquefaction phenomena in residual soils (or lack thereof) and attempt to build a compelling argument that the nature of residual soil deposits (strong fabric, high variability, etc.) effectively precludes liquefaction in these soils.	The majority of Category I structures within the power block area are rock-founded for which liquefaction is not a consideration. Three Category I structures in the power block area and buried piping are completely or partially founded on soil – either residual soils or compacted engineered backfill underlain by residual soils. The soil material supporting these strucutures is saprolitic material. The engineered backfill consists of excavated saprolitic soil. The saprolite is classified into two zones based on the extensive site investigations carried out for North Anna Units 1 and 2 construction and for North Anna Unit 3 licensing. The material in these zones are described as:

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					Zone IIA: Saprolite – medium dense silty sand, with some fine-grained layers Zone IIB: Saprolite – very dense silty sand All Zone 1 material (residual clays and clayey silts) was removed during construction. Material underlying the Zones IIA and IIB material consists of zones of weathered to moderately weathered to fresh rock. As would be expected with these residual Zone IIA and IIB soils, the fabric is that of the parent rock, mainly a biotitic quartz gneiss. There is strong foliation in the saprolite, dipping at angles of about 50 degrees to the horizontal. The fabric is strongly anisotropic. The texture shows angular geometrically interlocking grains with a lack of void network. The mineralogy

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					also reflects the parent rock, with 30-40 percent quartz, 20 to 30 percent microline, 25 to 40 percent clay minerals, and 5 to 20 percent biotite (mica). The major clay mineral is halloysite (a hydrated form of kaolinite) with lesser amounts of illite and montmorillonite Much of the halloysite is in the form of aggregates that are larger than 2 micrometers and, therefore, would be classified as silt, allowing the sand to be classified as non-plastic. The fabric of the saprolite contrasts strongly with that of an alluvial or marine deposited sand. Such sand shows no foliation and no interlocking of grains, even though the grains can be quite angular. The fabric of saprolite is, therefore, not one of a transported soil but one of the parent rock material. Its age, fabric and interlocking angular grain structure, along with the

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					significant portion of low plasticity clay minerals present in the material, have been demonstrated to give the grain structure a low susceptibility to pore pressure build-up or liquefaction. This material would not lose a significant proportion of its shear strength during shaking. Although much of the fabric of the saprolite is lost during excavation and subsequent backfilling, some of its interlocking grain structure will remain, providing a low susceptibility of liquefaction of the saprolite fill. On the basis of the types of soil materials supporting the soil-founded structures and buried piping, liquefaction can be screened out from further consideration. However, the liquefaction analysis further evaluated the

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					potential for liquefaction based on correlations using blowcounts and shear wave velocity. For these evaluations, several conservatisms were included in determining the application of various correction factors, age factor, water table location (at surface), and material properties. In addition, the correlations are intended for liquefiable soils and the benefits of fabric and texture of the Zones IIA and IIB soils are not reflected in the calculations. Consequently, the results (factors of safety against liquefaction [FS]) were considered conservatively low and in fact some FS were below the lower limit of 1.1. The documentation of the liquefaction evaluation provided a qualitative basis for the conclusion that liquefaction was screened out based on the conservatisms in the application of the correlations.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					Since the documentation provides a basis for screening liquefaction out from further consideration based on the susceptibility of the material alone, this finding is considered resolved. There is no effect on the results or conclusions of the SPRA.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
SHA-I2	20-9	The analysis of the potential liquefaction-induced damage to the SWR lacks rigor. (This F&O originated from SR SHA-I2)	The objective of Section 6 of Calculation 2S784-000-KOC-0000-00017 is to estimate an appropriate FS against liquefaction for subsequent use in calculating a HCLPF capacity for liquefaction-induced stability of the SWR dike. The analyses used to estimate the appropriate FS are based on qualitative arguments such as "Although the FS is low, it is very improbable that liquefaction of 10 ft of soil at 40 to 50 ft depth could cause significant settlement or collapse" and "It is possible that liquefaction of 15 ft of soil at 40 to 55 ft depth could cause significant settlement or collapse and resulting water loss if the lateral extent of the liquefiable zone is	It may be possible to address this issue via one or more of the following approaches: (i) construct a compelling argument that the nature of the soils at the site and/or the lack of continuity of potentially liquefiable zones precludes liquefaction-induced instability of the dike and/or (ii) if liquefaction is assumed to occur, perform a post-earthquake stability analysis using appropriately selected residual undrained shear strengths to demonstrate that the post-earthquake factor of safety is adequate.	As part of the design, construction, and the licensing process for NAPS Units 1 and 2, the Service Water Reservoir (SWR) and its component materials have been subjected to extensive subsurface exploration, laboratory testing, analyses, and instrumentation monitoring. Twenty two borings were performed with depths ranged from 27 to 105 ft, and averaged 70 ft. Borings used standard penetration test (SPT) sampling and thin-walled tube samplers. The borings encountered fill, residual soil, and saprolite grading to sound rock. The soils underlying the SWR area are primarily residual soils and saprolites similar to those encountered at the main plant site. As would be expected, these soils are erratic in terms of spatial and property distribution. Typically, silts of low-to-

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			sufficiently large." No basis is provided for these conclusions, and thus the selection of FS = 1.34 as the most appropriate value to calculate a HCLPF capacity is not compelling.		moderate plasticity (ML and MH) are found in the upper portions of the soil profile. These finer materials grade to coarser-grained saprolite soils (SP, SM, and SP-SM) which are encountered in the lower portions of the profile. Sound bedrock is found at depths of about 65 ft to 100 ft below original ground surface. The saprolite is classified into two zones: Zone IIA – Saprolite – medium dense silty sand, with some fine-grained layers Zone IIB – Saprolite – very dense silty sand All Zone 1 material (residual clays and clayey silts) was removed during construction. Material underlying the Zones IIA and IIB material consists of zones of weathered to moderately

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					weathered to fresh rock. Materials with N values over about 30 blows/ft will typically not liquefy. Since the N-value for Zone IIB material is generally more than 50 blows/ft, it is not expected to liquefy. The Zone IIA material was further evaluated. As would be expected with these residual Zone IIA and IIB soils, the fabric is that of the parent rock, mainly a biotitic quartz gneiss. There is strong foliation in the saprolite, dipping at angles of about 50 degrees to the horizontal. The fabric is strongly anisotropic. The texture shows angular geometrically interlocking grains with a lack of void network. The mineralogy also reflects the parent rock, with 30-40 percent quartz, 20 to 30 percent microline, 25 to 40 percent clay minerals, and 5 to 20 percent biotite (mica). The

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					major clay mineral is halloysite (a hydrated form of kaolinite) with lesser amounts of illite and montmorillonite Much of the halloysite is in the form of aggregates that are larger than 2 micrometers and, therefore, would be classified as silt, allowing the sand to be classified as non-plastic. The fabric of the saprolite contrasts strongly with that of an alluvial or marine deposited sand. Such sand shows no foliation and no interlocking of grains, even though the grains can be quite angular. The fabric of saprolite is, therefore, not one of a transported soil but one of the parent rock material. Its age, fabric and interlocking angular grain structure, along with the significant portion of low plasticity clay minerals present in the material, have been demonstrated to give the grain structure a low susceptibility to

