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

CNL-19-122

December 17, 2019

10 CFR 50.4
10 CFR 50.54(f)

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

Browns Ferry Nuclear Plant, Units 1, 2, and 3
Renewed Facility Operating License Nos. DPR-33, DPR-52, and DPR-68
NRC Docket Nos. 50-259, 50-260, and 50-296

Subject: **Seismic Probabilistic Risk Assessment for Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Response to NRC Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident**

References:
1. NRC Letter, "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," dated March 12, 2012 (ML12053A340)
2. EPRI Report 1025287, "Seismic Evaluation Guidance, Screening, Prioritization and Implementation Details [SPID] for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," dated November 2012 (ML12333A170)
3. TVA letter to NRC, "Tennessee Valley Authority's Seismic Hazard and Screening Report (CEUS Sites), Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," dated March 31, 2014 (ML14098A478)
4. NRC letter to TVA, "Browns Ferry Nuclear Plant, Units 1, 2 and 3 - Staff Assessment of Information Provided Pursuant to Title 10 of the Code of Federal Regulations Part 50, Section 50.54(f), Seismic Hazard Reevaluations Relating to Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident (TAC Nos. MF3764, MF3765 and MF3766)," dated April 21, 2015 (ML15090A745)

5. NRC Letter, “Final Determination of Licensee Seismic Probabilistic Risk Assessments Under the Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendation 2.1 ‘Seismic’ of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident,” dated October 27, 2015 (ML15194A015)

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

EPRI Report 1025287 (Reference 2) provides the guidance for screening, prioritization, and implementation details for the resolution of the Fukushima Near-Term Task Force (NTTF) Recommendation 2.1: Seismic. The EPRI Screening, Prioritization and Implementation Details (SPID) guidance was used to compare the reevaluated seismic hazard to the design basis seismic hazard for Browns Ferry Nuclear Plant, Units 1, 2, and 3. As described in Reference 3, Enclosure 2, it was initially proposed by TVA that a seismic probabilistic risk assessment was not warranted because the reevaluated ground motion response spectra (GMRS) was bounded by a seismic response spectra developed for the Individual Plant Examination of External Events (IPEEE). It was understood, however, that the GMRS exceeded the design basis response spectrum in the 1 to 10 Hz screening range utilized by NRC.

Reference 4 is the NRC Staff Assessment for Browns Ferry Nuclear Plant, Units 1, 2, and 3, seismic hazard submittals which concluded that the reevaluated seismic hazards described in Reference 3, Enclosure 2, are suitable for other actions associated with NTTF Recommendation 2.1: Seismic. NRC also concluded that that a seismic risk evaluation was merited.

Following additional interaction with NRC, TVA chose to perform a seismic probabilistic risk assessment for Browns Ferry Nuclear Plant.

In the Reference 5 letter to multiple licensees, NRC indicated that a seismic probabilistic risk assessment was required for Browns Ferry Nuclear Plant, Units 1, 2, and 3, and should be submitted to NRC by December 31, 2019.

The Enclosure to this letter provides the Seismic Probabilistic Risk Assessment Summary Report for Browns Ferry Nuclear Plant, Units 1, 2, and 3, as requested in Reference 5. The Enclosure provides the information requested in Item (8)B of the 50.54(f) letter associated with NTTF Recommendation 2.1: Seismic.

The updated Browns Ferry Nuclear Plant internal events probabilistic risk assessment model was used as the basis for the Seismic Probabilistic Risk Assessment. The internal events probabilistic risk assessment finding-level peer review Facts and Observations were dispositioned as discussed in Appendix A of the Enclosure as part of the development of the Seismic Probabilistic Risk Assessment.

This letter contains no new regulatory commitments.

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 17th day of December 2019.

Respectfully,

James T. Polickoski
Director, Nuclear Regulatory Affairs

Enclosure:

Browns Ferry Nuclear Plant, Units 1, 2, and 3, Seismic Probabilistic Risk Assessment in Response to 50.54(f) Letter with Regard to NTTF 2.1 Seismic Summary Report

cc (Enclosure):

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

ENCLOSURE

Browns Ferry Nuclear Plant, Units 1, 2, and 3
Seismic Probabilistic Risk Assessment
in Response to
50.54(f) Letter with Regard to NTTF 2.1 Seismic
Summary Report

Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3 Seismic Probabilistic Risk Assessment in Response to 50.54(f) Letter with Regard to NTTF 2.1 Seismic

SUMMARY REPORT

December 2019

Table of Contents

Executive Summary

In response to the 10 CFR 50.54(f) letter issued by the Nuclear Regulatory Commission (NRC) on March 12, 2012, a Seismic Probabilistic Risk Assessment (SPRA) has been developed for Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3. The point estimate results of the BFN SPRA are summarized below:

	Unit 1	Unit 2	Unit 3
Core Damage Frequency	6.30E-06	6.40E-06	7.13E-06
Large Early Release Frequency	3.00E-06	3.10E-06	3.31E-06

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

1.0 Purpose and Objective

Following the accident at the Fukushima Dai-ichi nuclear power plant resulting from the March 11, 2011, Great Tohoku Earthquake and subsequent tsunami, the Nuclear Regulatory Commission (NRC) established a Near-Term Task Force (NTTF) to conduct a systematic review of NRC processes and regulations and to determine if the agency should make additional improvements to its regulatory system. The NTTF developed a set of recommendations intended to clarify and strengthen the regulatory framework for protection against natural phenomena. Subsequently, the NRC issued a 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 Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3 has been performed, in accordance with the guidance in Electric Power Research Institute (EPRI) 1025287, “Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic,” [2] and previously submitted to the NRC [3]. That comparison concluded that the Ground Motion Response Spectra (GMRS), which was developed based on the reevaluated seismic hazard, exceeds the design basis seismic response spectrum in the 1 to 10 Hz range, and a seismic risk assessment is required. An SPRA has been developed to perform the seismic risk assessment for BFN in response to the 50.54(f) letter, specifically item (8) in Enclosure 1 of the 50.54(f) letter.

This report describes the SPRA developed for BFN 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 BFN, 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 the NRC to understand the inputs and methods used, the evaluations performed, and the decisions made as a result of the insights gained from the BFN SPRA.

2.0 Information Provided in This Report

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

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

(2) A summary of the methodologies used to estimate the SCDF and 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 SPRA 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 BFN Seismic Hazard Submittal [3]. Further, 50.54(f) letter Enclosure 1 paragraph 9 requesting information on the Spent Fuel Pool has been satisfied [4,5].

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

The SPID [2] defines the principal parts of a SPRA, and the BFN SPRA has been developed and documented in accordance with the SPID. The main elements of the SPRA performed for BFN in response to the 50.54(f) Seismic letter correspond to those described in Section 6.1.1 of the SPID, i.e.:

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

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

The BFN SPRA and associated documentation has been peer reviewed [6] against the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard [7, 8] in accordance with the process defined in Nuclear Energy Institute (NEI) 12-13 [9] as documented in the BFN SPRA Peer Review Report. The BFN SPRA, complete SPRA documentation, and details of the peer review are available for NRC review. Reference 7 is the ASME Code Case, which provides Part 5, Requirements for Seismic Events At-Power PRA. Reference 8 is the 2013 Addenda to ASME/ANS RA-S–2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications and is included because the Code Case has supporting requirements (SR) that back reference the SRs from the 2013 Addenda. Throughout this document, these two documents will be collectively referred to as the PRA Standard.

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

This submittal provides a summary of the SPRA development, results and insights, the peer review process and results, and the independent assessment, sufficient to meet the 50.54(f) information request in a manner intended to enable the 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 BFN seismic hazard analysis.
- Section 4 provides information related to the determination of seismic fragilities for BFN 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 and abbreviations 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 BFN SPRA peer review and independent assessment as well as a discussion of the open findings related to the BFN Internal Events PRA (IEPRA).

- Appendix B provides a response for each of the generic observations associated with the staff's review of SPRA reports provided in response to the March 12, 2012, 50.54(f) letter associated with reevaluated seismic hazards.

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

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

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

SPID Section 6.8 Item Description	Location in this Report
A report should be submitted to the NRC summarizing the SPRA inputs, methods, and results.	Entirety of the report addresses this.
The level of detail needed in the submittal should be sufficient to enable NRC to understand and determine the validity of all input data and calculation models used.	Entirety of the report addresses this. The key methods of analysis and referenced codes and standards are identified in the report.
The level of detail needed in the submittal should be sufficient to assess the sensitivity of the results to all key aspects of the analysis.	Entirety of the report addresses this. Results sensitivities are discussed in Section 5.7 (SPRA Quantification Sensitivity Analysis).
The level of detail needed in the submittal should be sufficient to make necessary regulatory decisions as a part of NTTF Phase 2 activities.	Entirety of the report addresses this.
It is not necessary to submit all the SPRA documentation for such an NRC review. Relevant documentation should be cited in the submittal and be available for NRC review in easily retrievable form.	Entirety of report addresses this. This report summarizes important information from the SPRA, with detailed information in lower-tier documentation.
Documentation criteria for an SPRA are identified throughout the PRA Standard. Utilities are expected to retain that documentation consistent with the PRA 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 BFN Seismic Hazard and Plant Response

3.1 Seismic Hazard Analysis

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

The seismic hazard analysis determines the annual frequency of exceedance (AFE) for selected ground motion parameters. The analysis involves use of earthquake source models, ground motion attenuation models, characterization of the site response (e.g., soil column), and accounts for the uncertainties and randomness of these parameters to arrive at the site seismic hazard. Detailed information regarding the BFN site hazard was provided to the NRC in the seismic hazard information submitted to the NRC in response to the NTTF 2.1 Seismic information request [3]. As further discussed below, a supplemental seismic hazard analysis has been performed for BFN [11].

3.1.1 Seismic Hazard Analysis Methodology

A supplemental seismic hazard analysis [11] was performed for the BFN SPRA in lieu of the NTTF 2.1 Submittal [3] since the site analysis develops the additional elements required for the SPRA such as updated seismic source catalog, Foundation Input Response Spectra (FIRS), hazard-consistent strain-compatible properties, and vertical ground motions. In addition, a site-specific surface geophysics investigation was performed to support the development of the BFN soil profiles.

The GMRS at BFN is defined at the foundation control point corresponding to the base of the Reactor Building (RB).

The following four FIRS were developed for the structures listed in Table 3.1-1 and are summarized below:

- ***GMRS/FIRS1*** – equivalent to the GMRS. GMRS/FIRS1 is located at a control point corresponding to an outcrop spectra at Elevation (EL) 515 ft, re: Mean Sea Level (MSL) at the top of the rock (Fort Payne Formation), at the base of the RB. The GMRS/FIRS1 soil profile includes 50 ft of compacted earth fill at the top with the zero depth corresponding to EL 565 ft, re: MSL. The soil profile for GMRS/FIRS1 is designated as GMRS/FIRS1/FIRS4 soil profile (since FIRS4 uses the same profile but corresponds to a control point defined at the ground surface as discussed below). The three branches used to develop the GMRS/FIRS1/FIRS4 soil profile adopted the same single branch below EL 515 ft, re: MSL, as discussed in the development of the FIRS3 soil profile below.
- ***FIRS2*** – corresponds to the outcrop spectra at EL 556 ft, re: MSL, located at a control point at the base of the Diesel Generating Building (DGB). The FIRS2 soil profile includes 12 ft of compacted earth fill from EL 565 ft to EL 553 ft, re: MSL, followed by 38 ft of crushed rock fill down to EL 515 ft, re: MSL, below which the rock subsurface (Fort Payne Formation and deeper units) profile ties in. The soil profile for FIRS2 is designated as FIRS2. The three branches used to develop the FIRS2 soil profile adopted the same single branch below EL 515 ft, re: MSL, as discussed in the development of the FIRS3 soil profile below.

- ***FIRS3 –*** corresponds to the outcrop spectra at EL 515 ft, re: MSL, located at a control point at the base of the Intake Pumping Station (IPS). The FIRS3 soil profile is identical to FIRS1, FIRS2, and FIRS4 when the top 50 ft of compacted earth fill/crushed rock fill are removed. Since this profile only includes the rock (Fort Payne Formation and deeper units), and based on the geophysics interpretations completed as part of the supplemental seismic hazard analysis [11], a single branch was used to model the epistemic uncertainty for the shear-wave velocity profile. A direct application of the SPID [2] guidelines relative to implementation of epistemic uncertainty by developing multiple branches would not have been appropriate for rock layers below EL 515 ft, re: MSL, due to the knowledge obtained from surface geophysics campaign completed during the supplemental seismic hazard analysis, which was collected around the site perimeter and confirmed the uniformity of those rock layers. While the geophysics campaign did not provide any basis for implementing epistemic uncertainty explicitly in our modeling, and a single branch was used, it is recognized that the uncertainty attributed to aleatory variability (0.25 natural log sigma) in principle does include a minor portion attributed to epistemic uncertainty. However, due to the interpretation of the underlying geology and the lack of any data collected through the geophysics program to support modeling epistemic uncertainty explicitly, a single value was used for modeling purposes in the form of the aleatory variability, with the recognition that a very minor component of that variability is in reality a representation of epistemic uncertainty, which would have an insignificant impact on the quantification of the total uncertainties. The soil profile for FIRS3 is designated as FIRS3.
- ***FIRS4 –*** corresponds to a surface-founded FIRS located at control point EL 565 ft, re: MSL, at the base of the Yard Equipment. The FIRS4 soil profile is identical to FIRS1 with the exception that the FIRS is located at the ground surface at EL 565 ft, re: MSL, as opposed to FIRS1, at which the FIRS was located at 50 ft depth (EL 515 ft, re: MSL). The soil profile for FIRS4 is the GMRS/FIRS1/FIRS4 soil profile defined above. The three branches used to develop the GMRS/FIRS1/FIRS4 soil profile adopted the same single branch below EL 515 ft, re: MSL, as discussed in the development of the FIRS3 soil profile above.

To perform the site response analyses for BFN, the random vibration theory approach was employed. This process is consistent with existing NRC guidance and the SPID. The guidance contained in Appendix B of the SPID on incorporating epistemic uncertainty in shear-wave velocities, non-linear dynamic properties and source spectra was followed for BFN in addition to development of High Frequency (HF) and Low Frequency (LF) controlling earthquakes (control motions) per recommendations in NRC Regulatory Guide (RG) 1.208 [13] for mean annual frequency of exceedance (MAFE) corresponding to 10^{-2}, 10^{-3}, 10^{-4}, 10^{-5}, $5x10^{-6}$, and 10^{-6} at reference rock.

Idealized shear-wave velocity profiles were developed incorporating the existing geotechnical data, onshore geophysics survey, and the derived geologic profile at depth derived for the BFN NTTF 2.1 Seismic Hazard submittal [3], along with the general guidelines included in the SPID to account for the soil profiles epistemic uncertainty and aleatory variability. The idealized shear-wave velocities developed for each of the three base-case soil profiles for GMRS/FIRS1/FIRS4, FIRS2, and FIRS3 are presented in Figures 3.1-1 to 3.1-6.

Table 3.1-1: Category I Structures and Geotechnical Foundation Material

Category I Structure	Geotechnical. Foundation Material	Applicable Elevation
Reactor Building	Fort Payne Formation at EL 515 ft, with 50 ft of compacted earth fill on top (to EL 565 ft)	515 ft
Diesel Generator Building	3 ft of compacted earth fill above 38 ft of crushed rock fill below EL 556 ft, followed by Fort Payne Formation at EL 515 ft, with 9 ft of compacted earth fill (to EL 565 ft) above	556 ft
Intake Pumping Station	Fort Payne Formation at EL 515 ft	515 ft
Ground Surface (Yard Equipment)	Fort Payne Formation at EL 515 ft, with 50 ft of compacted earth fill on top (to EL 565 ft)	565 ft

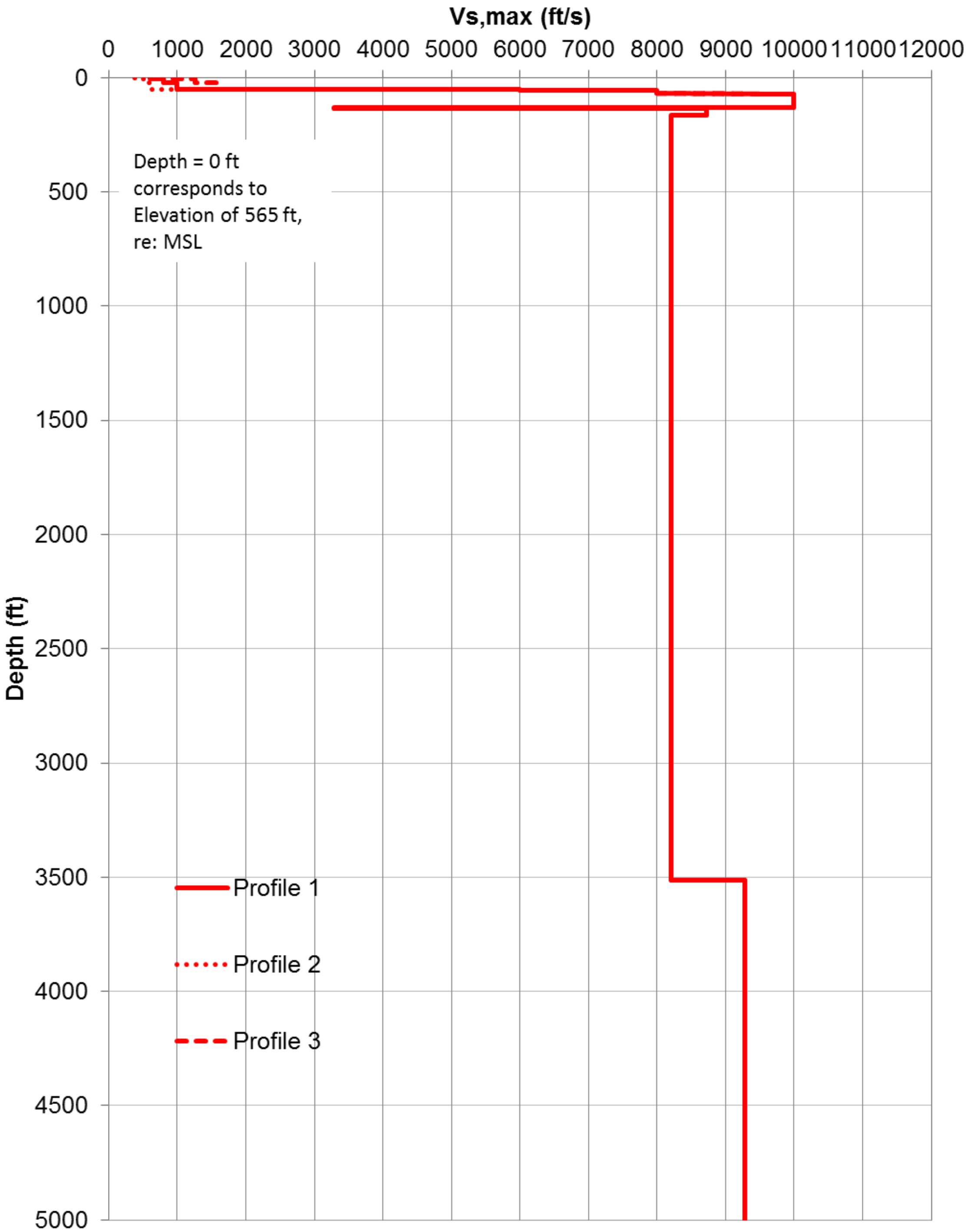

Figure 3.1-1: Idealized Shear-wave Velocity (V_S) Profiles (GMRS/FIRS1/FIRS4)

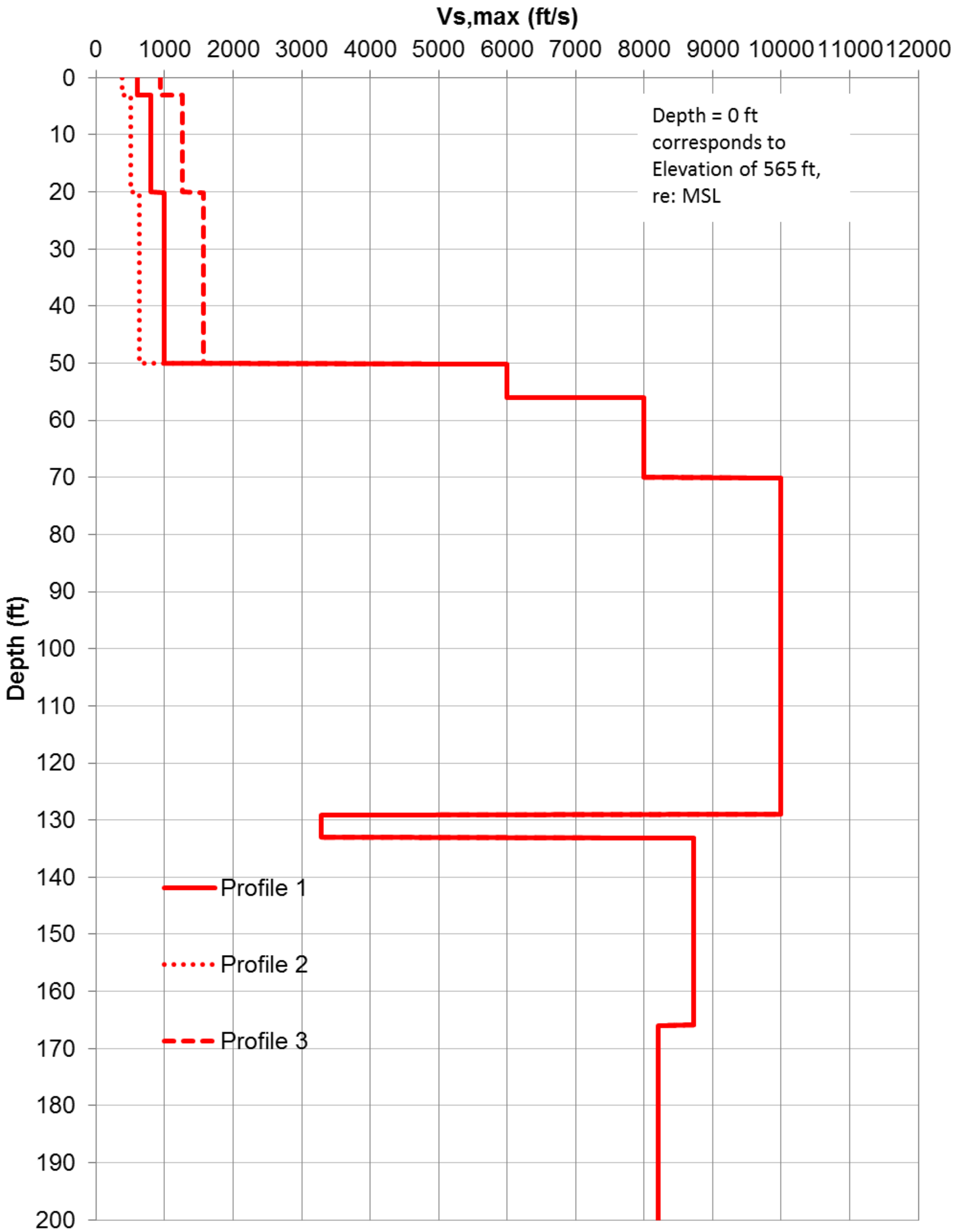

Figure 3.1-2: Idealized Shear-wave Velocity (V_S) Profiles in Top 200 ft (GMRS/FIRS1/FIRS4)

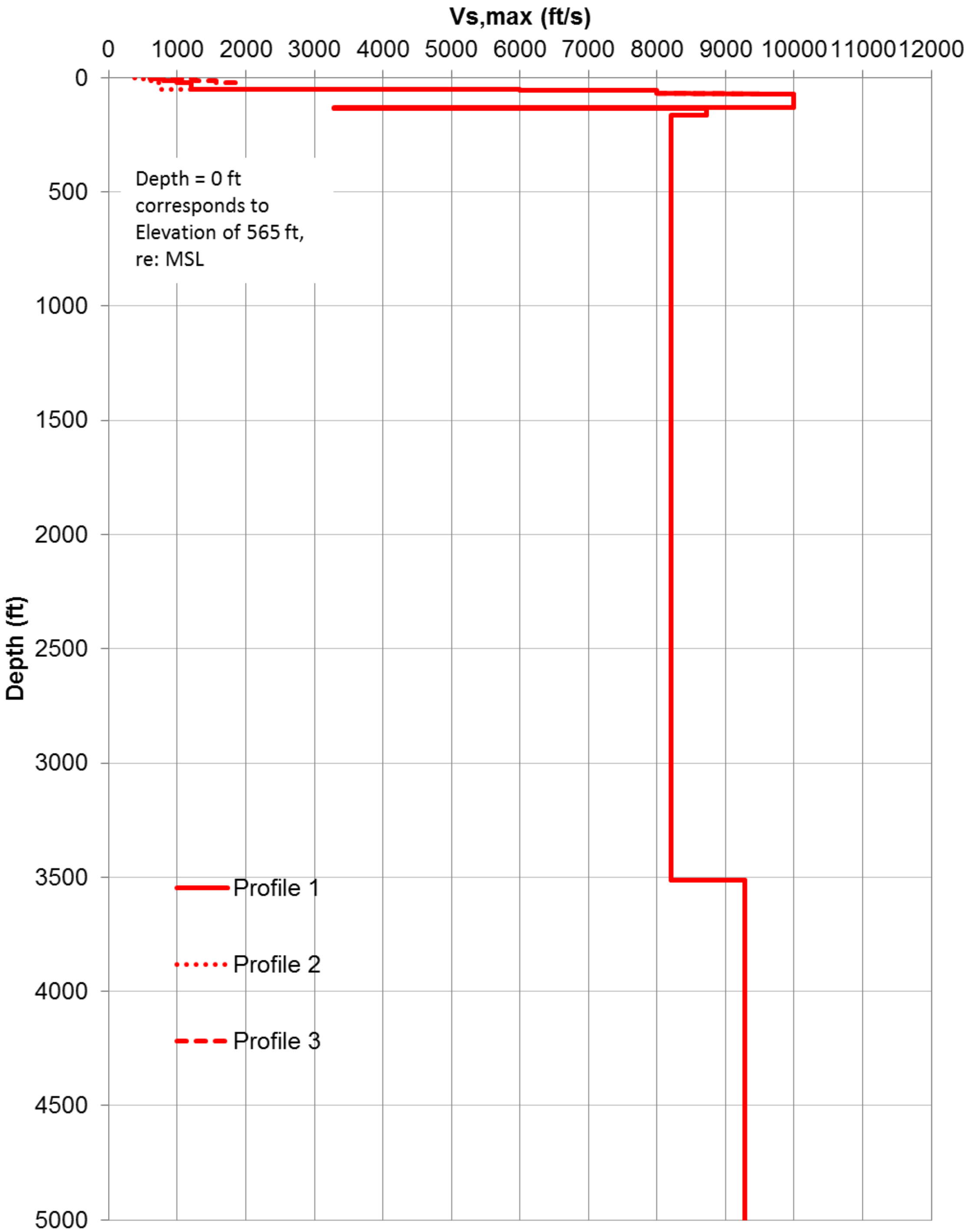

Figure 3.1-3: Idealized Shear-wave Velocity (V_S) Profile (FIRS2)

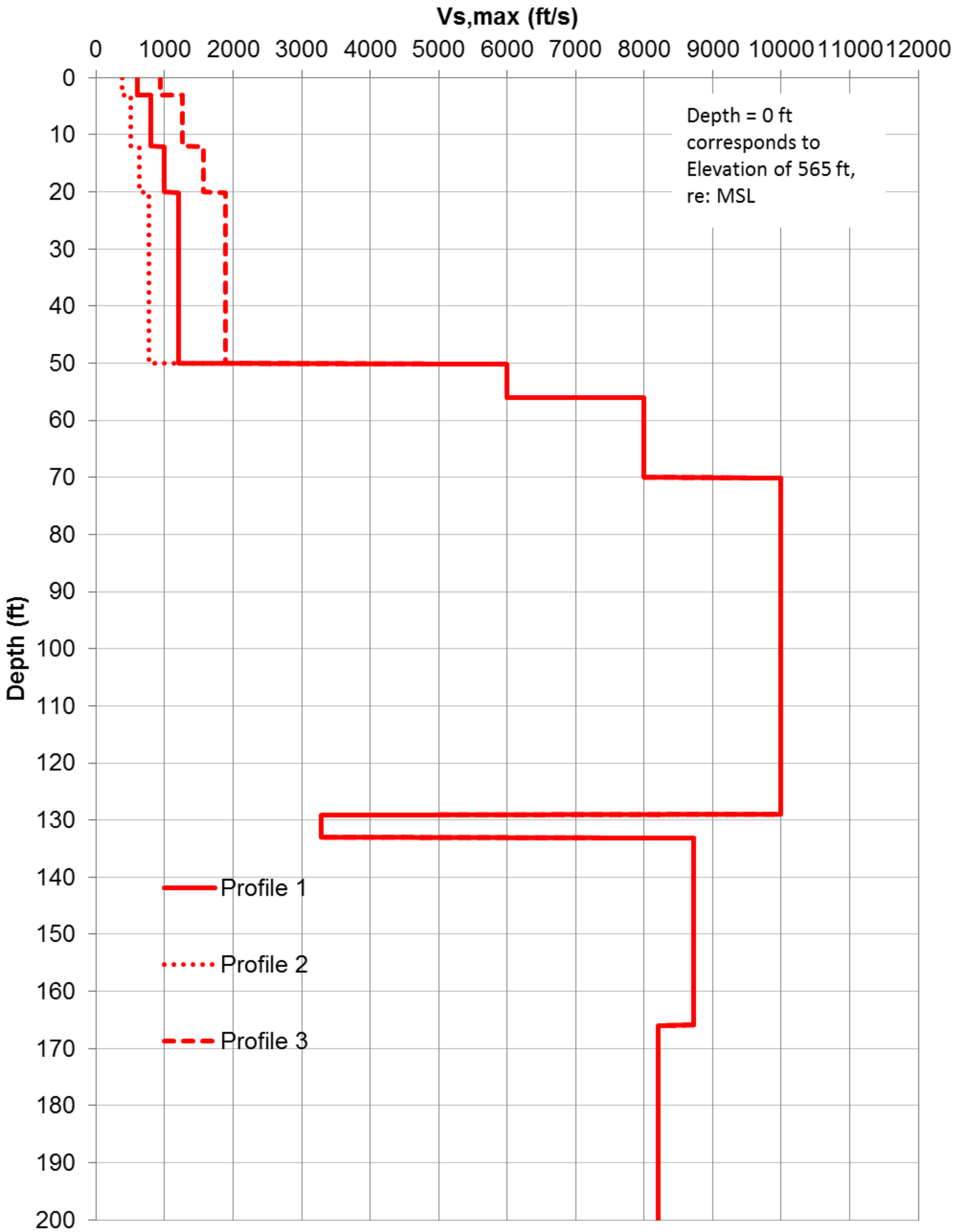

Figure 3.1-4: Idealized Shear-wave Velocity (V_S) Profile in Top 200 ft (FIRS2)

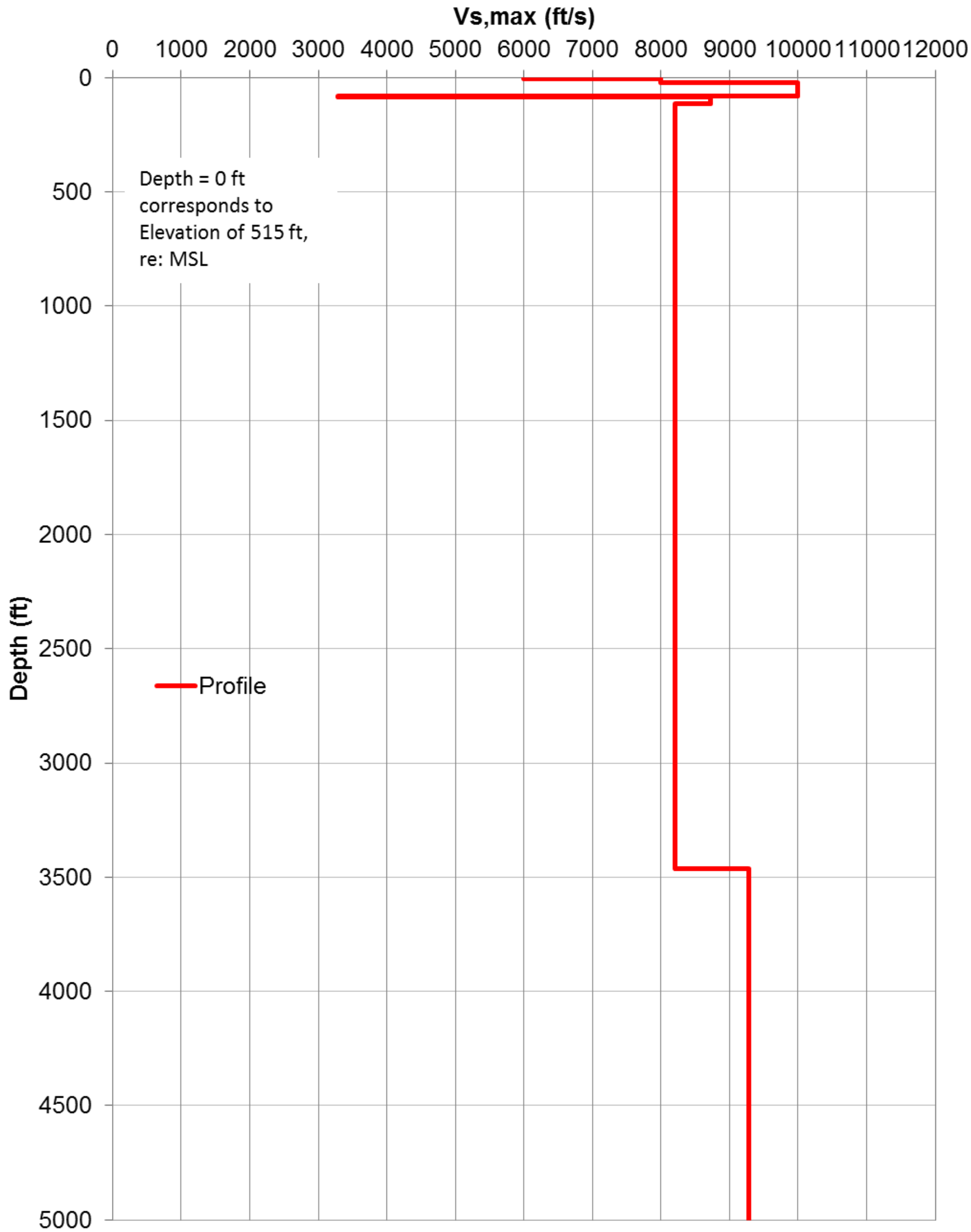

Figure 3.1-5: Idealized Shear-wave Velocity (V_S) Profile (FIRS3)

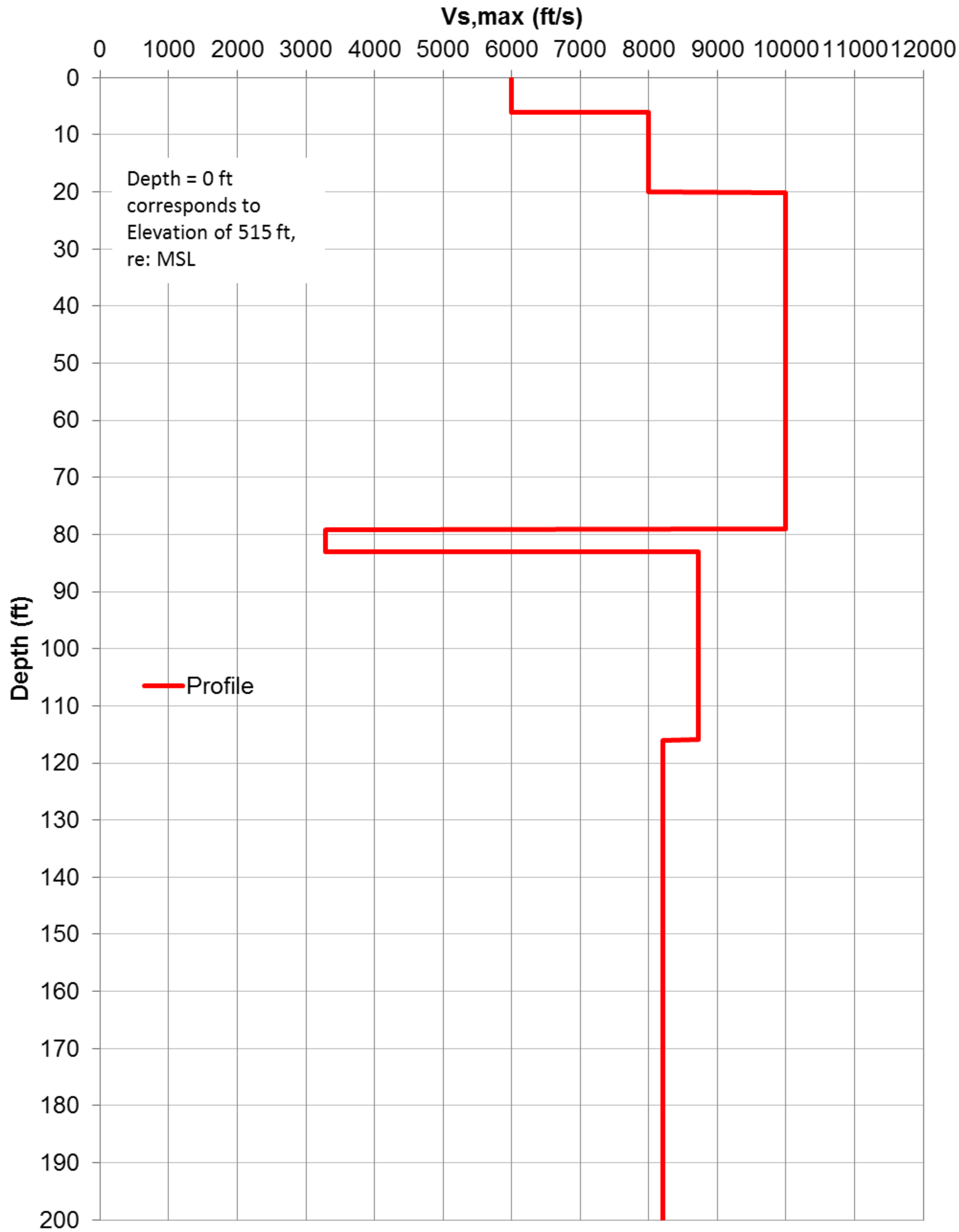

Figure 3.1-6: Idealized Shear-wave Velocity (V_S) Profile in Top 200 ft (FIRS3)

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

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

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

The seismic hazard calculations use a minimum earthquake moment magnitude of 5.0 since the cumulative absolute velocity filter is not used. Soil seismic hazard curves are calculated for frequencies of 0.5, 1, 2.5, 5, 10, and 25 Hz and peak ground acceleration (PGA) (100 Hz). Horizontal uniform hazard response spectrum (UHRS) are calculated for AFEs of 10^{-2}, 10^{-3}, 10^{-4}, 10^{-5}, $5x10^{-6}$, and 10^{-6}.

The GMRS and FIRS were developed in accordance with NRC RG 1.208 [13]. Sixty randomizations were generated for the site response for each epistemic branch in the soil logic tree, compared to a minimum of 30 recommended in the SPID. The site response analyses were completed using the HF and LF control motions. Site-specific horizontal hazard curves for each of the FIRS site conditions were used and were developed using Approach 3 of NUREG/CR-6728 [14].

Vertical spectra are developed using vertical-over-horizontal (V/H) scaling relations. The idealized V/H ratios are used to derive the vertical design response spectra from their horizontal equivalents.

For GMRS/FIRS1 and FIRS3, these FIRS are very close to Central and Eastern United States (CEUS) hard reference rock with the average time-weighted shear-wave velocity in the top 100 ft below the ground surface being 8,462 ft/s. As a result, the CEUS V/H scaling relation in NUREG/CR-6728 [14] was used with no modifications. For FIRS2 and FIRS4, the average time-weighted shear-wave velocity in the top 100 ft below those control points was equal to 2,304 ft/s and 1,613 ft/s, respectively. For these two FIRS, in the absence of CEUS V/H scaling relations appropriate for the FIRS, a logic tree was adopted to incorporate epistemic uncertainty by weighting alternative models consistent the methodology described in EPRI 3002004396 [15].

The reference earthquake ground motion to which the fragilities are referenced is represented by the horizontal UHRS at AFE corresponding to 1E-05 at the RB foundation control point. The PGA hazard curve is the ground motion parameter used for the SPRA.

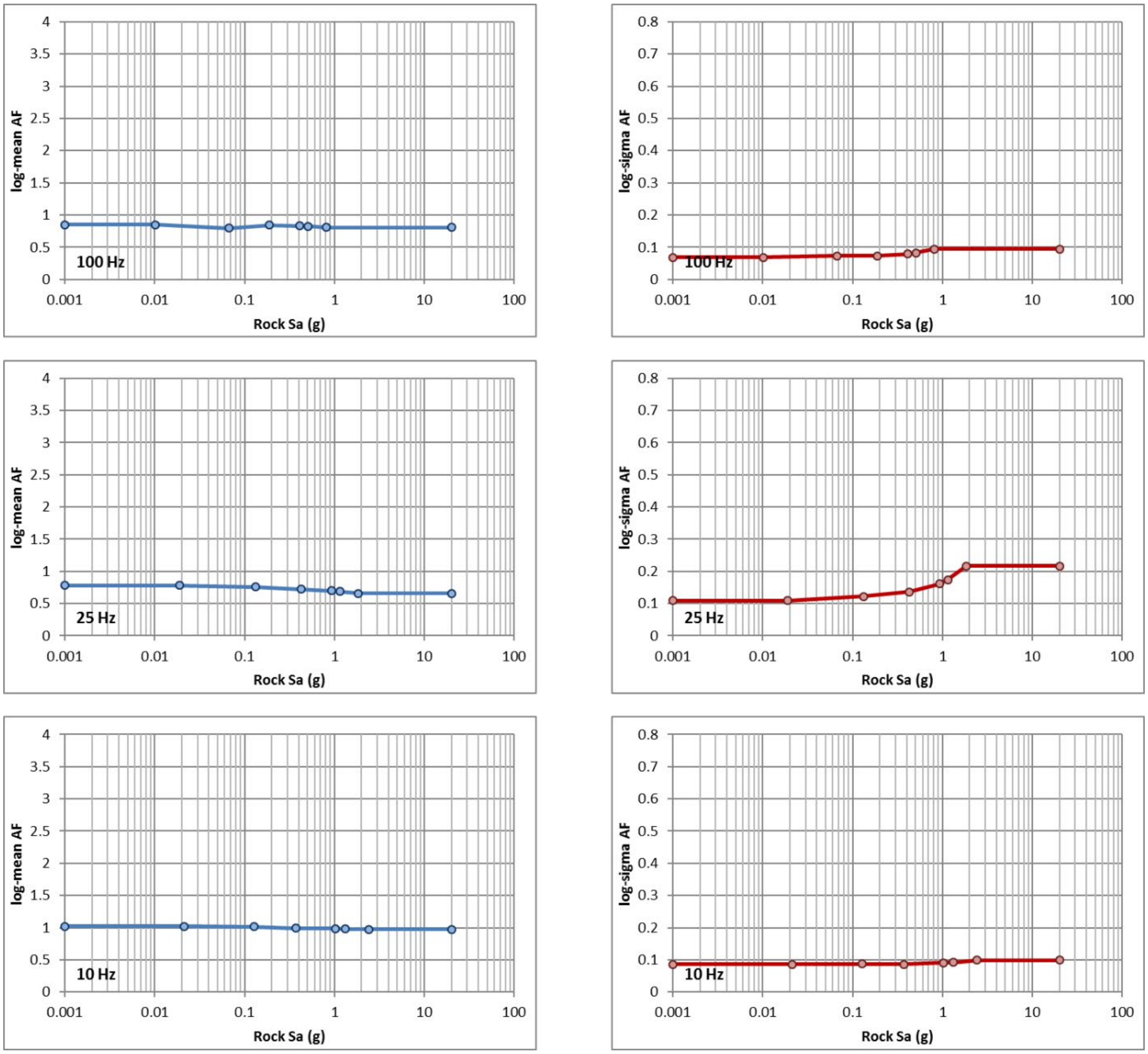

Figure 3.1-7: GMRS/FIRS1 Site Amplification Factor and Logarithmic Sigmas (100 Hz, 25 Hz, and 10 Hz)

3.1.2 Seismic Hazard Analysis Technical Adequacy

The BFN SPRA hazard methodology and analysis was subjected to an independent peer review against the pertinent requirements in the PRA Standard. After completion of the subsequent independent assessment, the full set of SRs was met. The seismic hazard analysis was determined to be acceptable for use in the SPRA.

The peer review assessment and subsequent disposition of peer review findings through an independent assessment are further described in Appendix A and references [6] and [16].

3.1.3 Seismic Hazard Analysis Results and Insights

Table 3.1-2 and Figure 3.1-8 present the mean and fractile exceedance frequencies for the control point corresponding to GMRS/FIRS1 at 100 Hz. Table 3.1-3 provides the final seismic hazard results used as input to the BFN SPRA, in terms of exceedance frequencies as a function of PGA level at the GMRS/FIRS1 control point.