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					pore pressure build-up or liquefaction. This material would not lose a significant proportion of its shear strength during shaking. Although much of the fabric of the saprolite is lost during excavation and subsequent backfilling, some of its interlocking grain structure will remain, providing a low susceptibility of liquefaction of the saprolite fill. On the basis of the types of soil materials supporting the SWR and comprising the construction of the dike, liquefaction can be screened out from further consideration. However, the liquefaction analysis further evaluated the potential for liquefaction based on correlations using blowcounts and shear wave velocity. For these evaluations, several conservatisms were included in

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
					determining the application of various correction factors, age factor, water table location (assumed at surface), and material properties. In addition, the correlations are intended for evaluation of liquefiable soils and the benefits of fabric and texture of the Zones IIA and IIB soils are not reflected in the calculations. Consequently, the results (factors of safety against liquefaction [FS]) were considered conservatively low and in fact some FS were below the lower limit of 1.1. The documentation of the liquefaction evaluation provided a qualitative basis for the conclusion that liquefaction potential remained limited based on the conservatisms in the application of the correlations. Since the documentation provides a basis for screening liquefaction out from further consideration based on the

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					susceptibility of the material alone, the SPRA has been updated to screen out liquefaction from consideration for the SWR. This finding is considered resolved. There is no effect on the results or conclusions of the SPRA.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
SHA-I2	20-10	Epistemic uncertainty is not adequately reflected in the mean fragility curves for slope failure of the SWR dike and liquefaction triggering for foundation soils in the SWR area. (This F&O originated from SR SHA-I2)	The analyses performed to estimate mean fragility curves for slope failure of the SWR dike and liquefaction triggering for foundation soils in the SWR area are based on values of composite beta that do not include consideration of epistemic uncertainty. While appropriate for calculating the HCLPF capacity, the resulting mean fragility curves are not appropriate for use in risk quantification.	Ensure that epistemic uncertainty is included in estimates of the composite beta for the mean fragility curves for slope failure of the SWR dike and liquefaction triggering for foundation soils in the SWR area that are used for risk quantification.	The SWR slope failure analysis has been updated to include consideration of epistemic uncertainty and provide appropriate input to the risk quantification. This finding is considered resolved and there is no effect on the SPRA results or conclusions.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
SHA-J2	20-11	The documentation of the methods and processes should be improved in a number of areas. (This F&O originated from SR SHA-J2)	The documentation of the PSHA was should be reviewed and improved at least in the following areas: 1. The documentation in several calculation packages (25784-000-K0C-0000-00006, 25784-000-K0C-0000-00013, and 25784-000-K0C-0000-00019) should be revised to make it clear that site-specific modulus reduction and damping curves were used for Zone II and III materials. 2. The assumed variation of shear modulus with shear strain for Zones III-IV and IV should be more clearly stated. 3. In several calculation packages (25784-000-K0C-0000-	Review the PSHA to ensure that each of the elements of the analysis is fully described, including the overall implementation.	These documentation clarification items will be considered for inclusion in any required future revisions of the applicable documents. The disposition of this documentation-related finding has no significant impact to the SPRA results or conclusions.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			00006, 25784-000-K0C-0000-00013, and 25784-000-K0C-0000-00019), plots are presented to verify that the median of the 60 simulated shear wave velocity profiles approximately matches the best-estimate profile. Similar plots should be developed to demonstrate that the random variability in the simulated profiles is a reasonable approximation to the (assumed) randomness observed in measured shear wave velocity profiles. 4. In Calculation Package 25784-000-K0C-0000-00006, Table 1 provides the mean thickness and standard deviation of thickness for each stratum. Later (p. 19 of 54) it is stated that the		

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			variability in stratum thickness is +/- 20%, which is inconsistent with the values provided in Table 1. 5. Figure 11 from Calculation 25161-G-017, Rev. 006, North Anna COL Unit 3 should be included in the documentation of the site response analysis to more completely illustrate the interpretation of available shear wave velocity measurements to derive the best-estimate profile and associated variability. 6. In Calculation Package 25784-000-KOC-0000-00016, the choice of a minimum acceptable factor of safety for pseudo-static slope stability analysis implies some tolerable displacement as indicated		

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			in Table 10.1 from Reference 16 (provided in Attachment 3). This relationship between factor of safety and tolerable displacement should be acknowledged and discussed in the calculation package. 7. In Calculation Package 25784-000-K0C-0000-00059, it would be helpful to include plots of the critical failure surfaces associated with each case included in Table 11. 8. Given the nature of the soil profile consisting of weathered material of varying thickness, the assumption that a one-dimensional site response is appropriate should be discussed and justified. 9. The implementation		

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			of Approach 3 to combine the hard-rock seismic hazard curves with the site amplification functions should be documented in greater detail, particularly with respect to the use of fractile hard-rock hazard curves rather than the suite of individual hard-rock hazard curves.		

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
SHA-J3	20-12	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. These methods look at the epistemic uncertainties associated with data, models and methods that could contribute to the uncertainty in elements of the PSHA. This supporting requirement states sources of model uncertainty and assumptions must be documented. What it does not say, is these topics must be documented in a manner that supports the	Within in the scope of the PSHA there are 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 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 fundamental understanding of tectonic processes (strain accumulation). A second practice is the fact that earthquake aftershocks are	Conduct a systematic review and evaluation to identify (e.g., a tabular summary) and document (e.g., discussion) sources of model uncertainty and analysis assumptions in the PSHA. The documentation of uncertainties and assumptions should provide the SPRA analysts with insight and guidance as to elements of the SHA.	Documentation of the North Anna PSHA has been enhanced to provide a complete description of the PSHA, including an overview and summary of the overall PSHA process, model uncertainties and assumptions, and reference to intermediate and final seismic hazard documents supporting the North Anna SPRA. Site conditions for North Anna Units 1 and 2 are consistent with the use of standard practice in modeling and analysis for the PSHA. In addition, the seismic hazard results from the North Anna Units 1 and 2 PSHA are consistent with (1) the results of the PSHA independently performed for North Anna Unit 3 [24] and (2) the reuslts of the NRC confirmatory analysis PSHA [16] performed for the review of the NAPS SHSR [3].

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
		SPRA analysts ability to assess whether identified sources of uncertainty or assumptions may have important implications to estimates of plant risk. (This F&O originated from SR SHA-J3)	not modeled in the PSHA, even though they may be significant events (depending on the size of the main event). In the spirit of this requirement 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. The model uncertainties and assumptions in a PSHA fall into the following categories: 1. Uncertainties that		The uncertainties in the PSHA are ultimately captured in the hazard curve distribution (mean, 16th, 50th, 84th) that is used in the parametric uncertainty analysis to estimate the distribution of the SPRA results due to variability in the SSC seismic failure probabilities and seismic hazard initiating event frequencies. The parametric uncertainty analysis uses the EPRI UNCERT code that employs the Monte Carlo technique to generate random samples for each probabilistically-varying event and to quantify the uncertainty distribution. The parametric uncertainty analysis would be expected to encompass the effects of model uncertainty and analysis assumptions that are not explicitly modeled since they are part of the standard-of-practice in the PSHA.