Table 3.1-2: BFN GMRS/FIRS1 Mean and Fractile Exceedance Frequencies at PGA (100 Hz)

Amplitude (g)	Mean	Fractile Hazard Curves				
		0.05	0.16	0.5	0.84	0.95
0.0001	1.641E-01	9.997E-02	1.267E-01	1.630E-01	2.010E-01	2.320E-01
0.00025	1.300E-01	7.065E-02	1.023E-01	1.297E-01	1.600E-01	1.822E-01
0.0005	1.022E-01	4.698E-02	7.965E-02	1.015E-01	1.292E-01	1.469E-01
0.00075	8.487E-02	3.410E-02	6.495E-02	8.369E-02	1.087E-01	1.288E-01
0.001	7.232E-02	2.650E-02	5.419E-02	7.054E-02	9.440E-02	1.153E-01
0.0015	5.513E-02	1.785E-02	3.959E-02	5.333E-02	7.300E-02	9.608E-02
0.002	4.392E-02	1.325E-02	2.962E-02	4.191E-02	5.787E-02	8.388E-02
0.003	3.044E-02	8.529E-03	1.819E-02	2.887E-02	3.966E-02	6.835E-02
0.005	1.804E-02	4.685E-03	8.959E-03	1.656E-02	2.336E-02	4.933E-02
0.0075	1.159E-02	2.730E-03	4.808E-03	1.010E-02	1.582E-02	3.640E-02
0.01	8.428E-03	1.827E-03	3.128E-03	6.615E-03	1.189E-02	2.841E-02
0.015	5.345E-03	9.230E-04	1.688E-03	3.951E-03	8.150E-03	1.889E-02
0.02	3.821E-03	5.199E-04	1.031E-03	2.733E-03	5.848E-03	1.377E-02
0.03	2.301E-03	2.411E-04	4.392E-04	1.405E-03	3.885E-03	8.889E-03
0.05	1.122E-03	9.118E-05	1.456E-04	5.010E-04	2.032E-03	4.701E-03
0.075	5.903E-04	4.492E-05	6.967E-05	2.106E-04	9.901E-04	2.629E-03
0.1	3.603E-04	2.868E-05	4.532E-05	1.193E-04	5.485E-04	1.567E-03
0.15	1.680E-04	1.547E-05	2.465E-05	5.804E-05	2.190E-04	6.621E-04
0.2	9.208E-05	9.688E-06	1.512E-05	3.579E-05	1.143E-04	3.132E-04
0.3	3.654E-05	4.358E-06	7.123E-06	1.652E-05	4.991E-05	1.041E-04
0.5	1.062E-05	1.211E-06	2.332E-06	5.939E-06	1.666E-05	2.786E-05
0.75	3.838E-06	3.463E-07	7.886E-07	2.183E-06	6.492E-06	1.099E-05
1	1.799E-06	1.179E-07	3.179E-07	9.992E-07	3.099E-06	5.613E-06
1.5	5.706E-07	1.591E-08	6.850E-08	2.785E-07	9.617E-07	2.022E-06

Table 3.1-2: BFN GMRS/FIRS1 Mean and Fractile Exceedance Frequencies at PGA (100 Hz)

Amplitude (g)	Mean	Fractile Hazard Curves				
		0.05	0.16	0.5	0.84	0.95
2	2.350E-07	1.645E-09	1.693E-08	1.010E-07	3.933E-07	9.215E-07
3	5.961E-08	4.555E-17	5.849E-10	1.905E-08	9.632E-08	2.781E-07
5	8.475E-09	5.263E-29	1.611E-14	1.532E-09	1.254E-08	4.847E-08
7.5	1.488E-09	2.202E-29	1.492E-21	1.377E-10	1.858E-09	9.810E-09
10	3.922E-10	2.200E-29	1.069E-24	2.207E-11	4.271E-10	2.622E-09

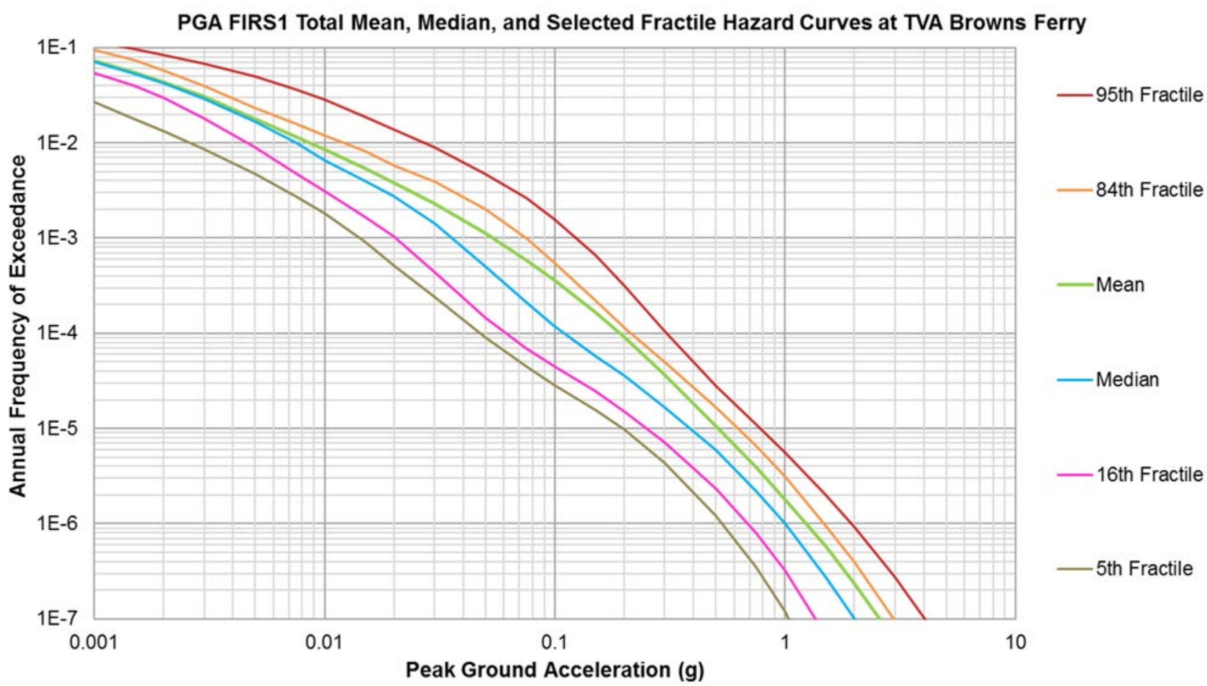

Figure 3.1-8: PGA (100 Hz) GMRS/FIRS1 Soil Profile Fractile Hazard Curves for BFN

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

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

The Central and Eastern United States Seismic Source Characterization (CEUS-SSC) concluded its data gathering efforts in 2008. As a result, a literature search of published and unpublished data was completed to identify any data that may have an impact on the CEUS-SSC, or any other site-specific modifications based on new information. An updated CEUS-SSC seismicity catalog was developed for the whole CEUS-SSC Study Region for the period of January 1, 2009 through April 30, 2016 for the region encompassed by the 250-mile (400-km) radius around the BFN site. The final seismicity catalog used for the BFN Probabilistic Seismic Hazard Analysis (PSHA) is the combination of the original CEUS-SSC seismicity catalog (1568 through 2008) and the updated BFN site regional catalog (January 1, 2009 through April 30, 2016). After the review and study of new information, it was concluded that the CEUS-SSC recurrence parameters did not require an update.

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

3.1.3.2 Horizontal and Vertical GMRS

This section provides the control point horizontal and vertical GMRS.

The GMRS at the control point is provided in Table 3.1-3 and plotted in Figure 3.1-9. The insights are summarized in Section 3.1.1 and further described in detail in the BFN PSHA report [11].

3.1.3.2.1 Vertical GMRS

Vertical ground motions were developed by applying V/H ratios to the horizontal GMRS and FIRS. For GMRS/FIRS1 and FIRS3, these FIRS are very close to the CEUS hard reference rock with the average time-weighted shear-wave velocity in the top 100 ft below the ground surface being 8,462 ft/s. As a result, the CEUS V/H scaling relation in NUREG/CR-6728 [14] was used with no modifications. In addition to the V/H scaling relations developed above for development of the vertical GMRS and FIRS, V/H scaling relations were also developed at annual frequency of exceedance of 10-5. As stated above, for GMRS/FIRS1 and FIRS3, these FIRS are very close to CEUS hard rock with an average time-weighted shear wave velocity in the top 100 ft below the ground surface of 8,462 ft/s. As a result, the CEUS Vertical over Horizontal (V/H) scaling relation in NUREG/CR-6728 (USNRC, 2001) was used with no changes (corresponding to PGA between 0.2g and 0.5g for the GMRS level where the PGA was equal to 0.254g). At annual frequency of exceedance of 10-5, the same V/H scaling relation could be used for GMRS/FIRS1 and FIRS3 in spite of the 10-5 hazard level PGA being 0.5155g.

For FIRS2 and FIRS4, the average time-weighted shear-wave velocity in the top 100 ft below those control points was equal to 2,304 ft/s and 1,613 ft/s, respectively. For these two FIRS, in the absence of CEUS V/H scaling relations appropriate for the FIRS, a logic tree was adopted to incorporate epistemic uncertainty by weighting alternative models consistent the methodology described in EPRI 3002004396 [15]. The development of the V/H ratios is documented in the BFN PSHA report.

Table 3.1-3 summarizes the horizontal and vertical response spectra at the control point. Figure 3.1-9 provides a plot of the vertical and horizontal GMRS as well as V/H ratios.

Table 3.1-3 Smoothed Horizontal and Vertical GMRS/FIRS1/FIRS3 and V/H Ratios

Frequency (Hz)	Horizontal GMRS/FIRS1/FIRS3 (g)	Vertical GMRS/FIRS1/FIRS3 (g)	V/H Ratio
0.1	1.64E-02	1.23E-02	0.750
0.125	2.06E-02	1.54E-02	0.750
0.15	2.49E-02	1.87E-02	0.750
0.2	3.38E-02	2.53E-02	0.750
0.3	5.30E-02	3.97E-02	0.750
0.4	7.28E-02	5.46E-02	0.750
0.5	8.94E-02	6.71E-02	0.750
0.6	9.98E-02	7.49E-02	0.750
0.7	1.06E-01	7.98E-02	0.750
0.8	1.11E-01	8.36E-02	0.750
0.9	1.17E-01	8.74E-02	0.750
1	1.26E-01	9.41E-02	0.750
1.25	1.55E-01	1.16E-01	0.750
1.5	1.85E-01	1.38E-01	0.750
2	2.36E-01	1.77E-01	0.750
2.5	2.63E-01	1.97E-01	0.750
3	2.82E-01	2.12E-01	0.750
4	3.21E-01	2.40E-01	0.750
5	3.54E-01	2.65E-01	0.750
6	3.89E-01	2.92E-01	0.750
7	4.22E-01	3.17E-01	0.750
8	4.53E-01	3.40E-01	0.750
9	4.80E-01	3.60E-01	0.750
10	5.01E-01	3.75E-01	0.750
12.5	5.28E-01	4.07E-01	0.771
15	5.34E-01	4.21E-01	0.788
20	5.08E-01	4.20E-01	0.826
25	4.67E-01	4.11E-01	0.880
30	4.27E-01	4.00E-01	0.937
35	3.91E-01	3.83E-01	0.981
40	3.64E-01	3.80E-01	1.042
45	3.50E-01	3.86E-01	1.102
50	3.43E-01	3.86E-01	1.124
60	3.29E-01	3.75E-01	1.137
70	3.01E-01	3.40E-01	1.128
80	2.75E-01	2.99E-01	1.090
90	2.60E-01	2.70E-01	1.038
100	2.54E-01	2.54E-01	1.000

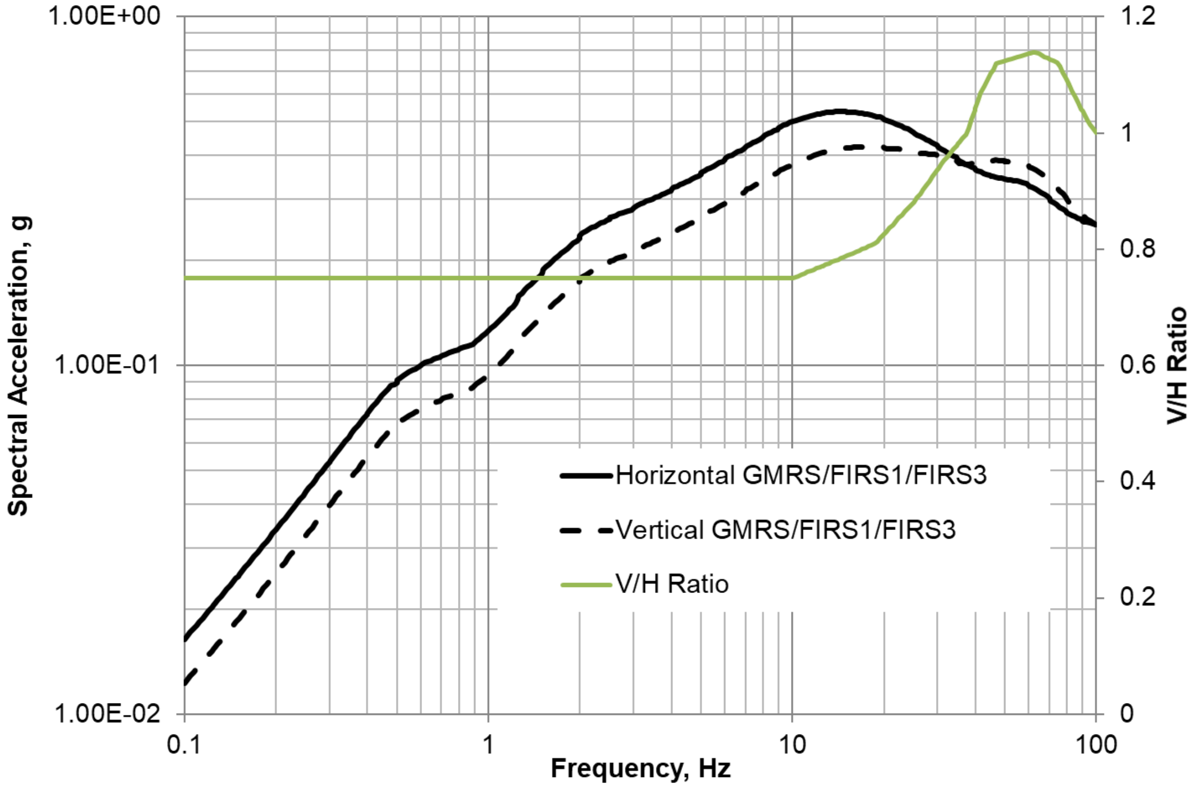

Figure 3.1-9: Horizontal and Vertical GMRS/FIRS1/FIRS3 and V/H Ratio

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

The BFN SPRA used the supplemental seismic hazard analysis documented in the BFN PSHA report [11]. Table 3.1-3 and Figure 3.1-9 provide the vertical and horizontal GMRS.

A site-specific surface geophysics program encompassing BFN was completed to better define the shear-wave velocities of the BFN units. The existing geotechnical information available at the project site was used to characterize the depth of the various units, e.g., Fort Payne formation, and the shear-wave velocities from the geophysics were then assigned to their corresponding units, since the geophysics surveys profiles were acquired at the perimeter of the BFN site.

Figures 3.2-1 to 3.2-3 compare the NTTF 2.1 Seismic Hazard submittal [3], assessed by the NRC staff, with the SPRA Supplemental Seismic Hazard Analysis.

Figures 3.2-1 and 3.2-2 show the idealized site profiles developed. The key difference between the base profile developed in the current study and the NTTF 2.1 Submittal study is that the profile developed in the current study is stiffer (faster) except for the top 25 ft, where the profile developed is slightly softer (slower) compared to the NTTF 2.1 Submittal. Due to the uniformity observed from the onshore geophysics program encompassing BFN, a single branch was used in the SPRA Supplemental Seismic Hazard Analysis. Another difference included the addition of a softer layer at approximate EL 435 ft, re: MSL with a significant impedance reversal. Sensitivity analyses were performed and documented in the BFN PSHA report, where this softer layer was removed, and it showed an insignificant impact on the final PSHA results.

Figure 3.2-3 compares the NTTF 2.1 Submittal GMRS and the current GMRS/FIRS1/FIRS3. Overall, the shapes of the spectra are comparable. At the lower frequencies, there are insignificant differences. At higher frequencies, GMRS/FIRS1/FIRS3 is lower than or equal to the NTTF 2.1 Seismic Hazard Submittal. The small differences between the two spectra could be attributed to a number of factors. Excluding factors like the use of different software or variations in the randomization algorithms for the base-case profiles, the total uncertainty (combined epistemic uncertainty and aleatory variability) are lower in the SPRA Supplemental Seismic Hazard Analysis compared to the NTTF 2.1 Submittal. This is due to the site-specific measurements that were completed at BFN, which confirmed the relative uniformity of the site. Another difference includes the slightly softer (slower) subsurface conditions in the top 25 ft on average. Reduction of the total sigmas would be expected to reduce the mean hazard directly even if the profiles were identical. The slightly softer (slower) subsurface conditions in the top 25 ft would be expected to slightly shift the spectrum towards the lower frequency, which is what is observed in Figure 3.2-3, where the spectral peak is at a slightly lower frequency (approximately 15 Hz as opposed to 25 Hz).

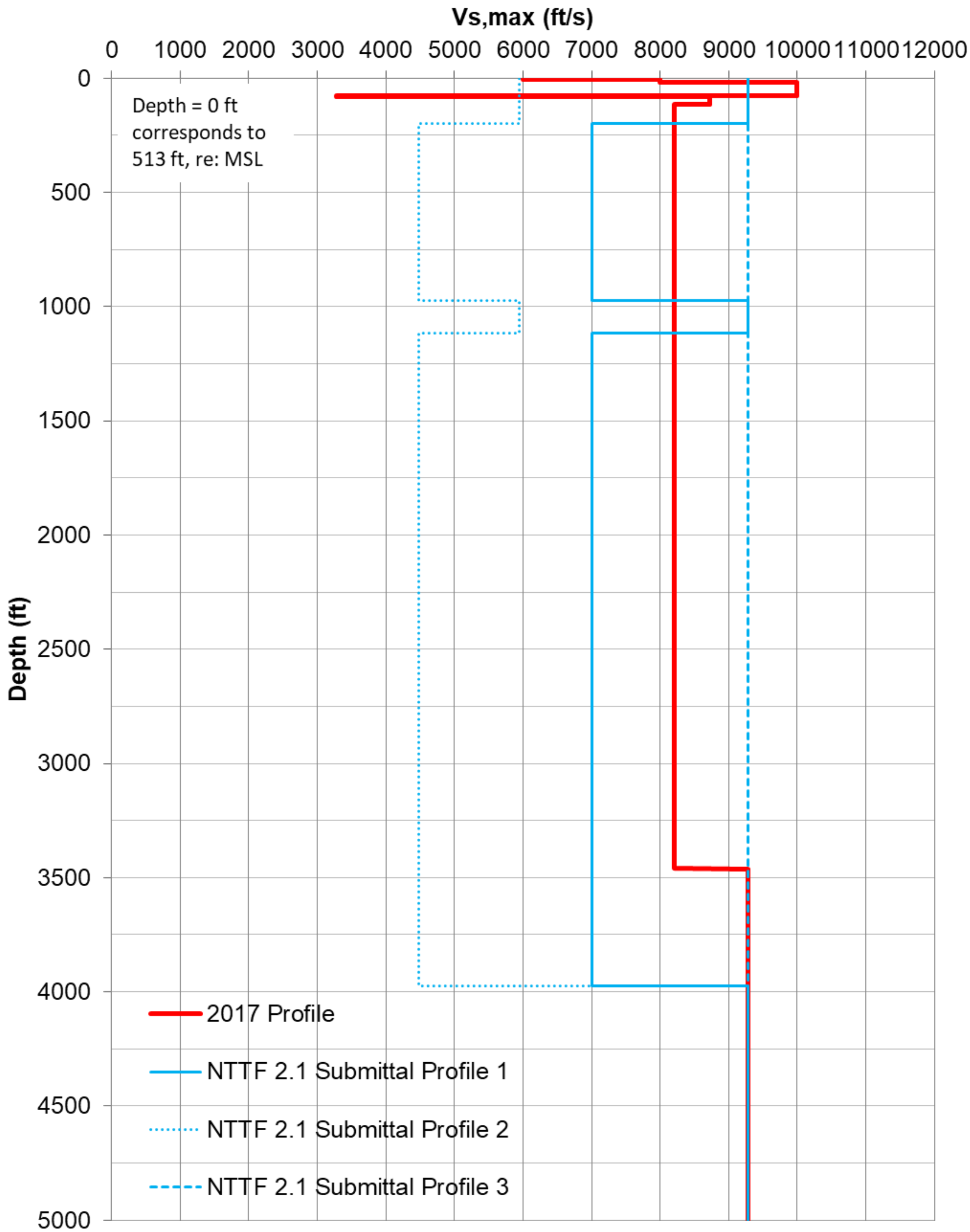

Figure 3.2-1: Comparison of Base-Case Soil Profiles NTTF 2.1 Seismic Hazard Submittal and SPRA Supplemental Seismic Hazard Analysis

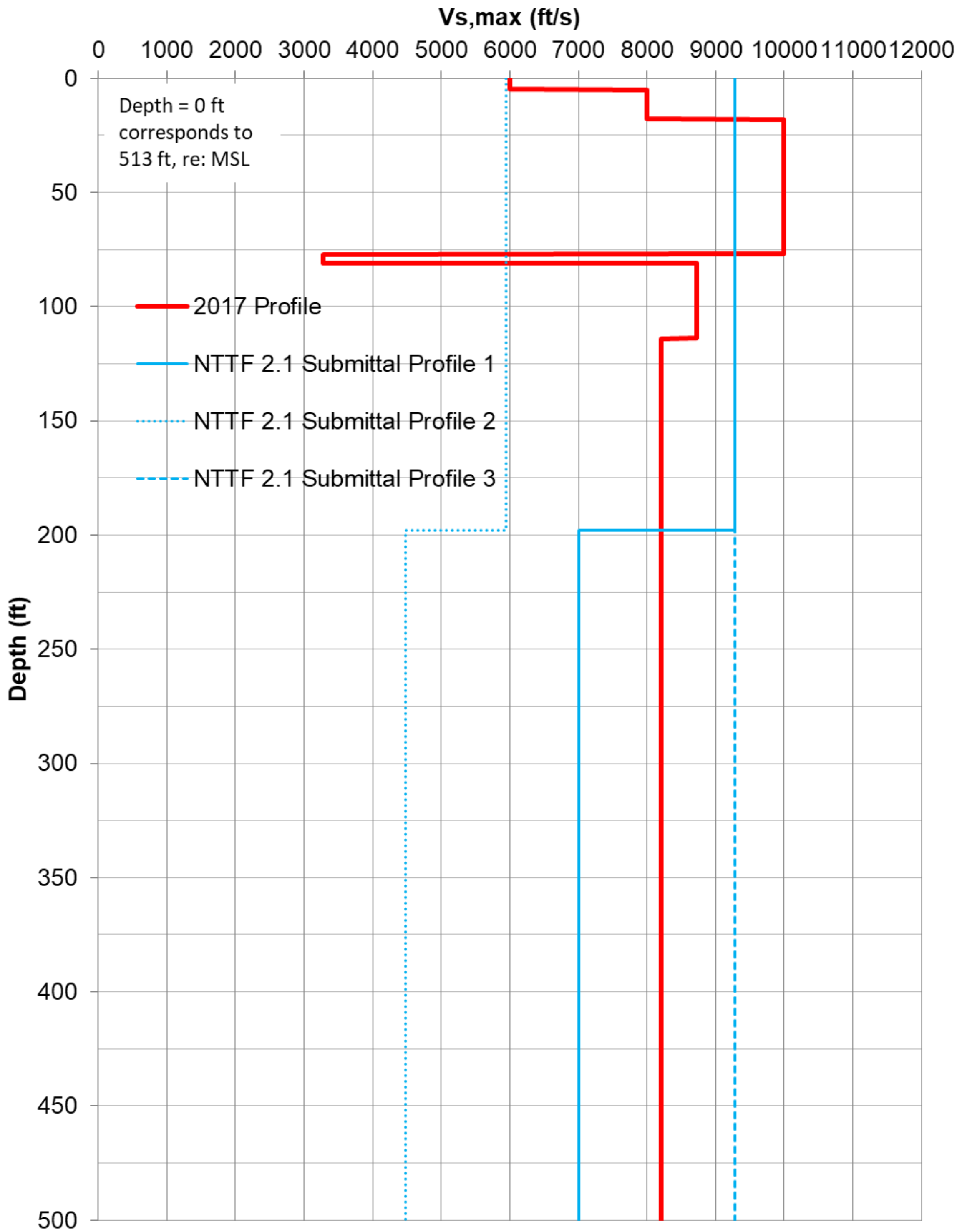

Figure 3.2-2: Comparison of Base-Case Soil Profiles NTTF 2.1 Seismic Hazard Submittal in Top 500 ft and SPRA Supplemental Seismic Hazard Analysis

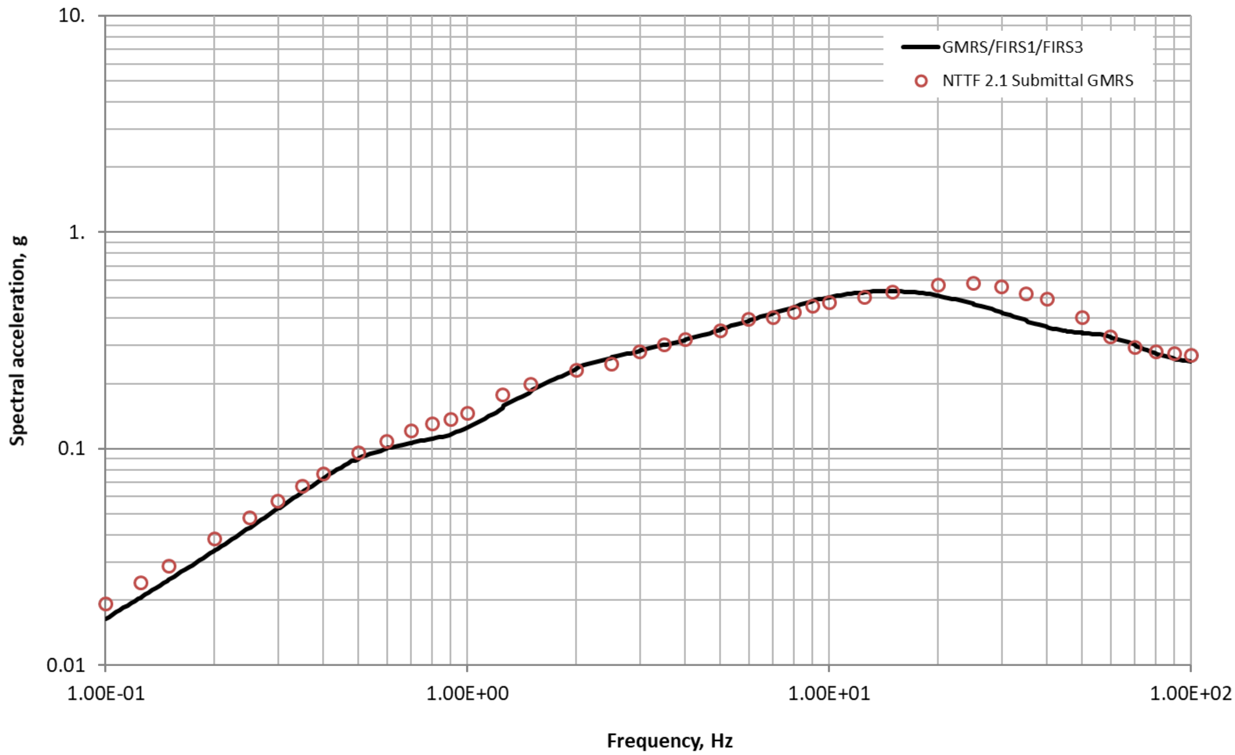

Figure 3.2-3: Comparison of Horizontal GMRS/FIRS1/FIRS3 NTTF 2.1 Seismic Hazard Submittal and Seismic PRA Supplemental Seismic Hazard Analysis

3.3 Soil Failure and Fragility Analysis

The SPRA soil failure and fragility analysis is performed in the report CJC-BFN-C-001 [17]. Soil failure modes considered in the analysis include liquefaction, seismic-induced settlements, seismic-induced lateral deformation, slope stability, sliding of earth and building structures, and seismic bearing capacity. The evaluations performed and described in this report followed an overall graded approach for developing soil failure mode fragilities for inputs into the BFN SPRA model. The graded approach uses increasing levels of rigor for screening out or estimating soil fragilities depending upon the contribution to risk of a given soil failure mode.

Seismic-induced soil failure primarily results in lateral and/or vertical displacements and increased lateral pressure on building walls for embedded structures founded on rock at the BFN site. CJC-BFN-C-001 provides estimates of vertical and lateral deformations due to soil failures for earthquakes ranging from AFE of 1E-4 to 1E-7.

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

The effect of soil deformation on buried piping strain levels are developed in report CJC-BFN-C-001 [17]. Piping fragility is determined based on the strain levels induced in the embedded piping due to localized and distributed soil deformations. The localized deformation case resulted in the largest strain demands in the piping. Strain levels are computed for ground motion levels corresponding to AFE 1E-4 to 1E-7.

The ground motion levels and associated SAFs for the analysis are taken from the BFN PSHA report [11].

Table 3.3-1: BFN Soil Failure and Fragility Analysis

Structure	Geotechnical Foundation Material	Evaluation
Reactor Building and Steel Containment Vessel	Shale/limestone bedrock	Screened out[1]
Reinforced Concrete Chimney	Bedrock	Screened out
Intake Pumping Station (IPS), including the Residual Heat Removal Service Water (RHRSW) Intake Structure	Bedrock	Screened out[1]
Gate Structure No. 2 and 3	Shale/limestone bedrock	Screened out
Diesel Generator Building (DGB)	Compacted earth/ crushed rock fill	Screened out
Standby Gas Treatment Building	Compacted earth backfill	Screened out
Off-gas Treatment Building	Bedrock	Screened out
Equipment Access Lock	Compacted earth backfill	Vertical Settlement and Lateral Deformation
Vacuum Pipe Building	Compacted earth backfill	Vertical Settlement and Lateral Deformation
Condensate Storage Tanks (CST)	Compacted earth backfill	Vertical Settlement and Lateral Deformation
Retaining Wall Between DGB & Radwaste Building	Compacted earth backfill	Screened out
Intake Channel Slope	In-situ soils	Screened out
Earth Berm south of Reactor Building	Compacted backfill and in-situ soils	Vertical Settlement and Lateral Deformation
North Bank of Cool Water Channel	In-situ soils	Vertical Settlement and Lateral Deformation
Cool Water Discharge Dike	In-situ soils and Hydro fill	Screened out

[1]Lateral soil pressures calculated

3.3.1 Soil Failure and Fragility Analysis Technical Adequacy

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

The peer review assessment, and subsequent disposition of peer review findings through an independent assessment [6] are further described in Appendix A [16].

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

4.1 Seismic Equipment List

For the BFN SPRA, a seismic equipment list (SEL) was developed to include SSCs that are important to achieving safe shutdown following a seismic event and to mitigating radioactivity release if core damage occurs, and that are included in the SPRA model. The methodology used to develop the SEL is consistent with the guidance provided in EPRI 3002000709 [18].

4.1.1 SEL Development

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

The resulting SEL includes a total of about 6,800 component entries for all three units combined (including common components). The final SEL was documented in the BFN SPRA Seismic Equipment List [19].

Table 4.1-1: List of Systems Considered in the SEL Development

System Number	System	Mitigation Potential	In IE PRA	In IPEEE	In 2.3 SWEL	In SEL	Containment Isolation
001	Main Steam	Yes	Yes	Yes	Yes	Yes	Yes
002	Condensate	No	Yes	No	No	Yes	No
003	Feedwater System	No	Yes	Yes	Yes	Yes	Yes
004	Hydrogen Water Chemistry	No	No	No	No	No	No
005	Extraction Steam	No	No	No	No	No	No
006	Heater Drains & Vents	No	No	No	No	No	No
008	Turbine Drains & Miscellaneous Piping	No	No	No	No	No	No
010	Rx Vessel Vents & Drains	No	No	Yes	Yes	Yes	No
012	Auxiliary Boiler	No	No	No	No	No	No
018	Fuel Oil	Yes	Yes	Yes	No	Yes	No
020	Lubricating Oil	No	No	No	No	No	No
023	Residual Heat Removal Service Water System	Yes	Yes	Yes	Yes	Yes	No
024	Raw Cooling Water	No	Yes	No	No	Yes	No
025	Raw Service Water	No	No	No	No	No	No
026	High Pressure Fire Protection	No	Yes	No	No	Yes	No
027	Condenser Circulating Water	No	Yes	No	No	Yes	No
030	Normal Ventilation	Yes	Yes	Yes	Yes	Yes	No
031	CREV/Ventilation	No	No	No	No	No	No
031	Chillers	Yes	Yes	No	Yes	Yes	No
032	Control Air	Yes	Yes	Yes	Yes	Yes	No
033	Service Air	Yes	Yes	No	No	Yes	No
034	Vacuum Priming	No	No	No	No	No	No
035	Generator Hydrogen Cooling	No	No	No	No	No	No
036	Auxiliary Boiler FW SEC Treatment	No	No	No	No	No	No
037	Gland Seal Water	No	No	No	No	No	No
040	Station Drainage	No	No	No	No	No	No
043	Chemistry Heat	No	No	Yes	No	Yes	No
044	Building Heat	No	Yes	No	No	Yes	No
046	Feedwater Control	Yes	Yes	No	No	No	No
047	EHC Control	Yes	Yes	No	No	Yes	No
049	Breathing Air	No	No	No	No	No	No
050	Sodium Hypochlorite	No	No	No	No	No	No
051	Raw Water Chlorination	No	No	No	No	No	No
052	Seismic Monitoring	No	No	No	No	No	No
053	Demineralizer Backwash Air	No	No	No	No	No	No
055	Annunciators	No	No	No	No	No	No
056	Temperature Monitoring	No	No	No	No	No	No
063	Standby Liquid Control	Yes	Yes	Yes	Yes	Yes	Yes
064A	Primary Containment	Yes	Yes	Yes	Yes	Yes	Yes
064B	Containment Purge	No	No	Yes	Yes	Yes	Yes
064C	Secondary Containment	Yes	Yes	Yes	Yes	Yes	Yes
064D	Primary Containment Isolation System	Yes	Yes	Yes	Yes	Yes	Yes

Table 4.1-1: List of Systems Considered in the SEL Development

System Number	System	Mitigation Potential	In IE PRA	In IPEEE	In 2.3 SWEL	In SEL	Containment Isolation
065	SGT	Yes	Yes	No	No	Yes	No
066	Off-Gas	No	No	No	No	Yes	No
067	Emergency Equipment Cooling Water	Yes	Yes	Yes	Yes	Yes	No
068	Recirculation	No	Yes	No	No	Yes	Yes
069	Reactor Water Clean Up	Yes	Yes	Yes	Yes	Yes	Yes
070	Reactor Building Closed Cooling Water	No	Yes	Yes	Yes	Yes	Yes
071	Reactor Core Isolation Cooling	Yes	Yes	Yes	Yes	Yes	Yes
072	Auxiliary Decay Heat Removal	No	No	No	No	No	No
073	High Pressure Coolant Injection	Yes	Yes	Yes	Yes	Yes	Yes
074	Residual Heat Removal	Yes	Yes	Yes	Yes	Yes	Yes
075	Core Spray System	Yes	Yes	Yes	Yes	Yes	Yes
076	Containment Air Monitoring	No	No	Yes	Yes	Yes	Yes
077	Rad waste	No	Yes	Yes	Yes	Yes	Yes
078	Fuel Pool Cooling	No	No	No	Yes	Yes	No
079	Refuel Tools	No	No	No	No	No	No
080	Primary Containment/Temperature Monitoring	No	No	No	No	No	No
082	Emergency Diesel Generators	Yes	Yes	Yes	Yes	Yes	No
084	CAD	Yes	Yes	Yes	Yes	Yes	Yes
085	Control Rod Drive	Yes	Yes	Yes	Yes	Yes	Yes
086	D/G Starting Air	Yes	Yes	Yes	Yes	Yes	No
090	Radiation Monitoring	No	No	No	No	No	No
092	Neutron Monitoring	No	No	Yes	Yes	Yes	Yes
094	TIP System	No	No	No	No	No	Yes
096	Reactor Recirculation Flow Control	No	No	No	No	Yes	No
099	Reactor Protection System	Yes	Yes	Yes	Yes	Yes	No
111	Cranes (Reactor & Turbine)	No	No	No	No	Yes	No
202	4-kV Unit Boards	No	Yes	No	No	Yes	No
203	4-kV Common Boards	No	Yes	No	No	Yes	No
204	4-kV Unit Start Board & Bus	No	Yes	No	No	Yes	No
205	4-kV Cooling Tower Switch Gear	No	No	No	No	No	No
206	4-kV Bio-Thermal Board	No	No	No	No	No	No
210	4-kV Bus Tie Board	Yes	Yes	No	No	Yes	No
211	4-kV Shutdown Board and Buses	Yes	Yes	Yes	Yes	Yes	Yes
215	480-V Common Board	Yes	Yes	No	No	Yes	No
219	480-V Diesel Aux Board	Yes	Yes	No	Yes	Yes	No
225	480-V Unit Boards	No	Yes	No	No	Yes	No
231	480-V Shutdown Boards	Yes	Yes	Yes	Yes	Yes	Yes
232	480-V Cooling Tower Boards	No	No	No	No	No	No
233	480-V Biothermal Boards	No	No	No	No	No	No
236	Main Transformers	Yes	Yes	No	No	Yes	No
237	480-V Service Building Main Board	No	No	No	No	No	No
238	480-V Transformer Yard Distribution	No	Yes	No	No	Yes	No
239	480-V Lighting Boards	No	No	No	No	No	No
240	480-V Water Supply Board	No	Yes	No	No	Yes	No

Table 4.1-1: List of Systems Considered in the SEL Development

System Number	System	Mitigation Potential	In IE PRA	In IPEEE	In 2.3 SWEL	In SEL	Containment Isolation
241	161-kV Switchyard	No	Yes	No	No	No	No
242	Main Generator	No	Yes	No	No	Yes	No
243	Unit Station Service Transformer	No	Yes	No	No	Yes	No
244	Communications	No	No	No	No	No	No
245	Common Station Service Transformer	No	Yes	No	No	Yes	No
246	Cooling Tower Transformer	No	No	No	No	No	No
248	250V DC System	Yes	Yes	Yes	Yes	Yes	Yes
249	Plant Preferred 120V AC	Yes	Yes	No	No	No	No
250	Plant Non-Preferred 120V AC	No	Yes	No	No	No	No
251	48-V DC Power System	No	No	No	No	No	No
252	Unit Preferred 120V AC	Yes	Yes	No	No	Yes	Yes
253	120-V AC Inst & Ctrl PWR	Yes	Yes	Yes	Yes	Yes	No
254	Diesel 125-V DC Sys	Yes	Yes	Yes	Yes	Yes	Yes
255	Data Logger	No	No	No	No	No	No
256	Inverters	Yes	Yes	Yes	Yes	Yes	No
258	Operation Recorder	No	No	No	No	No	No
259	480-V Load Shedding Logic	No	No	No	No	No	No
260	Security	No	No	No	No	No	No
261	Process Computer	No	No	No	No	No	No
262	Generator Bus Duct Cooling System	No	No	No	No	No	No
265	480-V Reactor Building Ventilation Boards	No	No	Yes	Yes	Yes	No
266	480-V Control Bay Vent Board	No	Yes	No	No	Yes	No
268	480-V Reactor MOV Boards	Yes	Yes	No	Yes	Yes	No
269	480-V Turbine Building MOV Board	No	Yes	No	No	Yes	No
270	480-V Condensate Demineralizer	No	No	No	No	No	No
271	480-V Aux Boiler Boards	No	No	No	No	No	No
272	480-V Water & Oil Storage	No	No	No	No	No	No
273	480-V Radwaste Boards	No	No	No	No	No	No
274	480-V Service Building Vent	No	No	No	No	No	No
275	480-V Office Building Vent Board	No	No	No	No	No	No
276	480-V Power Cabinets	No	No	No	No	No	No
277	Gatehouse Panel Board	No	No	No	No	No	No
278	Dist. Cabinets	No	Yes	No	No	Yes	No
280	Battery Boards	Yes	Yes	Yes	Yes	Yes	No
281	250-V Reactor MOV Boards	Yes	Yes	Yes	Yes	Yes	No
282	250-V DC Distribution Boards	Yes	Yes	No	No	Yes	No
283	24-V DC Power System	Yes	Yes	Yes	Yes	Yes	No
284	480-V Power Outlets	No	No	No	No	No	No
285	Computer Universal Power Supply	No	No	No	No	Yes	No
360	FLEX	Yes	No	No	No	Yes	No
Various	Miscellaneous Panels	Yes	Yes	Yes	Yes	Yes	No
Various	Miscellaneous Structures	Yes	No	No	No	Yes	Yes
Various	Additional potential SIFF fire & flooding sources	No	Yes	Yes	Yes	Yes	Yes

4.1.2 Relay and Breaker Evaluation

During a seismic event, vibratory ground motion can cause relays and breakers to chatter. The chattering of relays potentially can result in spurious signals to equipment. The chattering of breakers potentially can result in equipment either losing power or starting when it is not desired. Relay/breaker chatter can be acceptable (does not impact the associated equipment), self-correcting, or recovered by operator action. An extensive relay/breaker chatter evaluation was performed for all three BFN units [20] in accordance with SPID [2], Section 6.4.2, and PRA Standard, Section 5-2.2. The evaluation resulted in many relay/breaker chatter scenarios screened out from further evaluation based on no impact to component function. The 407 relays that were not screened out are listed in Table 4.1-2.

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-1-FCV-071-0002	BFN-1-FCV-071-0002, RCIC STEAM LINE INBD ISOLATION VLV	13A-K33	BFN-1-RLY-071-13A-K33	12HFA51A1F	SEIS_14-1R1-1
BFN-1-FCV-071-0002	BFN-1-FCV-071-0002, RCIC STEAM LINE INBD ISOLATION VLV	13A-K32	BFN-1-RLY-071-13A-K32	12HFA151A1F	SEIS_14-1R1-2
BFN-1-FCV-071-0002	BFN-1-FCV-071-0002, RCIC STEAM LINE INBD ISOLATION VLV	13A-K12	BFN-1-RLY-071-13A-K12	12HFA51A41F	SEIS_14-1R2-2
BFN-1-FCV-071-0002	BFN-1-FCV-071-0002, RCIC STEAM LINE INBD ISOLATION VLV	13A-K13	BFN-1-RLY-071-13A-K13	12HGA11A51F	SEIS_14-1R3-2
BFN-1-FCV-071-0002	BFN-1-FCV-071-0002, RCIC STEAM LINE INBD ISOLATION VLV	13A-K38	BFN-1-RLY-071-13A-K38	12HGA11A51F	SEIS_14-1R3-2
BFN-1-FCV-071-0002	BFN-1-FCV-071-0002, RCIC STEAM LINE INBD ISOLATION VLV	13A-K15	BFN-1-RLY-071-13A-K15	12HFA51A41F	SEIS_14-1R3-2
BFN-2-FCV-071-0002	BFN-2-FCV-071-0002, RCIC STEAM LINE INBD ISOLATION VLV	13A-K33	BFN-2-RLY-071-13A-K33	12HFA51A1F	SEIS_14-1R1-1
BFN-2-FCV-071-0002	BFN-2-FCV-071-0002, RCIC STEAM LINE INBD ISOLATION VLV	13A-K32	BFN-2-RLY-071-13A-K32	12HFA51A1F	SEIS_14-1R1-2
BFN-2-FCV-071-0002	BFN-2-FCV-071-0002, RCIC STEAM LINE INBD ISOLATION VLV	13A-K12	BFN-2-RLY-071-13A-K12	12HFA51A41F	SEIS_14-1R2-2

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-2-FCV-071-0002	BFN-2-FCV-071-0002, RCIC STEAM LINE INBD ISOLATION VLV	13A-K13	BFN-2-RLY-071-13A-K13	12HGA11A51F	SEIS_14-1R3-2
BFN-2-FCV-071-0002	BFN-2-FCV-071-0002, RCIC STEAM LINE INBD ISOLATION VLV	13A-K38	BFN-2-RLY-071-13A-K38	12HGA11A51F	SEIS_14-1R3-2
BFN-2-FCV-071-0002	BFN-2-FCV-071-0002, RCIC STEAM LINE INBD ISOLATION VLV	13A-K15	BFN-2-RLY-071-13A-K15	12HFA51A41F	SEIS_15-1R1
BFN-3-FCV-071-0002	BFN-3-FCV-071-0002, RCIC STEAM LINE INBD ISOLATION VLV	13A-K33	BFN-3-RLY-071-13A-K33	12HFA51A1F	SEIS_14-1R1-1
BFN-3-FCV-071-0002	BFN-3-FCV-071-0002, RCIC STEAM LINE INBD ISOLATION VLV	13A-K32	BFN-3-RLY-071-13A-K32	12HFA151A1F	SEIS_14-1R1-2
BFN-3-FCV-071-0002	BFN-3-FCV-071-0002, RCIC STEAM LINE INBD ISOLATION VLV	13A-K12	BFN-3-RLY-071-13A-K12	12HFA51A41F	SEIS_14-1R2-2
BFN-3-FCV-071-0002	BFN-3-FCV-071-0002, RCIC STEAM LINE INBD ISOLATION VLV	13A-K13	BFN-3-RLY-071-13A-K13	12HGA11A51F	SEIS_14-1R3-2
BFN-3-FCV-071-0002	BFN-3-FCV-071-0002, RCIC STEAM LINE INBD ISOLATION VLV	13A-K38	BFN-3-RLY-071-13A-K38	12HGA11A51F	SEIS_14-1R3-2
BFN-3-FCV-071-0002	BFN-3-FCV-071-0002, RCIC STEAM LINE INBD ISOLATION VLV	13A-K15	BFN-3-RLY-071-13A-K15	12HFA51A41F	SEIS_15-1R1

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-1-FCV-071-0003	BFN-1-FCV-071-0003, RCIC STM LINE OUTBD ISOL FLOW CONT VLV	13A-K16	BFN-1-RLY-071-13A-K16	12HFA51A41F	SEIS_14-1R2-2
BFN-1-FCV-071-0003	BFN-1-FCV-071-0003, RCIC STM LINE OUTBD ISOL FLOW CONT VLV	13A-K32	BFN-1-RLY-071-13A-K32	12HFA151A1F	SEIS_14-1R1-2
BFN-1-FCV-071-0003	BFN-1-FCV-071-0003, RCIC STM LINE OUTBD ISOL FLOW CONT VLV	13A-K12	BFN-1-RLY-071-13A-K12	12HFA51A41F	SEIS_14-1R2-2
BFN-1-FCV-071-0003	BFN-1-FCV-071-0003, RCIC STM LINE OUTBD ISOL FLOW CONT VLV	13A-K13	BFN-1-RLY-071-13A-K13	12HGA11A51F	SEIS_14-1R3-2
BFN-1-FCV-071-0003	BFN-1-FCV-071-0003, RCIC STM LINE OUTBD ISOL FLOW CONT VLV	13A-K38	BFN-1-RLY-071-13A-K38	12HGA11A51F	SEIS_14-1R3-2
BFN-1-FCV-071-0003	BFN-1-FCV-071-0003, RCIC STM LINE OUTBD ISOL FLOW CONT VLV	13A-K15	BFN-1-RLY-071-13A-K15	12HFA51A41F	SEIS_14-1R3-2
BFN-1-FCV-071-0003	BFN-1-FCV-071-0003, RCIC STM LINE OUTBD ISOL FLOW CONT VLV	13A-K33	BFN-1-RLY-071-13A-K33	12HFA51A1F	SEIS_14-1R1-1
BFN-2-FCV-071-0003	BFN-2-FCV-071-0003, RCIC STEAM LINE OUTBD ISOLATION VLV	13A-K16	BFN-2-RLY-071-13A-K16	12HFA51A41F	SEIS_14-1R2-2
BFN-2-FCV-071-0003	BFN-2-FCV-071-0003, RCIC STEAM LINE OUTBD ISOLATION VLV	13A-K32	BFN-2-RLY-071-13A-K32	12HFA151A1F	SEIS_14-1R1-2

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-2-FCV-071-0003	BFN-2-FCV-071-0003, RCIC STEAM LINE OUTBD ISOLATION VLV	13A-K12	BFN-2-RLY-071-13A-K12	12HFA51A41F	SEIS_14-1R2-2
BFN-2-FCV-071-0003	BFN-2-FCV-071-0003, RCIC STEAM LINE OUTBD ISOLATION VLV	13A-K13	BFN-2-RLY-071-13A-K13	12HGA11A51F	SEIS_14-1R3-2
BFN-2-FCV-071-0003	BFN-2-FCV-071-0003, RCIC STEAM LINE OUTBD ISOLATION VLV	13A-K38	BFN-2-RLY-071-13A-K38	12HGA11A51F	SEIS_14-1R3-2
BFN-2-FCV-071-0003	BFN-2-FCV-071-0003, RCIC STEAM LINE OUTBD ISOLATION VLV	13A-K15	BFN-2-RLY-071-13A-K15	12HFA51A41F	SEIS_15-1R1
BFN-2-FCV-071-0003	BFN-2-FCV-071-0003, RCIC STEAM LINE OUTBD ISOLATION VLV	13A-K33	BFN-2-RLY-071-13A-K33	12HFA51A1F	SEIS_14-1R1-1
BFN-3-FCV-071-0003	BFN-3-FCV-071-0003, RCIC STEAM LINE OUTBD ISOLATION VLV	13A-K16	BFN-3-RLY-071-13A-K16	12HFA51A41F	SEIS_14-1R2-2
BFN-3-FCV-071-0003	BFN-3-FCV-071-0003, RCIC STEAM LINE OUTBD ISOLATION VLV	13A-K32	BFN-3-RLY-071-13A-K32	12HFA151A1F	SEIS_14-1R1-2
BFN-3-FCV-071-0003	BFN-3-FCV-071-0003, RCIC STEAM LINE OUTBD ISOLATION VLV	13A-K12	BFN-3-RLY-071-13A-K12	12HFA51A41F	SEIS_14-1R2-2
BFN-3-FCV-071-0003	BFN-3-FCV-071-0003, RCIC STEAM LINE OUTBD ISOLATION VLV	13A-K13	BFN-3-RLY-071-13A-K13	12HGA11A51F	SEIS_14-1R3-2

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-FCV-071-0003	BFN-3-FCV-071-0003, RCIC STEAM LINE OUTBD ISOLATION VLV	13A-K38	BFN-3-RLY-071-13A-K38	12HGA11A51F	SEIS_14-1R3-2
BFN-3-FCV-071-0003	BFN-3-FCV-071-0003, RCIC STEAM LINE OUTBD ISOLATION VLV	13A-K15	BFN-3-RLY-071-13A-K15	12HFA51A41F	SEIS_15-1R1
BFN-3-FCV-071-0003	BFN-3-FCV-071-0003, RCIC STEAM LINE OUTBD ISOLATION VLV	13A-K33	BFN-3-RLY-071-13A-K33	12HFA51A1F	SEIS_14-1R1-1
BFN-1-XX-071-0009	BFN-1-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID (SPECIAL POWER SUPPLY)	13A-K8	BFN-1-RLY-071-13A-K8	12HFA51A41F	SEIS_15-1R1
BFN-1-XX-071-0009	BFN-1-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID (SPECIAL POWER SUPPLY)	13A-K15	BFN-1-RLY-071-13A-K15	12HFA51A41F	SEIS_14-1R3-2
BFN-1-XX-071-0009	BFN-1-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID (SPECIAL POWER SUPPLY)	13A-K39	BFN-1-RLY-071-13A-K39	12HFA51A41F	SEIS_15-1R1
BFN-1-XX-071-0009	BFN-1-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID (SPECIAL POWER SUPPLY)	13A-K6	BFN-1-RLY-071-13A-K6	12HGA11A51F	SEIS_15-1R2
BFN-1-XX-071-0009	BFN-1-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID (SPECIAL POWER SUPPLY)	13A-K7	BFN-1-RLY-071-13A-K7	12HGA11A51F	SEIS_15-1R2