Table A-2: Summary of Finding F&Os and Disposition Status

SR	F&O	Description	Basis	Suggested Resolution	Disposition
			are explicitly identified and modeled in the PSHA logic trees 2. Methods, sources of uncertainty or modeling assumptions that are not explicitly modeled since they are part of the standard-of-practice in PSHA (i.e., earthquake occurrence modeling), site response analysis, etc. 3. Detailed modeling assumptions that are made as part of specific calculations (e.g., liquefaction assessment, slope stability failure criterion). The PSHA documentation addresses, at least in part,		This finding is considered resolved and there is no effect on the SPRA results or conclusions.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			Items 1 and 3. What the available documentation does not do is provide a comprehensive summary of the model uncertainties and assumptions in the PSHA and insight to their possible implication to estimates of plant risk.		

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
SFR-E5	23-4	During the peer review team walkdown performed on 7/18/17, several potential interaction sources were identified that were not in the SPRA walkdown documentation. However, other SEWS forms did identify potential interaction sources, often in detail. Therefore, it is clear that interaction was considered, but for a number SSCs, some potential interactions were not documented, and their disposition is likewise not documented. (This F&O originated from SR SFR-E5)	It is unclear from the available SPRA documentation whether the following potential interaction sources, which were identified during the PRT walkdown, are significant to the SPRA: a) Round duct in quench spray pump house that has fixed supports on either side of a building joint. Potential for seismic anchor motion across the building separation joint was not documented in the SEWS or evaluated subsequent to walkdown. b) Rack 1-802 supports some instruments that have attached lines anchored to the safeguards building wall, while the rack is	Review the walkdown documentation for consistency across various teams, equipment types, locations, etc., to assess whether the kinds of issues identified here are contained to a limited extent within the documentation. Supplement the walkdowns and documentation as necessary to provide confidence that the review for potential interactions was comprehensive.	With the exception of Item (b), each of the listed items identified during the PRT walkdown were dispositioned at the time of the peer review. When appropriate, document updates have been completed to address omissions and / or document dispositions provided. Item (b) identified one of a few SEL items that were not walked down prior to the peer review. These items were identified in the walkdown summary report. Since the peer review, walkdown inspections have been completed for those SEL items missing walkdown documentation including the transmitters on Rack 1-802. In addition, subsequent walkdowns were conducted in various plant areas containing SEL equipment (with a particular emphasis on the Emergency Switchgear / Instrument Rack and Relay Rooms) to confirm the adequacy

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			anchored to the floor of the quench spray pump house. The potential for seismic anchor motion was not identified on walkdown SEWS or evaluated subsequent to walkdown. c) The 1H 4kV switchgear is in close proximity to a neighboring computer rack, 1-EI-CB-301A. The proximity was not noted on the SEWS or evaluated subsequent to the walkdown. d) On both lineups of the 1H 4kV switchgear, there is a copper bus bar on each end of the lineup that is flexible and free to slap against the side of the cabinet during an earthquake. This potential interaction was not noted on the SEWS or evaluated		of walkdowns performed. There were no obvious deficiencies in terms of identifying seismic interactions. This finding is considered resolved. There is no effect on the results or conclusions of the SPRA.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			subsequent to the walkdown. e) Inverter 1-VB-INV-02: There is about a 3/8" to 1/2" gap between the inverter and a unistrut that is attached to a neighboring cabinet. The proximity issue was not noted on the inverter SEWS or evaluated subsequent to the walkdown. f) There is a mobile CO2 firefighting cart located near the 1 EP CB 28A relay cabinet. It appears if the cart overturns, it could hit the cabinet and potentially affect the function of the relays. The potential interaction was not noted in the walkdown SEWS provided to the peer review team. The SPRA team indicated during the peer		

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			review week that this issue was identified during the walkdown and subsequently evaluated and dispositioned but inadvertently omitted from the walkdown documentation. g) A clamping mechanism on top of Relay Cabinet 1-EP-CB-28Ais 1/16 in. away from the top of cabinet 1HC-H2A-101 at the end of the lineup. The potential interaction was not noted in the SEWS or evaluated subsequently.		

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
SFR-F3, G1	23-5	Some numerical errors were found in the relay computations in Appendix B of Position Paper 9 for relay fragility group SEIS-EE-BKR-HJ8-RLY. (This F&O originated from SR SFR-G1)	Page 3 of 9, Cabinet 1-EE-BKR-15H8, Relays 12HFA151A2F, 12IJCV51B23A and 12PJC11AV1A, the demand acceleration should be 1.8225g instead of 1.821g and the associated Beta U should be 0.2961 instead of 0.253. It appears that making this correction should not affect the SPRA results because these relays are non-governing. Therefore, this appears to be a documentation issue only.	Correct the numerical errors and verify the change does not affect the SPRA results.	The identified numerical error has been corrected and documented. Calculated relay fragilities were only minimally changed as a result. The identified error had no impact on SPRA results. An extent of condition assessment identified no other similar errors. This finding is considered resolved. There is no effect on the results or conclusions of the SPRA.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
SFR-C4	23-6	A slightly conservative estimate of median concrete compressive strength is used to calculate structure stiffnesses. According to EPRI TR-103959, this approach is expected to overestimate structure stiffnesses. (This F&O originated from SR SFR-C4)	EPRI TR-103959 states that minimum specified 28-day strength should be used to estimate structure stiffnesses for concrete shear wall structures. It states that stiffnesses can alternatively be estimated using 0.7 times the median concrete strength. If either of these accepted industry-standard approaches are used, it is expected that the structure stiffnesses would be significantly lower than those used in the NAPS SPRA. The stiffness change could affect structure forces and ISRS, both in amplitude and peak frequencies.	JUSTIFY how the conservative approach used by NAPS to evaluate the concrete compressive strength yield realistic structural loads and floor response spectra for use in the seismic PRA. An appropriate sensitivity could be performed to show that the overall impact of not using the mentioned industry accepted approach for stiffness calculation is not significantly impacting the final fragility calculation.	The median concrete elastic modulus is calculated using ACI formulation after aging considerations. The specified design compressive strength of concrete for the NAPS Units 1 and 2 structures is 3000 psi. Test results from 2032 cylinder test specimens taken across different structures at the site early in plant life showed that 67.5% of the specimens had 28-day strength of more than 4500 psi, and, thus, the median strength is higher than 4500 psi. Therefore, the use of 3000 psi compressive strength would lead to un-conservatively low estimate of uncracked structural stiffness for North Anna structures. On the other hand, recent research has shown that typical shear walls are flexible compared to the stiffness representations given in ASCE 43-05. Considering both of the above points, for the North

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					Anna SPRA effort and for use with the ASCE 43-05 formulation for elastic modulus, E, the median 28 day strength of concrete is judged to be 4500 psi and a 70% value of E and modulus of rigidity, G, need not be considered. Per recommendations of EPRI TR-103959, an aging factor of 1.2 is applied to obtain the median strength of concrete as f'c = 5400 psi. Thus, the concrete elastic modulus is calculated as $E_c = 57000\sqrt{f_c'}$ = 4189 ksi or 603,200 ksf. This finding is considered resolved. There is no effect on the results or conclusions of the SPRA.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
SFR-A2, F2	23-8	Fragilities for some significant SLERF contributors are not realistic. If the fragilities are refined, the SPRA results and insights could be affected substantially. (This F&O originated from SR SFR-F2)	The PRT reviewed a sample of the significant contributors to SCDF and SLERF. As defined in the PRA notebooks, 'significant' SSCs have Fussel-Vessely importance of 0.005 or greater. Some significant SLERF contributors are not realistic. For example, the following significant structure and tank fragilities are computed based on the screening level capacities in EPRI NP-6041 Tables 2.3: - Reactor Containment Building - Auxiliary Feed Water Pump House - Emergency Condensate Storage Tank	Realistic fragilities should be developed for the SSCs that are significant contributors to seismic risk.	Detailed fragility calculations for the Auxiliary Feedwater Pump House and the Emergency Condensate Storage Tank that provide more realistic inputs to the SPRA have been performed and the results have been incorporatd into the SPRA. Sensitivity studies have been performed for the Containment and Service Water Valve House fragility values have been performed to determine the effect on the SPRA results. Although higher fragility values provide some SPRA results improvements, the changes are not significant. Motor-operated valve (MOV) and the MS PORV fragility evaluations have been refined where possible and the more realistic results have been incorporated into the SPRA.In addition. This finding is considered

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
			- Service Water Valve House Additionally, all the MOVs that are credited in the SPRA (some of which are significant to SLERF) are assigned a HCLPF of 0.6g based on the limiting fragility value of all those MOVs. This is conservative for most MOVs. The PRT also reviewed the MS PORVs fragility as part of the sample review. The PORVs are the #5 top SLERF contributor according to Table 3-12 in Notebook SA.1. The HCLPF is 0.32g and appears to be based on a 1.8g generic spectral capacity. This is probably conservative for this valve, and if a component-specific		resolved. The results of improved fragility evaluations have been incorporated into the SPRA and minor improvements in SCDF and/or SLERF were realized. This finding is considered resolved.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			evaluation were performed based on plant-specific qualification levels or component-specific stress analysis, this fragility could likely be significantly refined.		

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
SFR-A2	23-10	When fragilities are developed based on capacities from EPRI NP-6041 Table 2-3 or 2-4, those capacities are interpreted as a geomean of two horizontal directions. Accordingly, demands are likewise characterized as geomean of two horizontal directions for comparison to these capacities. In these cases, horizontal direction peak response variability (HDPRV) should not be included in separation of variables (SOV) or CDFM HCLPF calculations. Including HDPRV in these cases conservatively overestimates aleatory variability in SOV calculations, and conservatively	The HDPRV factor in one direction is the inverse of the other direction such that the geomean of the factors in the two directions is always 1.0, and there is no variability on this geomean factor.	For SOV calculations where the geomean of two horizontal directions is used to characterize demands, the HDPRV should be omitted from the variability calculations. For CDFM calculations, the 84% NEP demands should be adjusted to remove HDPRV. Alternatively, assess whether these changes might significantly affect the SPRA, and adjust the fragilities only as necessary to ensure meaningful results and insights. As another alternative, compare the current fragilities to the alternate, more	When functional fragilities are derived using EPRI NP-6041 SL-R1, Table 2-3 or 2-4, the use of geometric mean of the two horizontal spectral peaks is judged to be reasonable and will be about the same as using the arithmetic average of the two horizontal spectral values which is recommended on p. 2-44 of EPRI NP-6041 SL. An exception is when one direction clearly governs; the spectral accelerations in that direction were used. The majority of fragility calculations for functional failure mode using NP-6041-SL, Tables 2-3 and 2-4 were performed with the CDFM approach. In these calculations, 84% ISRS from the response analyses were used and the variabilities were used from the SPID; thus HDPRV was not considered explicitly for any

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
		overestimates 84% NEP demands for CDFM calculations. (This F&O originated from SR SFR-A2)		conventional approach in which EPRI 6041 capacities are compared to the maximum direction response rather than the geomean. Assess whether the current fragilities as calculated can be justified based on this comparison.	direction. Therefore, for CDFM calculations, the 84% NEP demands need not be adjusted since explicitly calculated variabilities from structural response were not used. When an SOV analysis is performed, the use of geometric mean could slightly overestimate the aleatory variability since the variabilities due to structural response were explicitly calculated. However, since the composite variability remains the same, a small redistribution of the aleatory and epistemic variabilities is judged not to affect the fragility curve significantly. In addition, the top risk contributor SSCs, where the SOV approach was used, include several relays; however, in these analyses the governing horizontal direction was used rather than the geometric mean. Other risk-significant components using the

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					SOV approach are the vital inverters where geometric mean was used. A review of this calculation shows that the anchorage controls the fragility and not the function (which was based on NP-6041 Table 2-4) therefore, there is no impact. Other SSCs where SOV calculations were performed are not among the top contributors to risk. Therefore, if HDPRV was removed, the effect on the SPRA results and risk insights would be negligible. This finding is considered resolved. There is no effect on the results or conclusions of the SPRA.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
SFR-A2	23-11	The 0.6g HCLPF for the Emergency Condensate Storage Tank (ECST) is not adequately justified as representative of the realistic failure behavior. (This F&O originated from SR SFR-A2)	The ECST IPEEE HCLPF was relatively low and identified as important in that evaluation. Similarly, a fragility evaluation was performed for the SPRA, and it was likewise relatively low and important. The fragilities were based on failure of the steel tank, and did not address additional strength or dynamic influence of connection to concrete missile shield. The final fragility that is used to represent the ECST in the SPRA is based on failure of the concrete missile shield. The fragility is based on EPRI NP-6041 Table 2-3 for reinforced concrete shear wall structures. Table 2-3 indicates penetrations must	The fragility should be refined to realistically characterize the dynamic response of the tank and the progression of failure. Alternatively, a sensitivity could be performed to assess how sensitive the SPRA results are to assumptions regarding the progression of failure and the concrete's ability to retain the fluid.	A realistic fragility analysis has been performed for the Emergency Condensate Storage Tank (ECST) missile shield structure and the HCLPF and median fragility of the structure is no longer based on EPRI NP-6041 Table 2-3 for reinforced concrete shear wall structures. The HCLPF capacity for the structure is greater than 1g. The seismic fragility evaluation for the steel tank concluded that overturning and sliding were the governing failure modes. This fragility analysis is not a realistic representation of the failure of the function of the ECST since the steel tank is completely surrounded by a 2-foot thick reinforced concrete missile shield that would restrict sliding or overturning. Additionally, in the event of a breach of the pressure boundary of the tank within the