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-1-XX-071-0009	BFN-1-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID (SPECIAL POWER SUPPLY)	1-71-9C	BFN-1-RLY-071-009C	Agastat GPFNR	SEIS_15-1R2
BFN-1-XX-071-0009	BFN-1-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID (SPECIAL POWER SUPPLY)	PS-71-13A	BFN-1-PS-071-0013A	Mercoid DS-7043-804	SEIS_15-1R1-1
BFN-1-XX-071-0009	BFN-1-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID (SPECIAL POWER SUPPLY)	PS-71-13B	BFN-1-PS-071-0013B	Mercoid DS-7043-804	SEIS_15-1R1-1
BFN-1-XX-071-0009	BFN-1-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID (SPECIAL POWER SUPPLY)	PS-71-21A	BFN-1-PS-071-0021A	Static-O-ring 54N4-GG118-M4-C1A-TT	SEIS_15-1R2
BFN-2-XX-071-0009	BFN-2-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID	13A-K8	BFN-2-RLY-071-13A-K8	12HFA51A41F	SEIS_15-1R1
BFN-2-XX-071-0009	BFN-2-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID	13A-K15	BFN-2-RLY-071-13A-K15	12HFA51A41F	SEIS_15-1R1
BFN-2-XX-071-0009	BFN-2-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID	13A-K39	BFN-2-RLY-071-13A-K39	12HFA51A41F	SEIS_15-1R1
BFN-2-XX-071-0009	BFN-2-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID	13A-K6	BFN-2-RLY-071-13A-K6	12HGA11A51F	SEIS_15-1R2
BFN-2-XX-071-0009	BFN-2-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID	13A-K7	BFN-2-RLY-071-13A-K7	12HGA11A51F	SEIS_15-1R2

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-2-XX-071-0009	BFN-2-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID	2-71-9C	2-71-9C*	Square D	SEIS_15-1R2
BFN-2-XX-071-0009	BFN-2-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID	PS-71-13A	BFN-2-PS-071-0013A	Mercoid DS-7043-804	SEIS_15-1R1-1
BFN-2-XX-071-0009	BFN-2-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID	PS-71-13B	BFN-2-PS-071-0013B	Mercoid DS-7043-804	SEIS_15-1R1-1
BFN-2-XX-071-0009	BFN-2-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID	PS-71-21A	BFN-2-PS-071-0021A	Static-O-ring 54N4-GG118-M4-C1A-TT	SEIS_15-1R2
BFN-3-XX-071-0009	BFN-3-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID	13A-K8	BFN-3-RLY-071-13A-K8	12HFA51A41F	SEIS_13-2R1
BFN-3-XX-071-0009	BFN-3-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID	13A-K15	BFN-3-RLY-071-13A-K15	12HFA51A41F	SEIS_15-1R1
BFN-3-XX-071-0009	BFN-3-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID	13A-K39	BFN-3-RLY-071-13A-K39	12HFA51A41F	SEIS_13-2R1
BFN-3-XX-071-0009	BFN-3-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID	13A-K6	BFN-3-RLY-071-13A-K6	12HGA11A51F	SEIS_13-2R2
BFN-3-XX-071-0009	BFN-3-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID	13A-K7	BFN-3-RLY-071-13A-K7	12HGA11A51F	SEIS_13-2R2

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-XX-071-0009	BFN-3-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID	3-71-9C	3-71-9C*	Square D	SEIS_13-2R2
BFN-3-XX-071-0009	BFN-3-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID	PS-71-13A	BFN-3-PS-071-0013A	Mercoid DS-7043-804	SEIS_13-2R1-1
BFN-3-XX-071-0009	BFN-3-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID	PS-71-13B	BFN-3-PS-071-0013B	Mercoid DS-7043-804	SEIS_13-2R1-1
BFN-3-XX-071-0009	BFN-3-XX-071-0009, RCIC TURB STOP VLV TRIP SOLENOID	PS-71-21A	BFN-3-PS-071-0021A	Static-O-ring 54N4-GG118-M4-C1A-TT	SEIS_13-2R2
BFN-0-GEN-082-000A	BFN-0-GEN-082-000A, DIESEL GENERATOR A	86GA	BFN-0-86-082-2547A/GA	12HEA61C238	SEIS_1C-1R5
BFN-0-GEN-082-000A	BFN-0-GEN-082-000A, DIESEL GENERATOR A	41	BFN-0-41-082-000A/1	Exciter Breaker Shunt trip relay (coil is internal to the bkr)	SEIS_9-1R4
BFN-0-GEN-082-000A	BFN-0-GEN-082-000A, DIESEL GENERATOR A	87GA phase A	BFN-0-87G-082-2547A/AA	GE 12CFD12B1A	SEIS_9-1R3
BFN-0-GEN-082-000A	BFN-0-GEN-082-000A, DIESEL GENERATOR A	87GA phase B	BFN-0-87G-082-2547A/AB	GE 12CFD12B1A	SEIS_9-1R3
BFN-0-GEN-082-000A	BFN-0-GEN-082-000A, DIESEL GENERATOR A	87GA phase C	BFN-0-87G-082-2547A/AC	GE 12CFD12B1A	SEIS_9-1R3

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-GEN-082-000A	BFN-0-GEN-082-000A, DIESEL GENERATOR A	A/OTX	BFN-0-RLY-082-A/OTX	12HFA51A42F	SEIS_1C-1R2
BFN-0-GEN-082-000A	BFN-0-GEN-082-000A, DIESEL GENERATOR A	OTR	BFN-0-RLY-082-A/OTR	Square D DO/7001	SEIS_1C-1R1
BFN-0-GEN-082-000A	BFN-0-GEN-082-000A, DIESEL GENERATOR A	CRA	BFN-0-RLY-082-CRA	12HFA51A42F	SEIS_9-1R2
BFN-0-GEN-082-000A	BFN-0-GEN-082-000A, DIESEL GENERATOR A	GRRA	BFN-0-RLY-082-GRRA	12HFA51A42F	SEIS_9-1R2
BFN-0-GEN-082-000A	BFN-0-GEN-082-000A, DIESEL GENERATOR A	GRLA	BFN-0-RLY-082-GRLA	12HFA51A42F	SEIS_9-1R2
BFN-0-GEN-082-000A	BFN-0-GEN-082-000A, DIESEL GENERATOR A	VLRA	BFN-0-RLY-082-VLRA	12HFA51A42F	SEIS_1C-1R2
BFN-0-GEN-082-000A	BFN-0-GEN-082-000A, DIESEL GENERATOR A	VRRA	BFN-0-RLY-082-VRRA	12HFA51A42F	SEIS_1C-1R2
BFN-0-GEN-082-000B	BFN-0-GEN-082-000B, DIESEL GENERATOR B	86GB	BFN-0-86-082-2547B/GB	12HEA61C238	SEIS_1C-3R5
BFN-0-GEN-082-000B	BFN-0-GEN-082-000B, DIESEL GENERATOR B	41	BFN-0-41-082-000B/1	Exciter Breaker Shunt trip relay (coil is internal to the bkr)	SEIS_9-1R4

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-GEN-082-000B	BFN-0-GEN-082-000B, DIESEL GENERATOR B	87GB phase A	BFN-0-87G-082-2547B/BA	GE 12CFD12B1A	SEIS_9-1R3
BFN-0-GEN-082-000B	BFN-0-GEN-082-000B, DIESEL GENERATOR B	87GB phase B	BFN-0-87G-082-2547B/BB	GE 12CFD12B1A	SEIS_9-1R3
BFN-0-GEN-082-000B	BFN-0-GEN-082-000B, DIESEL GENERATOR B	87GB phase C	BFN-0-87G-082-2547B/BC	GE 12CFD12B1A	SEIS_9-1R3
BFN-0-GEN-082-000B	BFN-0-GEN-082-000B, DIESEL GENERATOR B	B/OTX	BFN-0-RLY-082-B/OTX	12HFA51A42F	SEIS_1C-3R2
BFN-0-GEN-082-000B	BFN-0-GEN-082-000B, DIESEL GENERATOR B	OTR	BFN-0-RLY-082-B/OTR	Square D Class 8501 Type XUD0-1200	SEIS_1C-3R1
BFN-0-GEN-082-000B	BFN-0-GEN-082-000B, DIESEL GENERATOR B	CRB	BFN-0-RLY-082-CRB	12HFA51A42F	SEIS_9-1R2
BFN-0-GEN-082-000B	BFN-0-GEN-082-000B, DIESEL GENERATOR B	GRRB	BFN-0-RLY-082-GRRB	12HFA51A42F	SEIS_9-1R2
BFN-0-GEN-082-000B	BFN-0-GEN-082-000B, DIESEL GENERATOR B	GRLB	BFN-0-RLY-082-GRLB	12HFA51A42F	SEIS_9-1R2
BFN-0-GEN-082-000B	BFN-0-GEN-082-000B, DIESEL GENERATOR B	VLRB	BFN-0-RLY-082-VLRB	12HFA51A42F	SEIS_1C-3R2

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-GEN-082-000B	BFN-0-GEN-082-000B, DIESEL GENERATOR B	VRRB	BFN-0-RLY-082-VRRB	12HFA51A42F	SEIS_1C-3R2
BFN-0-GEN-082-000C	BFN-0-GEN-082-000C, DIESEL GENERATOR C	86GC	BFN-0-86-082-2547C/GC	12HEA61C238	SEIS_1C-6R5
BFN-0-GEN-082-000C	BFN-0-GEN-082-000C, DIESEL GENERATOR C	41	BFN-0-41-082-000C/1	Exciter Breaker Shunt trip relay (coil is internal to the bkr)	SEIS_9-1R4
BFN-0-GEN-082-000C	BFN-0-GEN-082-000C, DIESEL GENERATOR C	87GC phase A	BFN-0-87G-082-2547C/CA	GE 12CFD12B1A	SEIS_9-1R3
BFN-0-GEN-082-000C	BFN-0-GEN-082-000C, DIESEL GENERATOR C	87GC phase B	BFN-0-87G-082-2547C/CB	GE 12CFD12B1A	SEIS_9-1R3
BFN-0-GEN-082-000C	BFN-0-GEN-082-000C, DIESEL GENERATOR C	87GC phase C	BFN-0-87G-082-2547C/CC	GE 12CFD12B1A	SEIS_9-1R3
BFN-0-GEN-082-000C	BFN-0-GEN-082-000C, DIESEL GENERATOR C	C/OTX	BFN-0-RLY-082-C/OTX	12HFA51A42F	SEIS_1C-6R2
BFN-0-GEN-082-000C	BFN-0-GEN-082-000C, DIESEL GENERATOR C	OTR	BFN-0-RLY-082-C/OTR	Square D Class 8501 Type XUD0-1200	SEIS_1C-6R1
BFN-0-GEN-082-000C	BFN-0-GEN-082-000C, DIESEL GENERATOR C	CRC	BFN-0-RLY-082-CRC	12HFA51A42F	SEIS_9-1R2

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-GEN-082-000C	BFN-0-GEN-082-000C, DIESEL GENERATOR C	GRRC	BFN-0-RLY-082-GRRC	12HFA51A42F	SEIS_9-1R2
BFN-0-GEN-082-000C	BFN-0-GEN-082-000C, DIESEL GENERATOR C	GRLC	BFN-0-RLY-082-GRLC	12HFA51A42F	SEIS_9-1R2
BFN-0-GEN-082-000C	BFN-0-GEN-082-000C, DIESEL GENERATOR C	VLRC	BFN-0-RLY-082-VLRC	12HFA51A42F	SEIS_1C-6R2
BFN-0-GEN-082-000C	BFN-0-GEN-082-000C, DIESEL GENERATOR C	VRRC	BFN-0-RLY-082-VRRC	12HFA51A42F	SEIS_1C-6R2
BFN-0-GEN-082-000D	BFN-0-GEN-082-000D, DIESEL GENERATOR D	86GD	BFN-0-86-082-2547D/GD	12HEA61C238X2	SEIS_1C-3R5
BFN-0-GEN-082-000D	BFN-0-GEN-082-000D, DIESEL GENERATOR D	41	BFN-0-41-082-000D/1	Exciter Breaker Shunt trip relay (coil is internal to the bkr)	SEIS_9-1R4
BFN-0-GEN-082-000D	BFN-0-GEN-082-000D, DIESEL GENERATOR D	87GD phase A	BFN-0-87G-082-2547D/DA	GE 12CFD12B1A	SEIS_9-1R3
BFN-0-GEN-082-000D	BFN-0-GEN-082-000D, DIESEL GENERATOR D	87GD phase B	BFN-0-87G-082-2547D/DB	GE 12CFD12B1A	SEIS_9-1R3
BFN-0-GEN-082-000D	BFN-0-GEN-082-000D, DIESEL GENERATOR D	87GD phase C	BFN-0-87G-082-2547D/DC	GE 12CFD12B1A	SEIS_9-1R3

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-GEN-082-000D	BFN-0-GEN-082-000D, DIESEL GENERATOR D	D/OTX	BFN-0-RLY-082-D/OTX	12HFA51A42F	SEIS_1C-3R2
BFN-0-GEN-082-000D	BFN-0-GEN-082-000D, DIESEL GENERATOR D	OTR	BFN-0-RLY-082-D/OTR	Square D Class 8501 Type XUD0-1200	SEIS_1C-3R1
BFN-0-GEN-082-000D	BFN-0-GEN-082-000D, DIESEL GENERATOR D	CRD	BFN-0-RLY-082-CRD	12HFA51A42F	SEIS_9-1R2
BFN-0-GEN-082-000D	BFN-0-GEN-082-000D, DIESEL GENERATOR D	GRRD	BFN-0-RLY-082-GRRD	12HFA51A42F	SEIS_9-1R2
BFN-0-GEN-082-000D	BFN-0-GEN-082-000D, DIESEL GENERATOR D	GRLD	BFN-0-RLY-082-GRLD	12HFA51A42F	SEIS_9-1R2
BFN-0-GEN-082-000D	BFN-0-GEN-082-000D, DIESEL GENERATOR D	VLRD	BFN-0-RLY-082-VLRD	12HFA51A42F	SEIS_1C-3R2
BFN-0-GEN-082-000D	BFN-0-GEN-082-000D, DIESEL GENERATOR D	VRRD	BFN-0-RLY-082-VRRD	12HFA51A42F	SEIS_1C-3R2
BFN-3-GEN-082-0003A	BFN-3-GEN-082-0003A, DIESEL GENERATOR 3A	86G3A	BFN-3-86-082-2547A/GA	12HEA61C238X2	SEIS_1C-4R6
BFN-3-GEN-082-0003A	BFN-3-GEN-082-0003A, DIESEL GENERATOR 3A	41	BFN-3-410-82-000A/1	Exciter Breaker Shunt trip relay (coil is internal to the bkr)	SEIS_9-1R4

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-GEN-082-0003A	BFN-3-GEN-082-0003A, DIESEL GENERATOR 3A	87G3A phase A	BFN-3-87G-082-2547A/AA	GE 12CFD12B1A	SEIS_9-1R3
BFN-3-GEN-082-0003A	BFN-3-GEN-082-0003A, DIESEL GENERATOR 3A	87G3A phase B	BFN-3-87G-082-2547A/AB	GE 12CFD12B1A	SEIS_9-1R3
BFN-3-GEN-082-0003A	BFN-3-GEN-082-0003A, DIESEL GENERATOR 3A	87G3A phase C	BFN-3-87G-082-2547A/AC	GE 12CFD12B1A	SEIS_9-1R3
BFN-3-GEN-082-0003A	BFN-3-GEN-082-0003A, DIESEL GENERATOR 3A	A/OTX	BFN-3-RLY-082-A/OTX	12HFA51A42F	SEIS_1C-4R2
BFN-3-GEN-082-0003A	BFN-3-GEN-082-0003A, DIESEL GENERATOR 3A	OTR	BFN-3-RLY-082-A/OTR	Square D Class 8501 Type XUD0-1200	SEIS_1C-4R1
BFN-3-GEN-082-0003A	BFN-3-GEN-082-0003A, DIESEL GENERATOR 3A	CRA	BFN-3-RLY0-82-CRA	12HFA51A42F	SEIS_9-1R2U3
BFN-3-GEN-082-0003A	BFN-3-GEN-082-0003A, DIESEL GENERATOR 3A	GRRA	BFN-3-RLY-082-GRRA	12HFA51A42F	SEIS_9-1R2U3
BFN-3-GEN-082-0003A	BFN-3-GEN-082-0003A, DIESEL GENERATOR 3A	GRLA	BFN-3-RLY0-82-GRLA	12HFA51A42F	SEIS_9-1R2U3
BFN-3-GEN-082-0003A	BFN-3-GEN-082-0003A, DIESEL GENERATOR 3A	VLRA	BFN-3-RLY0-82-VLRA	12HFA51A42H	SEIS_9-1R2U3

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-GEN-082-0003A	BFN-3-GEN-082-0003A, DIESEL GENERATOR 3A	VRRA	BFN-3-RLY-082-VRRA	12HFA51A42H	SEIS_9-1R2U3
BFN-3-GEN-082-0003B	BFN-3-GEN-082-0003B, DIESEL GENERATOR 3B	86G3B	BFN-3-86-082-2547B/GB	12HEA61C238X2	SEIS_1C-5R6
BFN-3-GEN-082-0003B	BFN-3-GEN-082-0003B, DIESEL GENERATOR 3B	41	BFN-3-41-082-000B/1	Exciter Breaker Shunt trip relay (coil is internal to the bkr)	SEIS_9-1R4
BFN-3-GEN-082-0003B	BFN-3-GEN-082-0003B, DIESEL GENERATOR 3B	87G3B phase A	BFN-3-87G-082-2547B/BA	GE 12CFD12B1A	SEIS_9-1R3
BFN-3-GEN-082-0003B	BFN-3-GEN-082-0003B, DIESEL GENERATOR 3B	87G3B phase B	BFN-3-87G-082-2547B/BB	GE 12CFD12B1A	SEIS_9-1R3
BFN-3-GEN-082-0003B	BFN-3-GEN-082-0003B, DIESEL GENERATOR 3B	87G3B phase C	BFN-3-87G-082-2547B/BC	GE 12CFD12B1A	SEIS_9-1R3
BFN-3-GEN-082-0003B	BFN-3-GEN-082-0003B, DIESEL GENERATOR 3B	B/OTX	BFN-3-RLY-082-B/OTX	12HFA51A42F	SEIS_1C-5R3
BFN-3-GEN-082-0003B	BFN-3-GEN-082-0003B, DIESEL GENERATOR 3B	OTR	BFN-3-RLY-082-B/0TR	Square D Class 8501 Type XUD0-1200	SEIS_1C-5R1
BFN-3-GEN-082-0003B	BFN-3-GEN-082-0003B, DIESEL GENERATOR 3B	CRB	BFN-3-RLY-082-CRB	12HFA51A42F	SEIS_9-1R2U3-1

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-GEN-082-0003B	BFN-3-GEN-082-0003B, DIESEL GENERATOR 3B	GRRB	BFN-3-RLY-082-GRRB	12HFA51A42F	SEIS_9-1R2U3-1
BFN-3-GEN-082-0003B	BFN-3-GEN-082-0003B, DIESEL GENERATOR 3B	GRLB	BFN-3-RLY-082-GRLB	12HFA51A42F	SEIS_9-1R2U3-1
BFN-3-GEN-082-0003B	BFN-3-GEN-082-0003B, DIESEL GENERATOR 3B	VLRB	BFN-3-RLY-082-VLRB	12HFA51A42H	SEIS_9-1R2U3-1
BFN-3-GEN-082-0003B	BFN-3-GEN-082-0003B, DIESEL GENERATOR 3B	VRRB	BFN-3-RLY-082-VRRB	12HFA51A42H	SEIS_9-1R2U3-1
BFN-3-GEN-082-0003C	BFN-3-GEN-082-0003C, DIESEL GENERATOR 3C	86G3C	BFN-3-86-82-2547C/GC	12HEA61C238X2	SEIS_1C-4R6
BFN-3-GEN-082-0003C	BFN-3-GEN-082-0003C, DIESEL GENERATOR 3C	41	BFN-3-41-082-000C/1	Exciter Breaker Shunt trip relay (coil is internal to the bkr)	SEIS_9-1R4
BFN-3-GEN-082-0003C	BFN-3-GEN-082-0003C, DIESEL GENERATOR 3C	87G3C phase A	BFN-3-87G-082-2547C/CA	GE 12CFD12B1A	SEIS_9-1R3
BFN-3-GEN-082-0003C	BFN-3-GEN-082-0003C, DIESEL GENERATOR 3C	87G3C phase B	BFN-3-87G-082-2547C/CB	GE 12CFD12B1A	SEIS_9-1R3
BFN-3-GEN-082-0003C	BFN-3-GEN-082-0003C, DIESEL GENERATOR 3C	87G3C phase C	BFN-3-87G-082-2547C/CC	GE 12CFD12B1A	SEIS_9-1R3

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-GEN-082-0003C	BFN-3-GEN-082-0003C, DIESEL GENERATOR 3C	C/OTX	BFN-3-RLY-082-C/OTX	12HFA51A42F	SEIS_1C-4R2
BFN-3-GEN-082-0003C	BFN-3-GEN-082-0003C, DIESEL GENERATOR 3C	OTR	BFN-3-RLY-082-C/0TR	Square D Class 8501 Type XUD0-1200	SEIS_1C-4R1
BFN-3-GEN-082-0003C	BFN-3-GEN-082-0003C, DIESEL GENERATOR 3C	CRC	BFN-3-RLY-082-CRC	12HFA51A42F	SEIS_9-1R2U3
BFN-3-GEN-082-0003C	BFN-3-GEN-082-0003C, DIESEL GENERATOR 3C	GRRC	BFN-3-RLY-082-GRRC	12HFA51A42F	SEIS_9-1R2U3
BFN-3-GEN-082-0003C	BFN-3-GEN-082-0003C, DIESEL GENERATOR 3C	GRLC	BFN-3-RLY-082-GRLC	12HFA51A42F	SEIS_9-1R2U3
BFN-3-GEN-082-0003C	BFN-3-GEN-082-0003C, DIESEL GENERATOR 3C	VLRC	BFN-3-RLY-082-VLRC	12HFA51A42H	SEIS_9-1R2U3
BFN-3-GEN-082-0003C	BFN-3-GEN-082-0003C, DIESEL GENERATOR 3C	VRRC	BFN-3-RLY-082-VRRC	12HFA51A42H	SEIS_9-1R2U3
BFN-3-GEN-082-0003D	BFN-3-GEN-082-0003D, DIESEL GENERATOR 3D	86G3D	BFN-3-86-082-2547D/GD	12HEA61C238X2	SEIS_1C-5R6
BFN-3-GEN-082-0003D	BFN-3-GEN-082-0003D, DIESEL GENERATOR 3D	41	BFN-3-41-082-000D/1	Exciter Breaker Shunt trip relay (coil is internal to the bkr)	SEIS_9-1R4

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-GEN-082-0003D	BFN-3-GEN-082-0003D, DIESEL GENERATOR 3D	87G3D phase A	BFN-3-87G-082-2547D/DA	GE 12CFD22B1A	SEIS_9-1R3
BFN-3-GEN-082-0003D	BFN-3-GEN-082-0003D, DIESEL GENERATOR 3D	87G3D phase B	BFN-3-87G-082-2547D/DB	GE 12CFD22B1A	SEIS_9-1R3
BFN-3-GEN-082-0003D	BFN-3-GEN-082-0003D, DIESEL GENERATOR 3D	87G3D phase C	BFN-3-87G-082-2547D/DC	GE 12CFD22B1A	SEIS_9-1R3
BFN-3-GEN-082-0003D	BFN-3-GEN-082-0003D, DIESEL GENERATOR 3D	D/OTX	BFN-3-RLY-082-D/OTX	12HFA51A42H	SEIS_1C-5R3
BFN-3-GEN-082-0003D	BFN-3-GEN-082-0003D, DIESEL GENERATOR 3D	OTR	BFN-3-RLY-082-D/0TR	Square D Class 8501 Type XUD0-1200	SEIS_1C-5R1
BFN-3-GEN-082-0003D	BFN-3-GEN-082-0003D, DIESEL GENERATOR 3D	CRD	BFN-3-RLY-082-CRD	12HFA51A42H	SEIS_9-1R2U3-1
BFN-3-GEN-082-0003D	BFN-3-GEN-082-0003D, DIESEL GENERATOR 3D	GRRD	BFN-3-RLY-082-GRRD	12HFA51A42H	SEIS_9-1R2U3-1
BFN-3-GEN-082-0003D	BFN-3-GEN-082-0003D, DIESEL GENERATOR 3D	GRLD	BFN-3-RLY-082-GRLD	12HFA51A42H	SEIS_9-1R2U3-1
BFN-3-GEN-082-0003D	BFN-3-GEN-082-0003D, DIESEL GENERATOR 3D	VLRD	BFN-3-RLY-082-VLRD	12HFA51A42H	SEIS_9-1R2U3-1

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-GEN-082-0003D	BFN-3-GEN-082-0003D, DIESEL GENERATOR 3D	VRRD	BFN-3-RLY-082-VRRD	12HFA51A42H	SEIS_9-1R2U3-1
BFN-0-BKR-211-000A/022	BFN-0-BKR-211-000A/022, 4KV SD BD A/22, BKR 1818, TIE TO DIESEL GEN A	CAR	BFN-0-RLY-082-2547ACAR	12HFA51A42F	SEIS_1C-1R2
BFN-0-BKR-211-000A/022	BFN-0-BKR-211-000A/022, 4KV SD BD A/22, BKR 1818, TIE TO DIESEL GEN A	51X1-A	51X-82-2547A/1	Q12HGA111J2	Chatter Acceptable
BFN-0-BKR-211-000A/022	BFN-0-BKR-211-000A/022, 4KV SD BD A/22, BKR 1818, TIE TO DIESEL GEN A	51VX-A	51V-82-2547A/X	GE 12PJV11AM2A	Chatter Acceptable
BFN-0-BKR-211-000A/022	BFN-0-BKR-211-000A/022, 4KV SD BD A/22, BKR 1818, TIE TO DIESEL GEN A	51VZ-A	51V-82-2547A/Z	12HFA51A42F	Chatter Acceptable
BFN-0-BKR-211-000A/022	BFN-0-BKR-211-000A/022, 4KV SD BD A/22, BKR 1818, TIE TO DIESEL GEN A	OTX	BFN-0-RLY-082-A/OTX	12HFA51A42F	SEIS_1C-1R2
BFN-0-BKR-211-000A/022	BFN-0-BKR-211-000A/022, 4KV SD BD A/22, BKR 1818, TIE TO DIESEL GEN A	OTR	BFN-0-RLY-082-A/OTR	Square D DO/7001	SEIS_1C-1R1
BFN-0-BKR-211-000A/022	BFN-0-BKR-211-000A/022, 4KV SD BD A/22, BKR 1818, TIE TO DIESEL GEN A	VLRA	BFN-0-RLY-082-VLRA	12HFA51A42F	SEIS_1C-1R2
BFN-0-BKR-211-000A/022	BFN-0-BKR-211-000A/022, 4KV SD BD A/22, BKR 1818, TIE TO DIESEL GEN A	VRRA	BFN-0-RLY-082-VRRA	12HFA51A42F	SEIS_1C-1R2

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-BKR-211-000A/022	BFN-0-BKR-211-000A/022, 4KV SD BD A/22, BKR 1818, TIE TO DIESEL GEN A	A/ESTR	BFN-0-RLY-082-A/ESTR	Square D DO/7001	SEIS_1C-1R1
BFN-0-BKR-211-000A/022	BFN-0-BKR-211-000A/022, 4KV SD BD A/22, BKR 1818, TIE TO DIESEL GEN A	CASA-2	BFN-0-RLY-211-CASA-2	12HFA51A41H	SEIS_1C-1R8
BFN-0-BKR-211-000A/022	BFN-0-BKR-211-000A/022, 4KV SD BD A/22, BKR 1818, TIE TO DIESEL GEN A	86-GA-818	BFN-0-86-082-2547A/GA	12HEA61C238	SEIS_1C-1R5
BFN-0-BKR-211-000A/022	BFN-0-BKR-211-000A/022, 4KV SD BD A/22, BKR 1818, TIE TO DIESEL GEN A	86-1-614	BFN-0-86-211-000A/003	GE HEA-61B	SEIS_1C-1R4
BFN-0-BKR-211-000A/022	BFN-0-BKR-211-000A/022, 4KV SD BD A/22, BKR 1818, TIE TO DIESEL GEN A	51-614 Phase A	BFN-0-51-211-000A/03A	GE IAC-51A	SEIS_1C-1R3
BFN-0-BKR-211-000A/022	BFN-0-BKR-211-000A/022, 4KV SD BD A/22, BKR 1818, TIE TO DIESEL GEN A	51-614 Phase C	BFN-0-51-211-000A/03C	GE IAC-51A	SEIS_1C-1R3
BFN-0-BKR-211-000A/022	BFN-0-BKR-211-000A/022, 4KV SD BD A/22, BKR 1818, TIE TO DIESEL GEN A	86-2-716	BFN-0-86-211-000A/024	GE HEA-61B	SEIS_1C-1R4
BFN-0-BKR-211-000A/022	BFN-0-BKR-211-000A/022, 4KV SD BD A/22, BKR 1818, TIE TO DIESEL GEN A	51-716 Phase A	BFN-0-51-211-000A/24A	GE IAC-51A	SEIS_1C-1R3
BFN-0-BKR-211-000A/022	BFN-0-BKR-211-000A/022, 4KV SD BD A/22, BKR 1818, TIE TO DIESEL GEN A	51-716 Phase C	BFN-0-51-211-000A/24C	GE IAC-51A	SEIS_1C-1R3

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-BKR-211-000A/022	BFN-0-BKR-211-000A/022, 4KV SD BD A/22, BKR 1818, TIE TO DIESEL GEN A	86/SA	BFN-0-86-211-000A/023	GE HEA-61B	SEIS_1C-1R4
BFN-0-BKR-211-000A/022	BFN-0-BKR-211-000A/022, 4KV SD BD A/22, BKR 1818, TIE TO DIESEL GEN A	R1A	BFN-0-RLY-082-R1A	12HFA51A42F	SEIS_1C-1R2
BFN-0-BKR-211-000A/022	BFN-0-BKR-211-000A/022, 4KV SD BD A/22, BKR 1818, TIE TO DIESEL GEN A	Breaker	BFN-0-BKR-211-000A/022	GE Hitachi NE - 317A7502P005	Modeled as DG fails to start
BFN-0-BKR-211-000B/004	BFN-0-BKR-211-000B/004, 4KV SD BD B/4, BKR 1822, TIE TO DIESEL GEN B	CAR	BFN-0-RLY-082-2547BCAR	12HFA51A42F	SEIS_1C-3R2
BFN-0-BKR-211-000B/004	BFN-0-BKR-211-000B/004, 4KV SD BD B/4, BKR 1822, TIE TO DIESEL GEN B	51X1-B	51X-82-2547B/1	Q12HGA111J2	Chatter Acceptable
BFN-0-BKR-211-000B/004	BFN-0-BKR-211-000B/004, 4KV SD BD B/4, BKR 1822, TIE TO DIESEL GEN B	51VX-B	51V-82-2547B/X	GE12PJV11AM2A	Chatter Acceptable
BFN-0-BKR-211-000B/004	BFN-0-BKR-211-000B/004, 4KV SD BD B/4, BKR 1822, TIE TO DIESEL GEN B	51VZ-B	51V-82-2547B/Z	12HFA51A42F	Chatter Acceptable
BFN-0-BKR-211-000B/004	BFN-0-BKR-211-000B/004, 4KV SD BD B/4, BKR 1822, TIE TO DIESEL GEN B	OTX	BFN-0-RLY-082-B/OTX	12HFA51A42F	SEIS_1C-3R2
BFN-0-BKR-211-000B/004	BFN-0-BKR-211-000B/004, 4KV SD BD B/4, BKR 1822, TIE TO DIESEL GEN B	OTR	BFN-0-RLY-082-B/OTR	Square D Class 8501 Type XUD0-1200	SEIS_1C-3R1

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-BKR-211-000B/004	BFN-0-BKR-211-000B/004, 4KV SD BD B/4, BKR 1822, TIE TO DIESEL GEN B	VLRB	BFN-0-RLY-082-VLRB	12HFA51A42F	SEIS_1C-3R2
BFN-0-BKR-211-000B/004	BFN-0-BKR-211-000B/004, 4KV SD BD B/4, BKR 1822, TIE TO DIESEL GEN B	VRRB	BFN-0-RLY-082-VRRB	12HFA51A42F	SEIS_1C-3R2
BFN-0-BKR-211-000B/004	BFN-0-BKR-211-000B/004, 4KV SD BD B/4, BKR 1822, TIE TO DIESEL GEN B	B/ESTR	BFN-0-RLY-082-B/ESTR	Square D DO/7001	SEIS_1C-3R1
BFN-0-BKR-211-000B/004	BFN-0-BKR-211-000B/004, 4KV SD BD B/4, BKR 1822, TIE TO DIESEL GEN B	CASA-5	BFN-0-RLY-211-CASA-5	12HFA51A41F	SEIS_1C-3R8
BFN-0-BKR-211-000B/004	BFN-0-BKR-211-000B/004, 4KV SD BD B/4, BKR 1822, TIE TO DIESEL GEN B	86-GA-822	BFN-0-86-211-000B/004	GE 12HEA61A213X2	SEIS_1C-3R5
BFN-0-BKR-211-000B/004	BFN-0-BKR-211-000B/004, 4KV SD BD B/4, BKR 1822, TIE TO DIESEL GEN B	86-616	BFN-0-86-211-000B/002	HEA-61B	SEIS_1C-3R4
BFN-0-BKR-211-000B/004	BFN-0-BKR-211-000B/004, 4KV SD BD B/4, BKR 1822, TIE TO DIESEL GEN B	51-616 Phase A	BFN-0-51-211-000B/02A	GE IAC-51A	SEIS_1C-3R3
BFN-0-BKR-211-000B/004	BFN-0-BKR-211-000B/004, 4KV SD BD B/4, BKR 1822, TIE TO DIESEL GEN B	51-616 Phase C	BFN-0-51-211-000B/02C	GE IAC-51A	SEIS_1C-3R3
BFN-0-BKR-211-000B/004	BFN-0-BKR-211-000B/004, 4KV SD BD B/4, BKR 1822, TIE TO DIESEL GEN B	86-714	BFN-0-86-211-000B/020	HEA-61B	SEIS_1C-3R4

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-BKR-211-000B/004	BFN-0-BKR-211-000B/004, 4KV SD BD B/4, BKR 1822, TIE TO DIESEL GEN B	51-714 Phase A	BFN-0-51-211-000B/20A	GE IAC-51A	SEIS_1C-3R3
BFN-0-BKR-211-000B/004	BFN-0-BKR-211-000B/004, 4KV SD BD B/4, BKR 1822, TIE TO DIESEL GEN B	51-714 Phase c	BFN-0-51-211-000B/20C	GE IAC-51A	SEIS_1C-3R3
BFN-0-BKR-211-000B/004	BFN-0-BKR-211-000B/004, 4KV SD BD B/4, BKR 1822, TIE TO DIESEL GEN B	86/SB	BFN-0-86-211-000B/003	GE HEA-61B	SEIS_1C-3R4
BFN-0-BKR-211-000B/004	BFN-0-BKR-211-000B/004, 4KV SD BD B/4, BKR 1822, TIE TO DIESEL GEN B	R1B	BFN-0-RLY-082-R1B	12HFA51A42F	SEIS_1C-3R2
BFN-0-BKR-211-000B/004	BFN-0-BKR-211-000B/004, 4KV SD BD B/4, BKR 1822, TIE TO DIESEL GEN B	Breaker	BFN-0-BKR-211-000B/004	Wylie-Siemens Type 5-3AK-GEH-250-1200-58	Modeled as DG fails to start
BFN-0-BKR-211-000C/004	BFN-0-BKR-211-000C/004, 4KV SD BD C/4, BKR 1812, TIE TO DIESEL GEN C	CAR	BFN-0-RLY-082-2547CCAR	12HFA51A42F	SEIS_1C-6R2
BFN-0-BKR-211-000C/004	BFN-0-BKR-211-000C/004, 4KV SD BD C/4, BKR 1812, TIE TO DIESEL GEN C	51X1-C	51X-82-2547C/1	12HGA11J52	Chatter Acceptable
BFN-0-BKR-211-000C/004	BFN-0-BKR-211-000C/004, 4KV SD BD C/4, BKR 1812, TIE TO DIESEL GEN C	51VX-C	51V-82-2547C/X	GE 12PJV11AM2A	Chatter Acceptable
BFN-0-BKR-211-000C/004	BFN-0-BKR-211-000C/004, 4KV SD BD C/4, BKR 1812, TIE TO DIESEL GEN C	51VZ-C	51V-82-2547C/Z	GE 12PJV11AM2A	Chatter Acceptable

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-BKR-211-000C/004	BFN-0-BKR-211-000C/004, 4KV SD BD C/4, BKR 1812, TIE TO DIESEL GEN C	OTX	BFN-0-RLY-082-C/OTX	12HFA51A42F	SEIS_1C-6R2
BFN-0-BKR-211-000C/004	BFN-0-BKR-211-000C/004, 4KV SD BD C/4, BKR 1812, TIE TO DIESEL GEN C	OTR	BFN-0-RLY-082-C/OTR	Square D Class 8501 Type XUD0-1200	SEIS_1C-6R1
BFN-0-BKR-211-000C/004	BFN-0-BKR-211-000C/004, 4KV SD BD C/4, BKR 1812, TIE TO DIESEL GEN C	VLRC	BFN-0-RLY-082-VLRC	12HFA51A42F	SEIS_1C-6R2
BFN-0-BKR-211-000C/004	BFN-0-BKR-211-000C/004, 4KV SD BD C/4, BKR 1812, TIE TO DIESEL GEN C	VRRC	BFN-0-RLY-082-VRRC	12HFA51A42F	SEIS_1C-6R2
BFN-0-BKR-211-000C/004	BFN-0-BKR-211-000C/004, 4KV SD BD C/4, BKR 1812, TIE TO DIESEL GEN C	C/ESTR	BFN-0-RLY-082-C/ESTR	Square D DO/7001	SEIS_1C-6R1
BFN-0-BKR-211-000C/004	BFN-0-BKR-211-000C/004, 4KV SD BD C/4, BKR 1812, TIE TO DIESEL GEN C	CASB-2	BFN-0-RLY-211-CASB-2	12HFA51A41H	SEIS_1C-6R8
BFN-0-BKR-211-000C/004	BFN-0-BKR-211-000C/004, 4KV SD BD C/4, BKR 1812, TIE TO DIESEL GEN C	86-GA-812	BFN-0-86-211-000C/004	12HEA61A213X2	SEIS_1C-6R5
BFN-0-BKR-211-000C/004	BFN-0-BKR-211-000C/004, 4KV SD BD C/4, BKR 1812, TIE TO DIESEL GEN C	86-718	BFN-0-86-211-000C/022	HEA-61B	SEIS_1C-6R4
BFN-0-BKR-211-000C/004	BFN-0-BKR-211-000C/004, 4KV SD BD C/4, BKR 1812, TIE TO DIESEL GEN C	51-718 Phase A	BFN-0-51-211-000C/22A	GE IAC-51A	SEIS_1C-6R3

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-BKR-211-000C/004	BFN-0-BKR-211-000C/004, 4KV SD BD C/4, BKR 1812, TIE TO DIESEL GEN C	51-718 Phase C	BFN-0-51-211-000C/22C	GE IAC-51A	SEIS_1C-6R3
BFN-0-BKR-211-000C/004	BFN-0-BKR-211-000C/004, 4KV SD BD C/4, BKR 1812, TIE TO DIESEL GEN C	86-624	BFN-0-86-211-000C/002	HEA-61B	SEIS_1C-6R4
BFN-0-BKR-211-000C/004	BFN-0-BKR-211-000C/004, 4KV SD BD C/4, BKR 1812, TIE TO DIESEL GEN C	51-624 Phase A	BFN-0-51-211-000C/02A	GE IAC-51A	SEIS_1C-6R3
BFN-0-BKR-211-000C/004	BFN-0-BKR-211-000C/004, 4KV SD BD C/4, BKR 1812, TIE TO DIESEL GEN C	51-624 Phase C	BFN-0-51-211-000C/02C	GE IAC-51A	SEIS_1C-6R3
BFN-0-BKR-211-000C/004	BFN-0-BKR-211-000C/004, 4KV SD BD C/4, BKR 1812, TIE TO DIESEL GEN C	86/SC	BFN-0-86-211-000C/003	GE HEA-61B	SEIS_1C-6R4
BFN-0-BKR-211-000C/004	BFN-0-BKR-211-000C/004, 4KV SD BD C/4, BKR 1812, TIE TO DIESEL GEN C	R1C	BFN-0-RLY-082-R1C	12HFA51A42F	SEIS_1C-6R2
BFN-0-BKR-211-000C/004	BFN-0-BKR-211-000C/004, 4KV SD BD C/4, BKR 1812, TIE TO DIESEL GEN C	Breaker	BFN-0-BKR-211-000C/004	GE - 10AX012G10	Modeled as DG fails to start
BFN-0-BKR-211-000D/020	BFN-0-BKR-211-000D/020, 4KV SD BD D/20, BKR 1816, TIE TO DIESEL GEN D	CAR	BFN-0-RLY-082-2547DCAR	12HFA51A42F	SEIS_1C-3R2
BFN-0-BKR-211-000D/020	BFN-0-BKR-211-000D/020, 4KV SD BD D/20, BKR 1816, TIE TO DIESEL GEN D	51X1-D	51X-82-2547D/1	GE Q12HGA111J2	Chatter Acceptable

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-BKR-211-000D/020	BFN-0-BKR-211-000D/020, 4KV SD BD D/20, BKR 1816, TIE TO DIESEL GEN D	51VX-D	51V-82-2547D/X	GE 12PJV11AM2A	Chatter Acceptable
BFN-0-BKR-211-000D/020	BFN-0-BKR-211-000D/020, 4KV SD BD D/20, BKR 1816, TIE TO DIESEL GEN D	51VZ-D	51V-82-2547D/Z	Q12HGA111J2	Chatter Acceptable
BFN-0-BKR-211-000D/020	BFN-0-BKR-211-000D/020, 4KV SD BD D/20, BKR 1816, TIE TO DIESEL GEN D	OTX	BFN-0-RLY-082-D/OTX	12HFA51A42F	SEIS_1C-3R2
BFN-0-BKR-211-000D/020	BFN-0-BKR-211-000D/020, 4KV SD BD D/20, BKR 1816, TIE TO DIESEL GEN D	OTR	BFN-0-RLY-082-D/OTR	Square D Class 8501 Type XUD0-1200	SEIS_1C-3R1
BFN-0-BKR-211-000D/020	BFN-0-BKR-211-000D/020, 4KV SD BD D/20, BKR 1816, TIE TO DIESEL GEN D	VLRD	BFN-0-RLY-082-VLRD	12HFA51A42F	SEIS_1C-3R2
BFN-0-BKR-211-000D/020	BFN-0-BKR-211-000D/020, 4KV SD BD D/20, BKR 1816, TIE TO DIESEL GEN D	VRRD	BFN-0-RLY-082-VRRD	12HFA51A42F	SEIS_1C-3R2
BFN-0-BKR-211-000D/020	BFN-0-BKR-211-000D/020, 4KV SD BD D/20, BKR 1816, TIE TO DIESEL GEN D	D/ESTR	BFN-0-RLY-082-D/ESTR	Square D DO/7001	SEIS_1C-3R1
BFN-0-BKR-211-000D/020	BFN-0-BKR-211-000D/020, 4KV SD BD D/20, BKR 1816, TIE TO DIESEL GEN D	CASB-5	BFN-0-RLY-211-CASB-5	12HFA51A41H	SEIS_1C-3R8
BFN-0-BKR-211-000D/020	BFN-0-BKR-211-000D/020, 4KV SD BD D/20, BKR 1816, TIE TO DIESEL GEN D	86-GA-816	BFN-0-86-211-000D/020	12HEA61A213X2	SEIS_1C-3R5

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-BKR-211-000D/020	BFN-0-BKR-211-000D/020, 4KV SD BD D/20, BKR 1816, TIE TO DIESEL GEN D	86-724	BFN-0-86-211-000D/022	HEA-61B	SEIS_1C-3R4
BFN-0-BKR-211-000D/020	BFN-0-BKR-211-000D/020, 4KV SD BD D/20, BKR 1816, TIE TO DIESEL GEN D	51-724 Phase A	BFN-0-51-211-000D/22A	GE IAC-51A	SEIS_1C-3R3
BFN-0-BKR-211-000D/020	BFN-0-BKR-211-000D/020, 4KV SD BD D/20, BKR 1816, TIE TO DIESEL GEN D	51-724 Phase C	BFN-0-51-211-000D/22C	12IAC51A101A	SEIS_1C-3R3
BFN-0-BKR-211-000D/020	BFN-0-BKR-211-000D/020, 4KV SD BD D/20, BKR 1816, TIE TO DIESEL GEN D	86-618	BFN-0-86-211-000D/005	HEA-61B	SEIS_1C-3R4
BFN-0-BKR-211-000D/020	BFN-0-BKR-211-000D/020, 4KV SD BD D/20, BKR 1816, TIE TO DIESEL GEN D	51-618 Phase A	BFN-0-51-211-000D/05A	12IAC51A101A	SEIS_1C-3R3
BFN-0-BKR-211-000D/020	BFN-0-BKR-211-000D/020, 4KV SD BD D/20, BKR 1816, TIE TO DIESEL GEN D	51-618 Phase C	BFN-0-51-211-000D/05C	GE IAC-51A	SEIS_1C-3R3
BFN-0-BKR-211-000D/020	BFN-0-BKR-211-000D/020, 4KV SD BD D/20, BKR 1816, TIE TO DIESEL GEN D	86/SD	BFN-0-86-211-000D/21	GE HEA-61B	SEIS_1C-3R4
BFN-0-BKR-211-000D/020	BFN-0-BKR-211-000D/020, 4KV SD BD D/20, BKR 1816, TIE TO DIESEL GEN D	R1D	BFN-0-RLY-082-R1D	12HFA51A42F	SEIS_1C-3R2
BFN-0-BKR-211-000D/020	BFN-0-BKR-211-000D/020, 4KV SD BD D/20, BKR 1816, TIE TO DIESEL GEN D	Breaker	BFN-0-BKR-211-000D/020	GE - 10AX012G10	Modeled as DG fails to start

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-BKR-211-000A/005	BFN-0-BKR-211-000A/005, 4KV SD BD A CMPT 5, NOR FDR TO TRANS TS1A	50G-A/5	BFN-0-50G-211-000A/005	GE PJC-11A	SEIS_1C-1R6
BFN-0-BKR-211-000A/005	BFN-0-BKR-211-000A/005, 4KV SD BD A CMPT 5, NOR FDR TO TRANS TS1A	Breaker	BFN-0-BKR-211-000A/005	Wylie-Siemens Type 5-3AK-GEH-250-1200-58	SEIS_1C-1
BFN-0-BKR-211-000A/021	BFN-0-BKR-211-000A/021, 4KV SD BD A/21, NOR FDR TO TRANS TDA	50G-A/21	BFN-0-50G-211-000A/020	GE PJC-11A	SEIS_1C-1R6
BFN-0-BKR-211-000A/021	BFN-0-BKR-211-000A/021, 4KV SD BD A/21, NOR FDR TO TRANS TDA	Breaker	BFN-0-BKR-211-000A/021	Wylie-Siemens Type 5-3AK-GEH-250-1200-58	SEIS_1C-1
BFN-0-BKR-211-000B/005	BFN-0-BKR-211-000B/005, 4KV SD BD B/5, NOR FDR TO TRANS TS2A	50G-B/5	BFN-0-50G-211-000B/005	GE PJC-11A	SEIS_1C-3R6
BFN-0-BKR-211-000B/005	BFN-0-BKR-211-000B/005, 4KV SD BD B/5, NOR FDR TO TRANS TS2A	Breaker	BFN-0-BKR-211-000B/005	Wylie-Siemens Type 5-3AK-GEH-250-1200-58	SEIS_1C-3
BFN-0-BKR-211-000B/014	BFN-0-BKR-211-000B/014, 4KV SD BD B/14, NOR FDR TO TRANS TS1E AND TDE	50G-B/14	BFN-0-50G-211-000B/014	GE PJC-11A	SEIS_1C-3R6
BFN-0-BKR-211-000B/014	BFN-0-BKR-211-000B/014, 4KV SD BD B/14, NOR FDR TO TRANS TS1E AND TDE	Breaker	BFN-0-BKR-211-000B/014	Wylie-Siemens Type 5-3AK-GEH-250-1200-58	SEIS_1C-3
BFN-0-BKR-211-000C/020	BFN-0-BKR-211-000C/020, 4KV SD BD C/20, NOR FDR FOR TRANS TS1B	50G-C/20	BFN-0-50G-211-000C/020	GE PJC-11A	SEIS_1C-6R6