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			be evaluated, but there is no documented evaluation of the penetrations in the ECST fragility calculation. It is not clear from the available documentation whether the concrete shield wall is capable of retaining the ECST contents in case the steel tank fails. The penetrations, for example, are sealed with elastomeric sealant, and there is no evaluation whether this sealant would remain intact if the ECST were to fail. The lower fragilities representing failure of steel tank probably underestimate the actual fragility since they do not credit the support provided by the concrete shield wall. The fragility representing shield wall failure, however, may be unconservative		missile shield, significant inventory loss would not be expected since the reinforced concrete shield is essentially a monolithic structure and penetrations through the shield are sealed by caulking or grout. For this case, the missile shield would function as the tank pressure boundary and the limited displacement of the steel tank within the shield would not prevent the flow of tank contents through the connected piping. The mission time for the use of the tank contents is relatively short such that a small amount of leakage through the shield penetrations would not significantly affect available tank inventory or the function of the tank to provide an adequate water source to the Auxiliary Feedwater System pumps. Therefore, the seismic fragility of the reinforced concrete missile

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			because it does not adequately address the capability of the shield wall to retain the ECST contents in a useful state (e.g., no documentation of the capability of the penetrations to retain the fluid).		shield structure provides a realistic representation of the ECST seismic fragility and was used as the input to the SPRA model. This finding is considered resolved. The results of the SPRA are improved slightly by the refined fragility analysis of the ECST missile shield structure.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
SFR-A2, D1, F2	24-2	The Position Paper 12_R0 (Service Bldg) evaluates only one failure mode for the Turbine Building. (This F&O originated from SR SFR-D1)	Based on the review of plant design documents and observations made during the walkdown, there are additional failure modes for the Turbine Building which are not identified in the Seismic PRA model.	Estimate realistic failure modes for the Turbine Building (including potential seismic induced flood sources) and characterize its consequence.	Additional failure modes have been evaluated for the Turbine Building (TB) and the consequences have been characterized. The seismically-induced structural damage within the TB has been evaluated to determine the potential for significant flooding, fires, and toxic chemical releases that could adversely affect the function of core damage mitigating equipment or main control room (MCR) habitability. The TB has been modeled using the finite element method and a linear dynamic analysis of the TB response to seismic ground motions has been performed. The seismic ground motions were based on the re-evaluated seismic hazard, or Ground Motion Response Spectrum (GMRS), used for the seismic PRA.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					Results of the TB linear dynamic analysis were used, along with building walkdowns and design documentation reviews, to identify locations for first failure in the TB steel superstructure under GMRS-level loading conditions. Based on this information, a qualitative evaluation was made to determine bounding modes of failure for the TB. Two significant bounding modes of failure were evaluated for flooding, fire and impact on toxic chemical release; (1) complete collapse of theTB roof truss supporting structure and (2) derailment of the TB Unit 1 and 2 overhead bridge cranes resulting in crane free-fall to the TB operating deck. Flooding: These bounding structural modes of failure were evaluated for their potential to damage systems that would

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					constitute significant flood volume sources. (1) For the postulated failure of the TB roof truss supporting structure and its subsequent collapse onto the TB operating deck, it was concluded that the relatively lightweight roof truss members would not penetrate the concrete TB operating deck or cause collapse of the TB operating deck supporting structure. Since there are no significant flood sources on or above the operating deck, there were no flooding consequences identified from this bounding failure mode. (2) For the derailment of the overhead bridge cranes, each crane was assumed to free-fall to the operating deck as a result of the seismic motions. Significant flooding sources were identified in the Unit 1 TB, but are located in the basement and are west of the projected Unit 1 overhead

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
					crane fall path. No significant flooding sources are located near the fall path of the Unit 2 overhead crane. The bounding failure mode evaluation assumed that the Unit 1 overhead crane would derail and fall to the operating deck below with the north end of the crane passing through a large opening in the deck and coming to rest in the truck bay below. It was concluded that the TB operating deck would withstand the impact of the Unit 1 overhead crane with only local member damage and further progressive collapse of the TB operating deck would not occur. This conclusion was based on the substantial steel framing and thick reinforced concrete slab construction of the TB operating deck, which is designed to support heavy turbine dismantling / laydown equipment loads. Additionally,

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					Operating Experience from the ANO-1 stator drop (03/31/2013) was reviewed and provided support for the conclusion that operating deck damage would be limited to local member failure and not create a progressive collapse scenario. A review concluded that systems that constituted significant flood sources were located to the west of the fall zone of the Unit 1 overhead crane and that there would not be significant collateral damage to the TB operating deck from the postulated overhead crane drop that could adversely affect the water systems. Therefore, there were no flooding consequences identified from this bounding failure mode. Fire: There are systems in the TB that contain flammable materials, such as hydrogen for main generator cooling and

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					turbine lubricating oil. Structural damage in the TB could result in a breach of these systems and resulting fires. The evaluation concluded that firefighting would prevent the spread of fires to safety-related areas and that these areas are protected by fire-rated walls and doors. Therefore, there were no fire-related consequences from TB structural damage. Toxic Chemical Release / MCR Habitability: There are systems in the TB that contain toxic chemicals. Structural damage in the TB could result in a breach of these systems and result in a release of toxic chemicals to the environment, potentially affecting MCR habitability. The evaluation concluded that based on the limited amount of chemicals in the TB, and the manual initiation of MCR isolation by the operators in the

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					event that a toxic atmosphere is detected, MCR habitability would not be affected. Therefore, there are no consequences of toxic chemical release due to TB damage. Based on the evaluation of the effects of seismically-induced TB damage, this finding is considered resolved. There is no effect on the results or conclusions of the SPRA.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
SFR-C4	24-3	Median concrete damping of 5% is assigned for the seismic response analysis. (This F&O originated from SR SFR-C4)	The stress levels in most of the category I concrete structures are very low and the assigned damping is not consistent with the stress levels experienced by these structures.	Provide justification for the use of 5% median damping for the concrete structures which does not undergo cracking.	A median damping ratio of 5% was used for concrete materials, per Table 3-4 of EPRI TR 103959. This is based on demands at approximately ½ the yield strength for reinforced concrete with cracking. This value is also consistent with Table 4-1 of EPRI NP-6041-SL, Rev. 1, which recommends 5% damping for reinforced concrete with moderate cracking. If higher demands are observed based on a best estimate evaluation, a higher damping ratio of 10% for reinforced concrete could be justified along with the use of cracked properties for concrete. It is noted that more recent design codes such as ASCE 4-98 and ASCE 43-05 recommend the use of 4% damping ratio for uncracked concrete. While this value is appropriate for design, it is considered to be a

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					conservatively biased estimate of the median damping. Because the goal in the analysis supporting SPRA is to obtain an unbiased estimate of the median response, the use of a slightly higher damping ratio of 5% is considered appropriate. Furthermore, the shear demands in major concrete shear walls of Service Building and Auxiliary Building were evaluated and found to be generally between 1.5 to 3 square roots of f'c. These levels of stress in concrete shear walls are considered consistent with the adopted median damping ratio of 5%. It is also noted that some SPRA practitioners have used 7% or possibly higher concrete damping values in their structural dynamic analyses. For instance, a concrete damping of 7% was used for the

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					Watts Bar SPRA (Ref. NRC ADAMS Accession number ML17181A485) which has a lower GMRS than North Anna. Watts Bar assumed Damage Level 2 of ASCE/SEI Standard 43-05, 2005 and considered even the 7% damping somewhat conservatively biased relative to the likely damage state associated with the median seismic capacities of the SSCs. Thus the 5% structural damping used in dynamic analyses of structures is appropriate. This finding is considered resolved. There is no effect on the results or conclusions of the SPRA.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
SFR-C6	24-5	The SSI analysis of the embedded structures are performed using Modified Subtraction Method (MSM). (This F&O originated from SR SFR-C6)	The report does not provide justification on the adequacy of the MSM used for the embedded SSI analysis.	Perform a sensitivity study to validate the results from MSM by comparing with those of Direct method or Surrogate for the Direct method.	A sensitivity study has been performed to compare the soil structure interaction (SSI) analysis results using the modified subtraction method (MSM), also referred to as the extended subtraction method (ESM), and the direct method. The study results showed no significant differences for the two representative structures studied. This finding is considered resolved. There is no effect on the results or conclusions of the SPRA.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
SFR-G2	24-8	Additional documentation and corrections to the existing reports should be included.	The statement 'This calculation utilizes an unverified assumption about the accuracy of the provided in-structure response spectra, see Attachments 1 and 2.' on pages 4 and 41 of DMNNA023-CALC-006 Rev. 0 was inadvertently left in the calculation from an earlier draft and should have been removed. In Attachment 3 - SSC Fragility Summary Table of the calculation NAPS SA.5 R0, the revision/version numbers of the reference calculations should be updated for all the SSCs. Section 2.2.8 of the report (NAPS SA.4 R1) states that	Incorporate the changes to CCW report and the fragility summary table spreadsheet. Responses to the Block Walls screening approach, Collapse of fuel building and spent fuel pool, Incoherency Modes, Sliding and Overturning failure modes of the structures were provided during the on-site review and should be documented.	The administrative error in calculation DMNNA023-CALC-006 has been corrected. Attachment 3 - SSC Fragility Summary Table of NAPS SA.5 has been updated. The documentation associated with block wall evaluations has been updated to include a discussion of the approach for identifying block walls that could impact distributions systems (including their support if they are mounted on a block wall). The seismic capacity of the spent fuel pool has been evaluated and documented in accordance with the guidance in EPRI 3002009564 [15]. The SPRA documentation has been updated to reflect the

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
			'Of the 481 block walls in the plant, 34 walls have been identified for further evaluation if their failure could impact mitigating SSCs or operator pathways'. When pre-screening block walls, the documentation should also discuss the approach adopted for identifying block walls that could impact distributions systems (including their support if they are mounted on a block wall). The report (NAPS SA.4 R1) discusses the sloshing of the water in the spent fuel pool that could result in water "spilling" out of the pool and propagating into the Auxiliary building basement via the pipe tunnel between the two		conclusions of that evaluation. Ten (10) incoherency modes were used with SRSS combination for the computation of the SSI response due to incoherent input ground motion. While the number of incoherent modes selected was discussed during the in-process peer review, no sensitivity studies on the number of incoherent modes were suggested or performed. Based on the structural analyst's past experience with similar models and foundation dimensions, only the first few incoherency modes have significant contribution to the solution and the use of 10 incoherency modes is considered adequate. Because of the high frequency

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			buildings. However, it is unlikely there would be enough water from the SFP sloshing that would result in submergence damage. The report does not consider the failure of the spent fuel pool itself. The failure modes (collapse of the fuel building or the spent fuel pool) and its consequence should be evaluated and documented. For all the SSI analysis that included ground motion incoherency, the reports does not provide justification for the use of 10 incoherency modes. The sliding and overturning failure modes for the structures have not been		nature of the input motion at the site, and small building displacements calculated at the GMRS level of input, for the structures where specific fragility calculations were performed, the sliding and overturning modes for the structures were judged not to be governing. For structures where the fragility was determined from EPRI NP-6041 Table 2-3, the capacity is based on the information in the table and no specific failure modes were evaluated. This finding is considered resolved. There is no effect on the results or conclusions of the SPRA.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			evaluated. The basis for not including them as credible failure modes is not documented.		

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
SMU-B3	25-3	There's no explicit instruction to make model changes in accordance with RG 1.200, Rev. 2 and ASME/ANS RA-Sa-2009. (This F&O originated from SR SMU-B3)	Supporting guidance document NF-AA-PRA-4040 Revision 2 addresses the PRA Model of Record Revision process. This document primarily describes the work scope management process in making model changes. The PRA model elements of ASME/ANS RA-Sa-2009 are listed in definition item 5.3.10. However, there's no clearly stated direction for the PRA engineer to evaluate the model change in light of these PRA model elements and the governing document ASME/ANS RA-Sa-2009. The intent is clear but a set of instruction steps are missing.	Revise NF-AA-PRA-410 Revision 8, Step 3.6.1 to include ASME/ANS RA-Sa-2009, ASME/ANS RA-Sb-2013, and NRC Reg. Guide 1.200, Revision 2.	As a matter of practice, PRA model changes are developed and documented to meet the requirements of the ASME/ANS PRA Standard. But the PRA procedure, as noted by the peer review, lacked specific guidance for ensuring this. The PRA procedure was revised to include guidance for revising the PRA in accordance with the ASME/ANS PRA standard. This finding is considered resolved. There is no effect on the results or conclusions of the SPRA.
SMU-B4	25-4	There's no explicit instruction to review model changes to distinguish	There's a gap in the configuration control process where a peer	Revise NF-AA-PRA-410 Revision 8, to include steps to review the	As a matter of practice, PRA model changes are reviewed to identify changes that are