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-BKR-211-000C/020	BFN-0-BKR-211-000C/020, 4KV SD BD C/20, NOR FDR FOR TRANS TS1B	Breaker	BFN-0-BKR-211-000C/020	Wylie-Siemens Type 5-3AK-GEH-250-1200-58	SEIS_1C-1
BFN-0-BKR-211-000C/005	BFN-0-BKR-211-000C/005, 4KV SD BD C/5 NOR FDR FOR TRANS TS2E	50G-C/5	BFN-0-50G-211-000C/005	GE PJC-11A	SEIS_1C-6R6
BFN-0-BKR-211-000C/005	BFN-0-BKR-211-000C/005, 4KV SD BD C/5 NOR FDR FOR TRANS TS2E	Breaker	BFN-0-BKR-211-000C/005	Wylie-Siemens Type 5-3AK-GEH-250-1200-58	SEIS_1C-1
BFN-0-BKR-211-000D/013	BFN-0-BKR-211-000D/013, 4KV SD BD D/13, NOR FDR FOR TRANS TDB	50G-D/13	BFN-0-50G-211-000D/013	GE PJC-11A	SEIS_1C-3R6
BFN-0-BKR-211-000D/013	BFN-0-BKR-211-000D/013, 4KV SD BD D/13, NOR FDR FOR TRANS TDB	Breaker	BFN-0-BKR-211-000D/013	Wylie-Siemens Type 5-3AK-GEH-250-1200-58	SEIS_1C-3
BFN-0-BKR-211-000D/019	BFN-0-BKR-211-000D/019, 4KV SD BD D/19, NOR FDR FOR TRANS TS2B	50G-D/19	BFN-0-50G-211-000D/019	GE PJC-11A	SEIS_1C-3R6
BFN-0-BKR-211-000D/019	BFN-0-BKR-211-000D/019, 4KV SD BD D/19, NOR FDR FOR TRANS TS2B	Breaker	BFN-0-BKR-211-000D/019	GE Hitachi NE - 317A7502P005	SEIS_1C-3
BFN-0-BKR-211-000A/024	BFN-0-BKR-211-000A/024, 4KV SD BD A, ALT FDR BKR 1716	86-2-716	BFN-0-86-211-000A/024	GE HEA-61B	SEIS_1C-1R4
BFN-0-BKR-211-000A/024	BFN-0-BKR-211-000A/024, 4KV SD BD A, ALT FDR BKR 1716	86-716	BFN-0-86-211-000A/024	HEA-61B	SEIS_1C-1R4

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-BKR-211-000A/024	BFN-0-BKR-211-000A/024, 4KV SD BD A, ALT FDR BKR 1716	51-716 Phase A	BFN-0-51-211-000A/24A	GE IAC-51A	SEIS_1C-1R3
BFN-0-BKR-211-000A/024	BFN-0-BKR-211-000A/024, 4KV SD BD A, ALT FDR BKR 1716	51-716 Phase C	BFN-0-51-211-000A/24C	GE IAC-51A	SEIS_1C-1R3
BFN-0-BKR-211-000A/024	BFN-0-BKR-211-000A/024, 4KV SD BD A, ALT FDR BKR 1716	86-614	BFN-0-86-211-000A/003	HEA-61B	SEIS_1C-1R4
BFN-0-BKR-211-000A/024	BFN-0-BKR-211-000A/024, 4KV SD BD A, ALT FDR BKR 1716	51-614 Phase A	BFN-0-51-211-000A/03A	GE IAC-51A	SEIS_1C-1R3
BFN-0-BKR-211-000A/024	BFN-0-BKR-211-000A/024, 4KV SD BD A, ALT FDR BKR 1716	51-614 Phase C	BFN-0-51-211-000A/03C	GE IAC-51A	SEIS_1C-1R3
BFN-0-BKR-211-000A/024	BFN-0-BKR-211-000A/024, 4KV SD BD A, ALT FDR BKR 1716	86-SA	BFN-0-86-211-000A/23	HEA-61B	SEIS_1C-1R4
BFN-0-BKR-211-000A/024	BFN-0-BKR-211-000A/024, 4KV SD BD A, ALT FDR BKR 1716	27SAX	BFN-0-27-211-000A/12J	12HFA51A41F	SEIS_1C-1R7
BFN-0-BKR-211-000A/024	BFN-0-BKR-211-000A/024, 4KV SD BD A, ALT FDR BKR 1716	CASA-2	BFN-0-RLY-211-CASA-2	12HFA51A41H	SEIS_1C-1R8
BFN-0-BKR-211-000A/024	BFN-0-BKR-211-000A/024, 4KV SD BD A, ALT FDR BKR 1716	CASB-2	BFN-0-RLY-211-CASB-2	12HFA51A41H	SEIS_1C-6R8

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-BKR-211-000A/024	BFN-0-BKR-211-000A/024, 4KV SD BD A, ALT FDR BKR 1716	Breaker	BFN-0-BKR-211-000A/024	GE Hitachi NE - Q10AX012G6	Alternate feeder breakers not modeled
BFN-0-BKR-211-000B/020	BFN-0-BKR-211-000B/020, 4KV SD BD B/20, BKR 1714, ALT FDR FROM SD BUS 2	86-2-714	BFN-0-86-211-000B/020	GE HEA-61B	SEIS_1C-3R4
BFN-0-BKR-211-000B/020	BFN-0-BKR-211-000B/020, 4KV SD BD B/20, BKR 1714, ALT FDR FROM SD BUS 2	86-714	BFN-0-86-211-000B/020	HEA-61B	SEIS_1C-3R4
BFN-0-BKR-211-000B/020	BFN-0-BKR-211-000B/020, 4KV SD BD B/20, BKR 1714, ALT FDR FROM SD BUS 2	51-714 Phase A	BFN-0-51-211-000B/20A	GE IAC-51A	SEIS_1C-3R3
BFN-0-BKR-211-000B/020	BFN-0-BKR-211-000B/020, 4KV SD BD B/20, BKR 1714, ALT FDR FROM SD BUS 2	51-714 Phase C	BFN-0-51-211-000B/20C	GE IAC-51A	SEIS_1C-3R3
BFN-0-BKR-211-000B/020	BFN-0-BKR-211-000B/020, 4KV SD BD B/20, BKR 1714, ALT FDR FROM SD BUS 2	86-616	BFN-0-86-211-000B/002	HEA-61B	SEIS_1C-3R4
BFN-0-BKR-211-000B/020	BFN-0-BKR-211-000B/020, 4KV SD BD B/20, BKR 1714, ALT FDR FROM SD BUS 2	51-616 Phase A	BFN-0-51-211-000B/02A	GE IAC-51A	SEIS_1C-3R3
BFN-0-BKR-211-000B/020	BFN-0-BKR-211-000B/020, 4KV SD BD B/20, BKR 1714, ALT FDR FROM SD BUS 2	51-616 Phase C	BFN-0-51-211-000B/02C	GE IAC-51A	SEIS_1C-3R3
BFN-0-BKR-211-000B/020	BFN-0-BKR-211-000B/020, 4KV SD BD B/20, BKR 1714, ALT FDR FROM SD BUS 2	86-SB	BFN-0-86-211-000B/003	GE HEA-61B	SEIS_1C-3R4

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-BKR-211-000B/020	BFN-0-BKR-211-000B/020, 4KV SD BD B/20, BKR 1714, ALT FDR FROM SD BUS 2	27SBX	BFN-0-27-211-000B/12J	12HFA51A41F	SEIS_1C-3R7
BFN-0-BKR-211-000B/020	BFN-0-BKR-211-000B/020, 4KV SD BD B/20, BKR 1714, ALT FDR FROM SD BUS 2	CASA-5	BFN-0-RLY-211-CASA-5	12HFA51A41H	SEIS_1C-3R8
BFN-0-BKR-211-000B/020	BFN-0-BKR-211-000B/020, 4KV SD BD B/20, BKR 1714, ALT FDR FROM SD BUS 2	CASB-5	BFN-0-RLY-211-CASB-5	12HFA51A41H	SEIS_1C-3R8
BFN-0-BKR-211-000B/020	BFN-0-BKR-211-000B/020, 4KV SD BD B/20, BKR 1714, ALT FDR FROM SD BUS 2	Breaker	BFN-0-BKR-211-000B/020	Wylie-Siemens Type 5-3AK-GEH-250-1200-58	Alternate feeder breakers not modeled
BFN-0-BKR-211-000D/005	BFN-0-BKR-211-000D/005, 4KV SD BD D/5, BKR 1618, ALT FDR FROM SD BUS 1	86-2-618	BFN-0-86-211-000D/005	GE HEA-61B	SEIS_1C-3R4
BFN-0-BKR-211-000D/005	BFN-0-BKR-211-000D/005, 4KV SD BD D/5, BKR 1618, ALT FDR FROM SD BUS 1	86-618	BFN-0-86-211-000D/005	HEA-61B	SEIS_1C-3R4
BFN-0-BKR-211-000D/005	BFN-0-BKR-211-000D/005, 4KV SD BD D/5, BKR 1618, ALT FDR FROM SD BUS 1	51-618 Phase A	BFN-0-51-211-000D/05A	12IAC51A101A	SEIS_1C-3R3
BFN-0-BKR-211-000D/005	BFN-0-BKR-211-000D/005, 4KV SD BD D/5, BKR 1618, ALT FDR FROM SD BUS 1	51-618 Phase C	BFN-0-51-211-000D/05C	GE IAC-51A	SEIS_1C-3R3
BFN-0-BKR-211-000D/005	BFN-0-BKR-211-000D/005, 4KV SD BD D/5, BKR 1618, ALT FDR FROM SD BUS 1	86-718	BFN-0-86-211-000C/022	HEA-61B	SEIS_1C-6R4

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-BKR-211-000D/005	BFN-0-BKR-211-000D/005, 4KV SD BD D/5, BKR 1618, ALT FDR FROM SD BUS 1	51-718 Phase A	BFN-0-51-211-000C/22A	GE IAC-51A	SEIS_1C-6R3
BFN-0-BKR-211-000D/005	BFN-0-BKR-211-000D/005, 4KV SD BD D/5, BKR 1618, ALT FDR FROM SD BUS 1	51-718 Phase C	BFN-0-51-211-000C/22C	GE IAC-51A	SEIS_1C-6R3
BFN-0-BKR-211-000D/005	BFN-0-BKR-211-000D/005, 4KV SD BD D/5, BKR 1618, ALT FDR FROM SD BUS 1	86-SC	BFN-0-86-211-000C/003	HEA-61B	SEIS_1C-6R4
BFN-0-BKR-211-000D/005	BFN-0-BKR-211-000D/005, 4KV SD BD D/5, BKR 1618, ALT FDR FROM SD BUS 1	27SCX	BFN-0-27-211-000C/11H	12HFA51A41F	SEIS_11-1R1-1
BFN-0-BKR-211-000D/005	BFN-0-BKR-211-000D/005, 4KV SD BD D/5, BKR 1618, ALT FDR FROM SD BUS 1	CASA-5	BFN-0-RLY-211-CASA-5	12HFA51A41H	SEIS_1C-3R8
BFN-0-BKR-211-000D/005	BFN-0-BKR-211-000D/005, 4KV SD BD D/5, BKR 1618, ALT FDR FROM SD BUS 1	CASB-5	BFN-0-RLY-211-CASB-5	12HFA51A41H	SEIS_1C-3R8
BFN-0-BKR-211-000D/005	BFN-0-BKR-211-000D/005, 4KV SD BD D/5, BKR 1618, ALT FDR FROM SD BUS 1	Breaker	BFN-0-BKR-211-000D/005	GE Hitachi NE - Q10AX012G6	Alternate feeder breakers not modeled
BFN-0-BKR-211-000C/002	BFN-0-BKR-211-000C/002, 4KV SD BD C/2, BKR 1624, ALT FDR FROM SD BUS 1	86-2-624	BFN-0-86-211-000C/002	GE HEA-61B	SEIS_1C-6R4
BFN-0-BKR-211-000C/002	BFN-0-BKR-211-000C/002, 4KV SD BD C/2, BKR 1624, ALT FDR FROM SD BUS 1	86-624	BFN-0-86-211-000C/002	HEA-61B	SEIS_1C-6R4

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-BKR-211-000C/002	BFN-0-BKR-211-000C/002, 4KV SD BD C/2, BKR 1624, ALT FDR FROM SD BUS 1	51-624 Phase A	BFN-0-51-211-000C/02A	GE IAC-51A	SEIS_1C-6R3
BFN-0-BKR-211-000C/002	BFN-0-BKR-211-000C/002, 4KV SD BD C/2, BKR 1624, ALT FDR FROM SD BUS 1	51-624 Phase C	BFN-0-51-211-000C/02C	GE IAC-51A	SEIS_1C-6R3
BFN-0-BKR-211-000C/002	BFN-0-BKR-211-000C/002, 4KV SD BD C/2, BKR 1624, ALT FDR FROM SD BUS 1	86-724	BFN-0-86-211-000D/022	HEA-61B	SEIS_1C-3R4
BFN-0-BKR-211-000C/002	BFN-0-BKR-211-000C/002, 4KV SD BD C/2, BKR 1624, ALT FDR FROM SD BUS 1	51-724 Phase A	BFN-0-51-211-000D/22A	GE IAC-51A	SEIS_1C-3R3
BFN-0-BKR-211-000C/002	BFN-0-BKR-211-000C/002, 4KV SD BD C/2, BKR 1624, ALT FDR FROM SD BUS 1	51-724 Phase C	BFN-0-51-211-000D/22C	12IAC51A101A	SEIS_1C-3R3
BFN-0-BKR-211-000C/002	BFN-0-BKR-211-000C/002, 4KV SD BD C/2, BKR 1624, ALT FDR FROM SD BUS 1	86-SD	BFN-0-86-211-000D/021	HEA-61B	SEIS_1C-3R4
BFN-0-BKR-211-000C/002	BFN-0-BKR-211-000C/002, 4KV SD BD C/2, BKR 1624, ALT FDR FROM SD BUS 1	27SDX	BFN-0-27-211-000D/11H	12HFA51A41F	SEIS_11-1R1-2
BFN-0-BKR-211-000C/002	BFN-0-BKR-211-000C/002, 4KV SD BD C/2, BKR 1624, ALT FDR FROM SD BUS 1	CASA-2	BFN-0-RLY-211-CASA-2	12HFA51A41H	SEIS_1C-1R8
BFN-0-BKR-211-000C/002	BFN-0-BKR-211-000C/002, 4KV SD BD C/2, BKR 1624, ALT FDR FROM SD BUS 1	CASB-2	BFN-0-RLY-211-CASB-2	12HFA51A41H	SEIS_1C-6R8

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-BKR-211-000C/002	BFN-0-BKR-211-000C/002, 4KV SD BD C/2, BKR 1624, ALT FDR FROM SD BUS 1	Breaker	BFN-0-BKR-211-000C/002	GE Hitachi NE - Q10AX012G6	Alternate feeder breakers not modeled
BFN-0-BKR-211-000A/003	BFN-0-BKR-211-000A/003, 4KV SD BD A/3, BKR 1614, NOR FDR FROM SD BUS 1	86-1-614	BFN-0-86-211-000A/003	GE HEA-61B	SEIS_1C-1R4
BFN-0-BKR-211-000A/003	BFN-0-BKR-211-000A/003, 4KV SD BD A/3, BKR 1614, NOR FDR FROM SD BUS 1	86-614	BFN-0-86-211-000A/003	HEA-61B	SEIS_1C-1R4
BFN-0-BKR-211-000A/003	BFN-0-BKR-211-000A/003, 4KV SD BD A/3, BKR 1614, NOR FDR FROM SD BUS 1	51-614 Phase A	BFN-0-51-211-000A/03A	GE IAC-51A	SEIS_1C-1R3
BFN-0-BKR-211-000A/003	BFN-0-BKR-211-000A/003, 4KV SD BD A/3, BKR 1614, NOR FDR FROM SD BUS 1	51-614 Phase C	BFN-0-51-211-000A/03C	GE IAC-51A	SEIS_1C-1R3
BFN-0-BKR-211-000A/003	BFN-0-BKR-211-000A/003, 4KV SD BD A/3, BKR 1614, NOR FDR FROM SD BUS 1	86-716	BFN-0-86-211-000A/024	HEA-61B	SEIS_1C-1R4
BFN-0-BKR-211-000A/003	BFN-0-BKR-211-000A/003, 4KV SD BD A/3, BKR 1614, NOR FDR FROM SD BUS 1	51-716 Phase A	BFN-0-51-211-000A/24A	GE IAC-51A	SEIS_1C-1R3
BFN-0-BKR-211-000A/003	BFN-0-BKR-211-000A/003, 4KV SD BD A/3, BKR 1614, NOR FDR FROM SD BUS 1	51-716 Phase C	BFN-0-51-211-000A/24C	GE IAC-51A	SEIS_1C-1R3
BFN-0-BKR-211-000A/003	BFN-0-BKR-211-000A/003, 4KV SD BD A/3, BKR 1614, NOR FDR FROM SD BUS 1	86-SA	BFN-0-86-211-000A/023	HEA-61B	SEIS_1C-1R4

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-BKR-211-000A/003	BFN-0-BKR-211-000A/003, 4KV SD BD A/3, BKR 1614, NOR FDR FROM SD BUS 1	27SAX	BFN-0-27-211-000A/12J	12HFA51A41F	SEIS_1C-1R7
BFN-0-BKR-211-000A/003	BFN-0-BKR-211-000A/003, 4KV SD BD A/3, BKR 1614, NOR FDR FROM SD BUS 1	Breaker	BFN-0-BKR-211-000A/003	Wylie-Siemens Type 5-3AK-GEH-250-1200-58	SEIS_1C-1
BFN-0-BKR-211-000B/002	BFN-0-BKR-211-000B/002, 4KV SD BD B/2, BKR 1616, NOR FDR FROM SD BUS 1	86-1-616	BFN-0-86-211-000B/002	GE HEA-61B	SEIS_1C-3R4
BFN-0-BKR-211-000B/002	BFN-0-BKR-211-000B/002, 4KV SD BD B/2, BKR 1616, NOR FDR FROM SD BUS 1	86-616	BFN-0-86-211-000B/002	HEA-61B	SEIS_1C-3R4
BFN-0-BKR-211-000B/002	BFN-0-BKR-211-000B/002, 4KV SD BD B/2, BKR 1616, NOR FDR FROM SD BUS 1	51-616 Phase A	BFN-0-51-211-000B/02A	GE IAC-51A	SEIS_1C-3R3
BFN-0-BKR-211-000B/002	BFN-0-BKR-211-000B/002, 4KV SD BD B/2, BKR 1616, NOR FDR FROM SD BUS 1	51-616 Phase C	BFN-0-51-211-000B/02C	GE IAC-51A	SEIS_1C-3R3
BFN-0-BKR-211-000B/002	BFN-0-BKR-211-000B/002, 4KV SD BD B/2, BKR 1616, NOR FDR FROM SD BUS 1	86-714	BFN-0-86-211-000B/020	HEA-61B	SEIS_1C-3R4
BFN-0-BKR-211-000B/002	BFN-0-BKR-211-000B/002, 4KV SD BD B/2, BKR 1616, NOR FDR FROM SD BUS 1	51-714 Phase A	BFN-0-51-211-000B/20A	GE IAC-51A	SEIS_1C-3R3
BFN-0-BKR-211-000B/002	BFN-0-BKR-211-000B/002, 4KV SD BD B/2, BKR 1616, NOR FDR FROM SD BUS 1	51-714 Phase C	BFN-0-51-211-000B/20C	GE IAC-51A	SEIS_1C-3R3

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-BKR-211-000B/002	BFN-0-BKR-211-000B/002, 4KV SD BD B/2, BKR 1616, NOR FDR FROM SD BUS 1	86-SB	BFN-0-86-211-000B/003	HEA-61B	SEIS_1C-3R4
BFN-0-BKR-211-000B/002	BFN-0-BKR-211-000B/002, 4KV SD BD B/2, BKR 1616, NOR FDR FROM SD BUS 1	27SBX	BFN-0-27-211-000B/12J	12HFA51A41F	SEIS_1C-3R7
BFN-0-BKR-211-000B/002	BFN-0-BKR-211-000B/002, 4KV SD BD B/2, BKR 1616, NOR FDR FROM SD BUS 1	Breaker	BFN-0-BKR-211-000B/002	Wylie-Siemens Type 5-3AK-GEH-250-1200-58	SEIS_1C-3
BFN-0-BKR-211-000C/022	BFN-0-BKR-211-000C/022, 4KV SD BD C/22, BKR 1718, NOR FDR FROM SD BUS 2	86-1-718	BFN-0-86-211-000C/022	GE HEA-61B	SEIS_1C-6R4
BFN-0-BKR-211-000C/022	BFN-0-BKR-211-000C/022, 4KV SD BD C/22, BKR 1718, NOR FDR FROM SD BUS 2	86-718	BFN-0-86-211-000C/022	HEA-61B	SEIS_1C-6R4
BFN-0-BKR-211-000C/022	BFN-0-BKR-211-000C/022, 4KV SD BD C/22, BKR 1718, NOR FDR FROM SD BUS 2	51-718 Phase A	BFN-0-51-211-000C/22A	GE IAC-51A	SEIS_1C-6R3
BFN-0-BKR-211-000C/022	BFN-0-BKR-211-000C/022, 4KV SD BD C/22, BKR 1718, NOR FDR FROM SD BUS 2	51-718 Phase C	BFN-0-51-211-000C/22C	GE IAC-51A	SEIS_1C-6R3
BFN-0-BKR-211-000C/022	BFN-0-BKR-211-000C/022, 4KV SD BD C/22, BKR 1718, NOR FDR FROM SD BUS 2	86-624	BFN-0-86-211-000C/002	HEA-61B	SEIS_1C-6R4
BFN-0-BKR-211-000C/022	BFN-0-BKR-211-000C/022, 4KV SD BD C/22, BKR 1718, NOR FDR FROM SD BUS 2	51-624 Phase A	BFN-0-51-211-000C/02A	GE IAC-51A	SEIS_1C-6R3

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-BKR-211-000C/022	BFN-0-BKR-211-000C/022, 4KV SD BD C/22, BKR 1718, NOR FDR FROM SD BUS 2	51-624 Phase C	BFN-0-51-211-000C/02C	GE IAC-51A	SEIS_1C-6R3
BFN-0-BKR-211-000C/022	BFN-0-BKR-211-000C/022, 4KV SD BD C/22, BKR 1718, NOR FDR FROM SD BUS 2	86-SC	BFN-0-86-211-000C/003	HEA-61B	SEIS_1C-6R4
BFN-0-BKR-211-000C/022	BFN-0-BKR-211-000C/022, 4KV SD BD C/22, BKR 1718, NOR FDR FROM SD BUS 2	27SCX	BFN-0-27-211-000C/11H	12HFA51A41F	SEIS_11-1R1-1
BFN-0-BKR-211-000C/022	BFN-0-BKR-211-000C/022, 4KV SD BD C/22, BKR 1718, NOR FDR FROM SD BUS 2	Breaker	BFN-0-BKR-211-000C/022	GE Hitachi NE - Q10AX012G6	SEIS_1C-1
BFN-0-BKR-211-000D/022	BFN-0-BKR-211-000D/022, 4KV SD BD D/22, BKR 1724, NOR FDR FROM SD BUS 2	86-1-724	BFN-0-86-211-000D/022	GE HEA-61B	SEIS_1C-3R4
BFN-0-BKR-211-000D/022	BFN-0-BKR-211-000D/022, 4KV SD BD D/22, BKR 1724, NOR FDR FROM SD BUS 2	86-724	BFN-0-86-211-000D/022	HEA-61B	SEIS_1C-3R4
BFN-0-BKR-211-000D/022	BFN-0-BKR-211-000D/022, 4KV SD BD D/22, BKR 1724, NOR FDR FROM SD BUS 2	51-724 Phase A	BFN-0-51-211-000D/22A	GE IAC-51A	SEIS_1C-3R3
BFN-0-BKR-211-000D/022	BFN-0-BKR-211-000D/022, 4KV SD BD D/22, BKR 1724, NOR FDR FROM SD BUS 2	51-724 Phase C	BFN-0-51-211-000D/22C	12IAC51A101A	SEIS_1C-3R3
BFN-0-BKR-211-000D/022	BFN-0-BKR-211-000D/022, 4KV SD BD D/22, BKR 1724, NOR FDR FROM SD BUS 2	86-618	BFN-0-86-211-000D/005	HEA-61B	SEIS_1C-3R4