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
		between a PRA upgrade and an update. Furthermore, there's no guidance on when to require a peer review in accordance with this SR. (This F&O originated from SR SMU-B4)	review will be required. Furthermore, there's an increased emphasis on follow-on peer reviews in order to fully implement the guidance in NEI 05-04/07-12/12-06 Appendix X: Close Out of F&Os.	model impact classification and determine if it will result in an upgrade or an update. Also, include a step that stipulates a PRA model upgrade requires a follow-on peer review.	considered upgrades. A list of upgrades is maintained to track their status with respect to undergoing a peer review. But the PRA procedure, as noted by the peer review, lacked specific guidance for ensuring this. The PRA procedure has been revised to include guidance for reviewing model changes to distinguish between upgrades and updates. This finding is considered resolved. There is no effect on the results or conclusions of the SPRA.
SMU-E1	25-6	There is no software quality assurance (SQA) report that documents the impact assessment, classification, and verification/validation testing of FRANX 4.2 applied specifically for the seismic PRA quantification.	According to the subject matter expert PRA engineer, the version of FRANX used for the NAPS SPRA is 4.3. Testing was performed on FRANX 4.3 but an SQA report has not been issued for this version. Moreover, he stated that FRANX Version 4.4 is	Develop an SQA Software Code File in accordance with Administrative Procedure IT-AA-SQA-101 for the version of FRANX that quantifies the NAPS SPRA of record. In addition, ensure all relevant	PRA codes used in the development of PRA models at Dominion are maintained under the Software Quality Assurance (SQA) program. As noted by the peer review team, the SQA code file for the FRANX version used in the development of the SPRA was not up to date. Subsequent to the peer review, the SQA code

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
		(This F&O originated from SR SMU-E1)	expected to be released soon and they will be updating the FRANX SQA file for that version.	documents the state the version of FRANX in use.	file was updated to match the the version used in the SPRA development This finding is considered resolved. There is no effect on the results or conclusions of the SPRA.
SPR-E2, E6	25-8	Several sources of uncertainty are identified in the NAPS SPRA that are not addressed in the section, Sensitivity Studies Section 4.2 of NOTEBK-PRA-NAPS-SA.1 Revision 0. (This F&O originated from SR SPR-E2)	Many sources of modeling uncertainties are found unaddressed when comparing the sources of modeling uncertainties identified in the NAPS SPRA with those addressed in section 4.2 of NOTEBK-PRA-NAPS-SA.1 Revision 0. Specifically, the sources of uncertainty were gleaned from NOTEBK-PRA-NAPS-SA.3 Revision 0, Table 7-1 (Generic Sources of Uncertainty for Seismic PRA) and Table 7-2 (Plant-Specific Assumptions and Uncertainties). These	Review all the uncertainties listed in Tables 7-1 and 7-2 of SA.3. For the uncertainties that are not addressed with a sensitivity case in section 4.2 of SA.1, provide a sensitivity case or document the reason why a sensitivity case is not necessary to satisfy SR QU-E4.	The sources of uncertainties were updated and sensitivities added to the Seismic Quantification notebook as needed. This finding is considered resolved. There is no effect on the results or conclusions of the SPRA.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			uncertainties are compared to the Sensitivity Studies section (Section 4.2) of SA.1 Rev 0. Per SR QU-E4, the above sources of uncertainty need to be addressed and their potential impact assessed.		
SPR-B1	25-9	The SPRA model builds on an interim Internal Events PRA model that includes the modeling of low leakage RCP Flowserve seals. This is a PRA model upgrade that according to ASME/ANS RA-Sb-2013, Nonmandatory Appendix 1-A, requires a focused peer review prior to crediting.	NOTEBK-PRA-NAPS-SA.3 Revision 0, Section 3.1, Table 3-1, lists RCP low leakage Flowserve seal modeling as included in the SPRA in the Loss of RCP Seal Cooling Internal Events Event Tree. However, the Internal Events PRA notebook NOTEBK-PRA-NAPS-AS.1 Revision 5,	Perform a focused peer review and document in accordance with ASME/ANS RA-Sb-2013 1-6.2.4 and 1-6.6, respectively.	The Flowserve RCP seal model upgrade has not yet been peer reviewed. However, the Flowserve seal model in the North Anna PRA (and SPRA) is nearly identical to the Flowserve seal model in the Surry PRA, which had undergone a peer review in 2013. The F&Os from the Surry peer review of the seal model were reviewed for

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
		(This F&O originated from SR SPR-B1)	Accident Sequence Analysis, states that the logic for the Flowserve seals is disabled until the seals are replaced in all RCPs.		applicability to the North Anna PRA seal model. The conclusion is that the F&Os either are not applicable to the North Anna seal model or they have no impact on the results. This F&O will remain as unresolved until a peer review is performed. However, the SPRA results are not impacted.
SPR-E2, E6	25-11	The uncertainties of accelaration bin range and ACUBE parameters have not been identified and evaluated. (This F&O originated from SR SPR-E2)	It has been shown that accelaration bin range and ACUBE parameter selection can have significant impacts on CDF and LERF values. Per SR QU-E4, the above sources of uncertainty need to be addressed and their potential impact assessed.	Perform a sensitivity analysis on acceleration bin ranges and demonstrate CDF and LERF stability. Perform a sensitivity analysis on ACUBE parameters and demonstrate CDF and LERF stability.	The number of hazard intervals have been changed from 8 intervals to 10 intervals, which provides a better understanding of which ground motions contribute the most to seismic risk. Several variations on the number and size of the intervals were performed to establish the 10 intervals used in the final SPRA. ACUBE was used to process the CDF and LERF cutsets using the

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					Binary Decision Diagram (BDD) to obtain a more accurate result that reduces the over-counting that can occur with the minimum cutset upper bound (MCUB) when high probabilities are present in the cutsets. All SCDF cutsets were processed through ACUBE to obtain the SCDF. However, due to limitations in computer memory, not all SLERF cutsets were processed through ACUBE. The processing of the cutsets through ACUBE was refined to maximize the number of cutsets processed. For example, to process more SLERF cutsets, the SLERF for each hazard interval was processed through ACUBE separately, which allows processing nearly all of the cutsets for each initiator. Additional improvements in the processing of the cutsets for importance of the SSCs as well as for the HEPs and accident

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					sequence flags as documented in the SPRA quantification results. This finding is considered resolved. There is no effect on the results or conclusions of the SPRA.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
SPR-E2, E6	25-12	Internal events uncertainties are not reviewed and evaluated with respect to seismic impacts. (This F&O originated from SR SPR-E2)	The internal events uncertainties could have a significant impact on the seismic PRA if not sufficiently addressed. Per SR QU-E4, the above sources of uncertainty need to be addressed and their potential impact assessed.	Review the internal events uncertainties and assumptions and evaluate them with respect to the seismic PRA.	The internal events PRA model uncertainties were reviewed for applicability to the SPRA. The results are documented in the SPRA Model Development notebook. The review concluded that the uncertainties are either not applicable in the SPRA or they are already included as a source of uncertainty in the SPRA. This finding is considered resolved. There is no effect on the results or conclusions of the SPRA.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
SPR-E2, E6	25-13	The uncertainty of using of HRA calculator surrogates for HFEs was not evaluated. (This F&O originated from SR SPR-E2)	The FLEX related HFEs were reviewed: HEP-C-FLEX-LOADSHED-S(1-4), HEP-C-FLEX-REFUEL-S(1-4), HEP-C-FLEX-RIPS(1-4), and HEP-C-FLEX-VAC-S(1-4) and it was noticed that surrogate values are used to capture the contribution for the unique nature of the actions taken in FLEX that are outside the scope of the HRA calculator. This approach has inherent uncertainty that should be evaluated. Per SR QU-E4, this source of uncertainty need to be addressed and their potential impact assessed.	For the model uncertainty associated with the use of HEP calculator surrogate, IDENTIFY how the PRA model is affected (e.g., perform a sensitivity on the actions addressed with this technique).	A clarification was added to the seismic HRA notebook that discusses the use of surrogates in the HRA for the FLEX execution errors. Also, this was listed as a source of uncertainty, which was evaluated by a sensitivity. The results show a relatively minor impact on SCDF and SLERF. This finding is considered resolved.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
SPR-E4	25-14	There was no evaluation of correlation impact performed. (This F&O originated from SR SPR-E4)	The basis for correlation is documented in section 4.3 of the SA.3 notebook. Some of the fragility groups that appear to be significant may not be 100% correlated given that they have different orientations (e.g. vital buses) or have different designs (e.g. some vital bus inverters are 20kva and others are 15kva). Correlation doesn't have to be 100% correlated. There was no determination on whether or not the model is sensitive to correlation.	Perform appropriate sensitivity analyses of correlation for risk-significant fragilities. The fragility team may suggest additional possible correlation (for example based on orientation, design or other factors).	The basis for correlating SSCs is consistent with standard industry methods with respect to correlating redundant SSCs that are located in the same area and have similar design and installation. Orientation of the SSCs may be considered for uncorrelating SSCs if the SSCs are oriented differently. In the NAPS SPRA, redundant SSCs that have different orientation were modeled as uncorrelated only if the fragilities of the SSCs were significantly different. In the case of the vital bus panels, the HCLPF capacities of the panels are essentially the same regardless of orientation. Therefore, modeling these panels as correlated is considered appropriate. Likewise for the vital bus inverters, where one of the inverters has a higher power

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					rating (resulting in slightly larger mass) than the other three. The HCLPF capacities calculated for the inverters are not different enough to considered them uncorrelated due to the weight difference. Therefore, modeling of the inverters as correlated is considered appropriate. The other SSCs were reviewed and verified to be modeled appropriately with respect to correlation. This finding is considered resolved. There is no effect on the results or conclusions of the SPRA.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
SFR-D1, SPR-D1	26-1	Service water travelling screens were screened at the system analysis level as filters. As active moving components, they should have been passed to the fragility analysis as failure mode did not match the plugging that could be screened. (This F&O originated from SR SPR-D1)	The travelling screens were identified as components from the internal events PRA but were screened as a filter. The failure modes of a passive component like a filter do not have the same potential seismic failure modes and interaction as a travelling screen.	Evaluate the travelling screens for potential seismic failure and interactions that would impact functionality of service water.	The design and configuration of SW traveling screens were reviewed to determine if their failure could impact the SW pumps. The review concluded that seismic failure of the screens would not impact the SW pumps. The SPRA documentation was updated to document the conclusions of this review. This finding is considered resolved. There is no effect on the results or conclusions of the SPRA.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
SPR-B9	26-2	A potential for a flood in excess of the plant flood design in turbine building (which is assumed to fail in a seismic event) was identified during the plant walkdown. (This F&O originated from SR SPR-B9)	The postulated collapse failure of the turbine building could result in a flood with the inventories of circulating water, condensate, feedwater, condensate makeup, condensate polishing, main steam, turbine lube oil and any secondary side cooling water systems. The flood volume retained behind the wall would be reduced due to debris filling the retention volume. The flood sources alone would normally be in excess of what design basis flood protection in the form of walls/berms would be designed to contain. The propagation of this flood beyond the flood wall would impact all safety related AC and DC power distribution resulting a high	Confirm the effectiveness of the berm to control the impacts of the flood OR include the flood scenario in the SPRA.	Additional failure modes have been evaluated for the Turbine Building (TB). The seismically-induced structural damage within the TB has been evaluated to determine the potential for significant flooding that could adversely affect the function of core damage mitigating equipment as described in the disposition of finding F&O 24-2. The disposition of F&O 24-2 concluded that there were no flooding consequences from the bounding failure modes for the TB. Therefore, the existing flood barriers are adequate to protect the safety related AC and DC power distribution systems within the Emergency Switchgear Room (ESGR). This finding is considered resolved. There is no effect on the results or conclusions of the

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			Conditional Core Damage Probability (CCDP). The collapse of the turbine building may also preclude the use of an operator action to mitigate the flood by isolating the flood sources.		SPRA.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
SPR-C1	26-3	Human Action to secure CCW HX flood sources has a relatively high FV but is not credited without any evaluation of the potential impact on the model. (This F&O originated from SR SPR-C1)	This action could alter the SPRA CDF / LERF and there is no discussion of this potential conservatism.	JUSTIFY how this potential conservatism is imapcting the model. This can be done by performing a sensitivity analysis that show the effect of crediting this action. If this action is included in the evaluation, an appropriate feasibility assessment should be included.	The Operator action to isolate the SW flood is not credited in the SPRA due to the uncertainty in the size of the flood. This has been listed as a source of uncertainty. A sensitivity was performed to evaluate the impact of crediting this action if the flood size is lower and time is available to isolate it. The results show only a very little decrease in SCDF and SLERF if this action is credited for smaller breaks in the SW piping. This finding is considered resolved. There is no effect on the results or conclusions of the SPRA.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
SPR-A1	26-5	Process for determining earthquake caused initiating events is outlined in SA.3 section 3.1 using SPRAIG guidance - but all seismic fire interaction was screened. Industry experience has demonstrated that several of the SPRAIG guidance component type listed as "neglible" should still be considered. (This F&O originated from SR SPR-A1)	Recent industry experience and focus has heighted awareness and concern in the arena of high-energy cabinet fires. The following examples show some potential significance for this issue: • At Onagawa (2011) fire occurred in a non-seismically qualified power supply, but no count of total number of functional failures is provided. • At Kashiwazaki-Kariwa (2007) there were fires in non-seismically qualified equipment (it did not say how many) • At Kashiwazaki-Kariwa (2007) "Only minor damage to non-Class A or As SSCs was found, for example, a	Evaluate the impact of not screening the high energy electrical equipment from futher consideration in the areas/scenarios that were significant in the fire analysis.	The seismic-induced fire evaluation has been revised to include the evaluation of high energy electrical SSCs. The evaluation concluded that seismic risk due to high energy electrical SSCs is low and that no changes to the SPRA model were required to model seismic failure of high energy electrical SSCs. This finding is considered resolved. There is no effect on the results or conclusions of the SPRA.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			house transformer fire of Unit 3 • There are 2 transformer fires and one low voltage switchgear fire in the SQUG database • There was a medium voltage switchgear fire at a Kansai power sub-station (1995) • Recent (post Fukushima) shake table testing in Japan has shown HEAF in switchgear can occur • A recent study by FENOC and ABS concluded that HEAF due to seismic failure could not be excluded a priori (Screening of Seismic-Induced Fires by Lin, Wakefield and Reddington, PSAM 12)		