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-BKR-211-000D/022	BFN-0-BKR-211-000D/022, 4KV SD BD D/22, BKR 1724, NOR FDR FROM SD BUS 2	51-618 Phase A	BFN-0-51-211-000D/05A	12IAC51A101A	SEIS_1C-3R3
BFN-0-BKR-211-000D/022	BFN-0-BKR-211-000D/022, 4KV SD BD D/22, BKR 1724, NOR FDR FROM SD BUS 2	51-618 Phase B	BFN-0-51-211-000D/05C	GE IAC-51A	SEIS_1C-3R3
BFN-0-BKR-211-000D/022	BFN-0-BKR-211-000D/022, 4KV SD BD D/22, BKR 1724, NOR FDR FROM SD BUS 2	86-SD	BFN-0-86-211-000D/021	HEA-61B	SEIS_1C-3R4
BFN-0-BKR-211-000D/022	BFN-0-BKR-211-000D/022, 4KV SD BD D/22, BKR 1724, NOR FDR FROM SD BUS 2	27SDX	BFN-0-27-211-000D/11H	12HFA51A41F	SEIS_11-1R1-2
BFN-0-BKR-211-000D/022	BFN-0-BKR-211-000D/022, 4KV SD BD D/22, BKR 1724, NOR FDR FROM SD BUS 2	Breaker	BFN-0-BKR-211-000D/022	GE Hitachi NE - 317A7502P005	SEIS_1C-3
BFN-3-BKR-211-03EA/009	BFN-3-BKR-211-03EA/009, 4KV SD BD 3EA/9, BKR 1838, TIE TO DIESEL GEN 3A	A/ESTR	BFN-3-RLY-082-A/ESTR	Square D DO/7001	SEIS_1C-4R1
BFN-3-BKR-211-03EA/009	BFN-3-BKR-211-03EA/009, 4KV SD BD 3EA/9, BKR 1838, TIE TO DIESEL GEN 3A	51X1-A	BFN-3-RLY-082-2547A/1	12HGA11J51	Chatter Acceptable
BFN-3-BKR-211-03EA/009	BFN-3-BKR-211-03EA/009, 4KV SD BD 3EA/9, BKR 1838, TIE TO DIESEL GEN 3A	CAR	BFN-3-RLY-082-2547ACAR	12HFA51A42F	SEIS_1C-4R2
BFN-3-BKR-211-03EA/009	BFN-3-BKR-211-03EA/009, 4KV SD BD 3EA/9, BKR 1838, TIE TO DIESEL GEN 3A	51VX-A	BFN-3-51V-082-2547A/X	GE 12PJV11AM2A	Chatter Acceptable

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-BKR-211-03EA/009	BFN-3-BKR-211-03EA/009, 4KV SD BD 3EA/9, BKR 1838, TIE TO DIESEL GEN 3A	51VZ-A	BFN-3-51V-082-2547A/Z	12HFA51A42F	Chatter Acceptable
BFN-3-BKR-211-03EA/009	BFN-3-BKR-211-03EA/009, 4KV SD BD 3EA/9, BKR 1838, TIE TO DIESEL GEN 3A	R3A	BFN-3-RLY-082-R3A	12HFA51A42H	SEIS_1C-4R3
BFN-3-BKR-211-03EA/009	BFN-3-BKR-211-03EA/009, 4KV SD BD 3EA/9, BKR 1838, TIE TO DIESEL GEN 3A	CASA-2	BFN-3-RLY-211-CASA-2	12HFA51A41F	SEIS_1C-4R8
BFN-3-BKR-211-03EA/009	BFN-3-BKR-211-03EA/009, 4KV SD BD 3EA/9, BKR 1838, TIE TO DIESEL GEN 3A	CASB-2	BFN-3-RLY-211-CASB-2	12HFA151A1F	SEIS_1C-4R8
BFN-3-BKR-211-03EA/009	BFN-3-BKR-211-03EA/009, 4KV SD BD 3EA/9, BKR 1838, TIE TO DIESEL GEN 3A	86-G3A-838	BFN-3-86-211-03EA/09	12HEA61A213X2	SEIS_1C-4R6
BFN-3-BKR-211-03EA/009	BFN-3-BKR-211-03EA/009, 4KV SD BD 3EA/9, BKR 1838, TIE TO DIESEL GEN 3A	86-334	BFN-3-86-211-03EA/07	GE HEA61C218	SEIS_1C-4R4
BFN-3-BKR-211-03EA/009	BFN-3-BKR-211-03EA/009, 4KV SD BD 3EA/9, BKR 1838, TIE TO DIESEL GEN 3A	51-334 Phase A	BFN-3-51-211-03EA/07A	GE 12IAC51A101A	SEIS_1C-4R5-1
BFN-3-BKR-211-03EA/009	BFN-3-BKR-211-03EA/009, 4KV SD BD 3EA/9, BKR 1838, TIE TO DIESEL GEN 3A	51-334 Phase B	BFN-3-51-211-03EA/07C	GE 12IAC51A101A	SEIS_1C-4R5-1
BFN-3-BKR-211-03EA/009	BFN-3-BKR-211-03EA/009, 4KV SD BD 3EA/9, BKR 1838, TIE TO DIESEL GEN 3A	86-726	BFN-3-86-211-03EA/04	12HEA61C218X2	SEIS_1C-4R4

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-BKR-211-03EA/009	BFN-3-BKR-211-03EA/009, 4KV SD BD 3EA/9, BKR 1838, TIE TO DIESEL GEN 3A	51-726 Phase A	BFN-3-51-211-03EA/04A	GE 12IAC51A101A	SEIS_1C-4R5-1
BFN-3-BKR-211-03EA/009	BFN-3-BKR-211-03EA/009, 4KV SD BD 3EA/9, BKR 1838, TIE TO DIESEL GEN 3A	51-726 Phase B	BFN-3-51-211-03EA/04C	GE 12IAC51A101A	SEIS_1C-4R5-1
BFN-3-BKR-211-03EA/009	BFN-3-BKR-211-03EA/009, 4KV SD BD 3EA/9, BKR 1838, TIE TO DIESEL GEN 3A	86-S3A	BFN-3-86-211-03EA/08	12HEA61C220X2	SEIS_1C-4R5-2
BFN-3-BKR-211-03EA/009	BFN-3-BKR-211-03EA/009, 4KV SD BD 3EA/9, BKR 1838, TIE TO DIESEL GEN 3A	OTX	BFN-3-RLY-082-A/OTX	12HFA51A42H	SEIS_1C-4R2
BFN-3-BKR-211-03EA/009	BFN-3-BKR-211-03EA/009, 4KV SD BD 3EA/9, BKR 1838, TIE TO DIESEL GEN 3A	OTR	BFN-3-RLY-082-A/OTR	Square D Class 8501 Type XUD0-1200	SEIS_1C-4R1
BFN-3-BKR-211-03EA/009	BFN-3-BKR-211-03EA/009, 4KV SD BD 3EA/9, BKR 1838, TIE TO DIESEL GEN 3A	Breaker	BFN-3-BKR-211-03EA/009	Wylie-Siemens Type 5-3AK-GEH-250-1200-58	Modeled as DG fails to start
BFN-3-BKR-211-03EB/011	BFN-3-BKR-211-03EB/011, 4KV SD BD 3EB/11, BKR 1842, TIE TO DIESEL GEN 3B	B/ESTR	BFN-3-RLY-082-B/ESTR	Square D DO/7001	SEIS_1C-5R1
BFN-3-BKR-211-03EB/011	BFN-3-BKR-211-03EB/011, 4KV SD BD 3EB/11, BKR 1842, TIE TO DIESEL GEN 3B	51X1-B	BFN-3-RLY-082-2547B/1	12HGA11J51	Chatter Acceptable
BFN-3-BKR-211-03EB/011	BFN-3-BKR-211-03EB/011, 4KV SD BD 3EB/11, BKR 1842, TIE TO DIESEL GEN 3B	CAR	BFN-3-RLY-082-2547BCAR	12HFA51A42F	SEIS_1C-5R2

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-BKR-211-03EB/011	BFN-3-BKR-211-03EB/011, 4KV SD BD 3EB/11, BKR 1842, TIE TO DIESEL GEN 3B	51VX-B	BFN-3-51V-082-2547B/X	GE 12PJV11AM2A	Chatter Acceptable
BFN-3-BKR-211-03EB/011	BFN-3-BKR-211-03EB/011, 4KV SD BD 3EB/11, BKR 1842, TIE TO DIESEL GEN 3B	51VZ-B	BFN-3-51V-082-2547B/Z	12HFA51A42F	Chatter Acceptable
BFN-3-BKR-211-03EB/011	BFN-3-BKR-211-03EB/011, 4KV SD BD 3EB/11, BKR 1842, TIE TO DIESEL GEN 3B	R3B	BFN-3-RLY-082-R3B	12HFA51A42H	SEIS_1C-5R3
BFN-3-BKR-211-03EB/011	BFN-3-BKR-211-03EB/011, 4KV SD BD 3EB/11, BKR 1842, TIE TO DIESEL GEN 3B	CASA-5	BFN-3-RLY-211-CASA-5	12HFA51A41F	SEIS_1C-5R8
BFN-3-BKR-211-03EB/011	BFN-3-BKR-211-03EB/011, 4KV SD BD 3EB/11, BKR 1842, TIE TO DIESEL GEN 3B	CASB-5	BFN-3-RLY-211-CASB-5	12HFA151A1F	SEIS_1C-5R8
BFN-3-BKR-211-03EB/011	BFN-3-BKR-211-03EB/011, 4KV SD BD 3EB/11, BKR 1842, TIE TO DIESEL GEN 3B	86-G3B-842	BFN-3-86-211-03EB/11	12HEA61A213X2	SEIS_1C-5R6
BFN-3-BKR-211-03EB/011	BFN-3-BKR-211-03EB/011, 4KV SD BD 3EB/11, BKR 1842, TIE TO DIESEL GEN 3B	86-336	BFN-3-86-211-03EB/14	12HEA61C218X2	SEIS_1C-5R4-1
BFN-3-BKR-211-03EB/011	BFN-3-BKR-211-03EB/011, 4KV SD BD 3EB/11, BKR 1842, TIE TO DIESEL GEN 3B	51-336 Phase A	BFN-3-51-211-03EB/14A	GE 12IAC51A101A	SEIS_1C-5R5
BFN-3-BKR-211-03EB/011	BFN-3-BKR-211-03EB/011, 4KV SD BD 3EB/11, BKR 1842, TIE TO DIESEL GEN 3B	51-336 Phase C	BFN-3-51-211-03EB/14C	GE 12IAC51A101A	SEIS_1C-5R5

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-BKR-211-03EB/011	BFN-3-BKR-211-03EB/011, 4KV SD BD 3EB/11, BKR 1842, TIE TO DIESEL GEN 3B	86-728	BFN-3-86-211-03EB/08	12HEA61C218X2	SEIS_1C-5R4-1
BFN-3-BKR-211-03EB/011	BFN-3-BKR-211-03EB/011, 4KV SD BD 3EB/11, BKR 1842, TIE TO DIESEL GEN 3B	51-728 Phase A	BFN-3-51-211-03EB/08A	GE 12IAC51A101A	SEIS_1C-5R5
BFN-3-BKR-211-03EB/011	BFN-3-BKR-211-03EB/011, 4KV SD BD 3EB/11, BKR 1842, TIE TO DIESEL GEN 3B	51-728 Phase C	BFN-3-51-211-03EB/08C	GE 12IAC51A101A	SEIS_1C-5R5
BFN-3-BKR-211-03EB/011	BFN-3-BKR-211-03EB/011, 4KV SD BD 3EB/11, BKR 1842, TIE TO DIESEL GEN 3B	86-S3B	BFN-3-86-211-03EB/07	12HEA61C219X2	SEIS_1C-5R4-1
BFN-3-BKR-211-03EB/011	BFN-3-BKR-211-03EB/011, 4KV SD BD 3EB/11, BKR 1842, TIE TO DIESEL GEN 3B	B/OTX	BFN-3-RLY-082-B/OTX	12HFA51A42H	SEIS_1C-5R3
BFN-3-BKR-211-03EB/011	BFN-3-BKR-211-03EB/011, 4KV SD BD 3EB/11, BKR 1842, TIE TO DIESEL GEN 3B	OTR	BFN-3-RLY-082-B/0TR	Square D Class 8501 Type XUD0-1200	SEIS_1C-5R1
BFN-3-BKR-211-03EB/011	BFN-3-BKR-211-03EB/011, 4KV SD BD 3EB/11, BKR 1842, TIE TO DIESEL GEN 3B	Breaker	BFN-3-BKR-211-03EB/011	Wylie-Siemens Type 5-3AK-GEH-250-1200-58	Modeled as DG fails to start
BFN-3-BKR-211-03EC/010	BFN-3-BKR-211-03EC/010, 4KV SD BD 3EC/10, BKR 1832, TIE TO DG 3C	C/ESTR	BFN-3-RLY-082-C/ESTR	Square D DO/7001	SEIS_1C-4R1
BFN-3-BKR-211-03EC/010	BFN-3-BKR-211-03EC/010, 4KV SD BD 3EC/10, BKR 1832, TIE TO DG 3C	51X1-C	BFN-3-RLY-082-2547C/1	12HGA11J51	Chatter Acceptable

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-BKR-211-03EC/010	BFN-3-BKR-211-03EC/010, 4KV SD BD 3EC/10, BKR 1832, TIE TO DG 3C	CAR	BFN-3-RLY-082-2547CCAR	12HFA51A42F	SEIS_1C-4R2
BFN-3-BKR-211-03EC/010	BFN-3-BKR-211-03EC/010, 4KV SD BD 3EC/10, BKR 1832, TIE TO DG 3C	51VX-C	BFN-3-51V-082-2547C/X	GE 12PJV11AM2A	Chatter Acceptable
BFN-3-BKR-211-03EC/010	BFN-3-BKR-211-03EC/010, 4KV SD BD 3EC/10, BKR 1832, TIE TO DG 3C	51VZ-C	BFN-3-51V-082-2547C/X	12HFA51A42F	Chatter Acceptable
BFN-3-BKR-211-03EC/010	BFN-3-BKR-211-03EC/010, 4KV SD BD 3EC/10, BKR 1832, TIE TO DG 3C	R3C	BFN-3-RLY-082-R3C	12HFA51A42H	SEIS_1C-4R3
BFN-3-BKR-211-03EC/010	BFN-3-BKR-211-03EC/010, 4KV SD BD 3EC/10, BKR 1832, TIE TO DG 3C	CASA-2	BFN-3-RLY-211-CASA-2	12HFA51A41F	SEIS_1C-4R8
BFN-3-BKR-211-03EC/010	BFN-3-BKR-211-03EC/010, 4KV SD BD 3EC/10, BKR 1832, TIE TO DG 3C	CASB-2	BFN-3-RLY-211-CASB-2	12HFA151A1F	SEIS_1C-4R8
BFN-3-BKR-211-03EC/010	BFN-3-BKR-211-03EC/010, 4KV SD BD 3EC/10, BKR 1832, TIE TO DG 3C	86-G3C-832	BFN-3-86-211-03EC/10	12HEA61A213X2	SEIS_1C-4R6
BFN-3-BKR-211-03EC/010	BFN-3-BKR-211-03EC/010, 4KV SD BD 3EC/10, BKR 1832, TIE TO DG 3C	86-338	BFN-3-86-211-03EC/12	12HEA61C218X2	SEIS_1C-4R4
BFN-3-BKR-211-03EC/010	BFN-3-BKR-211-03EC/010, 4KV SD BD 3EC/10, BKR 1832, TIE TO DG 3C	51-338 Phase A	BFN-3-51-211-03EC/12A	GE 12IAC51A101A	SEIS_1C-4R5-1

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-BKR-211-03EC/010	BFN-3-BKR-211-03EC/010, 4KV SD BD 3EC/10, BKR 1832, TIE TO DG 3C	51-338 Phase C	BFN-3-51-211-03EC/12C	GE 12IAC51A101A	SEIS_1C-4R5-1
BFN-3-BKR-211-03EC/010	BFN-3-BKR-211-03EC/010, 4KV SD BD 3EC/10, BKR 1832, TIE TO DG 3C	86-626	BFN-3-86-211-03EC/03	12HEA61C218X2	SEIS_1C-4R4
BFN-3-BKR-211-03EC/010	BFN-3-BKR-211-03EC/010, 4KV SD BD 3EC/10, BKR 1832, TIE TO DG 3C	51-626 Phase A	BFN-3-51-211-03EC/03A	GE 12IAC51A101A	SEIS_1C-4R5-1
BFN-3-BKR-211-03EC/010	BFN-3-BKR-211-03EC/010, 4KV SD BD 3EC/10, BKR 1832, TIE TO DG 3C	51-626 Phase C	BFN-3-51-211-03EC/03C	GE 12IAC51A101A	SEIS_1C-4R5-1
BFN-3-BKR-211-03EC/010	BFN-3-BKR-211-03EC/010, 4KV SD BD 3EC/10, BKR 1832, TIE TO DG 3C	86-S3C	BFN-3-86-211-03EC/11	12HEA61C219X2	SEIS_1C-4R5-2
BFN-3-BKR-211-03EC/010	BFN-3-BKR-211-03EC/010, 4KV SD BD 3EC/10, BKR 1832, TIE TO DG 3C	C/OTX	BFN-3-RLY-082-C/OTX	12HFA51A42H	SEIS_1C-4R2
BFN-3-BKR-211-03EC/010	BFN-3-BKR-211-03EC/010, 4KV SD BD 3EC/10, BKR 1832, TIE TO DG 3C	OTR	BFN-3-RLY-082-C/0TR	Square D Class 8501 Type XUD0-1200	SEIS_1C-4R1
BFN-3-BKR-211-03EC/010	BFN-3-BKR-211-03EC/010, 4KV SD BD 3EC/10, BKR 1832, TIE TO DG 3C	Breaker	BFN-3-BKR-211-03EC/010	GE Hitachi NE - 317A7502P005	Modeled as DG fails to start
BFN-3-BKR-211-03ED/010	BFN-3-BKR-211-03ED/010, 4160V SD BD 3ED/10, BKR 1836, TIE TO DIESEL GEN 3D	D/ESTR	BFN-3-RLY-082-D/ESTR	Square D DO/7001	SEIS_1C-5R1

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-BKR-211-03ED/010	BFN-3-BKR-211-03ED/010, 4160V SD BD 3ED/10, BKR 1836, TIE TO DIESEL GEN 3D	51X1-D	BFN-3-RLY-082-2547D/1	12HGA11J51	Chatter Acceptable
BFN-3-BKR-211-03ED/010	BFN-3-BKR-211-03ED/010, 4160V SD BD 3ED/10, BKR 1836, TIE TO DIESEL GEN 3D	CAR	BFN-3-RLY-082-2547DCAR	12HFA51A42F	SEIS_1C-5R2
BFN-3-BKR-211-03ED/010	BFN-3-BKR-211-03ED/010, 4160V SD BD 3ED/10, BKR 1836, TIE TO DIESEL GEN 3D	51VX-D	BFN-3-51V-082-2547D/X	GE 12PJV11AM2A	Chatter Acceptable
BFN-3-BKR-211-03ED/010	BFN-3-BKR-211-03ED/010, 4160V SD BD 3ED/10, BKR 1836, TIE TO DIESEL GEN 3D	51VZ-D	BFN-3-51V-082-2547D/Z	12HFA51A42F	Chatter Acceptable
BFN-3-BKR-211-03ED/010	BFN-3-BKR-211-03ED/010, 4160V SD BD 3ED/10, BKR 1836, TIE TO DIESEL GEN 3D	R3D	BFN-3-RLY-082-R3D	12HFA51A42H	SEIS_1C-5R3
BFN-3-BKR-211-03ED/010	BFN-3-BKR-211-03ED/010, 4160V SD BD 3ED/10, BKR 1836, TIE TO DIESEL GEN 3D	CASA-5	BFN-3-RLY-211-CASA-5	12HFA51A41F	SEIS_1C-5R8
BFN-3-BKR-211-03ED/010	BFN-3-BKR-211-03ED/010, 4160V SD BD 3ED/10, BKR 1836, TIE TO DIESEL GEN 3D	CASB-5	BFN-3-RLY-211-CASB-5	12HFA151A1F	SEIS_1C-5R8
BFN-3-BKR-211-03ED/010	BFN-3-BKR-211-03ED/010, 4160V SD BD 3ED/10, BKR 1836, TIE TO DIESEL GEN 3D	86-G3D-836	BFN-3-86-211-03ED/10	12HEA61A213X2	SEIS_1C-5R6
BFN-3-BKR-211-03ED/010	BFN-3-BKR-211-03ED/010, 4160V SD BD 3ED/10, BKR 1836, TIE TO DIESEL GEN 3D	86-342	BFN-3-86-211-03ED/08	12HEA61C218X2	SEIS_1C-5R4-1

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-BKR-211-03ED/010	BFN-3-BKR-211-03ED/010, 4160V SD BD 3ED/10, BKR 1836, TIE TO DIESEL GEN 3D	51-342 Phase A	BFN-3-51-211-03ED/08A	GE 12IAC51A101A	SEIS_1C-5R5
BFN-3-BKR-211-03ED/010	BFN-3-BKR-211-03ED/010, 4160V SD BD 3ED/10, BKR 1836, TIE TO DIESEL GEN 3D	51-342 Phase C	BFN-3-51-211-03ED/08C	GE 12IAC51A101A	SEIS_1C-5R5
BFN-3-BKR-211-03ED/010	BFN-3-BKR-211-03ED/010, 4160V SD BD 3ED/10, BKR 1836, TIE TO DIESEL GEN 3D	86-628	BFN-3-86-211-03ED/01	12HEA61C218X2	SEIS_1C-5R4-1
BFN-3-BKR-211-03ED/010	BFN-3-BKR-211-03ED/010, 4160V SD BD 3ED/10, BKR 1836, TIE TO DIESEL GEN 3D	51-628 Phase A	BFN-3-51-211-03ED/01A	GE 12IAC51A101A	SEIS_1C-5R5
BFN-3-BKR-211-03ED/010	BFN-3-BKR-211-03ED/010, 4160V SD BD 3ED/10, BKR 1836, TIE TO DIESEL GEN 3D	51-628 Phase C	BFN-3-51-211-03ED/01C	GE 12IAC51A101A	SEIS_1C-5R5
BFN-3-BKR-211-03ED/010	BFN-3-BKR-211-03ED/010, 4160V SD BD 3ED/10, BKR 1836, TIE TO DIESEL GEN 3D	86-S3D	BFN-3-86-211-03ED/09	12HEA61C219X2	SEIS_1C-5R4-2
BFN-3-BKR-211-03ED/010	BFN-3-BKR-211-03ED/010, 4160V SD BD 3ED/10, BKR 1836, TIE TO DIESEL GEN 3D	D/OTX	BFN-3-RLY-082-D/OTX	12HFA51A42H	SEIS_1C-5R3
BFN-3-BKR-211-03ED/010	BFN-3-BKR-211-03ED/010, 4160V SD BD 3ED/10, BKR 1836, TIE TO DIESEL GEN 3D	OTR	BFN-3-RLY-082-D/0TR	Square D Class 8501 Type XUD0-1200	SEIS_1C-5R1
BFN-3-BKR-211-03ED/010	BFN-3-BKR-211-03ED/010, 4160V SD BD 3ED/10, BKR 1836, TIE TO DIESEL GEN 3D	Breaker	BFN-3-BKR-211-03ED/010	Wylie-Siemens Type 5-3AK-GEH-250-1200-58	Modeled as DG fails to start

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-BKR-211-03EA/010	BFN-3-BKR-211-03EA/010, 4KV SD BD 3EA/10, NORMAL FEEDER TO 480V SD BD 3A, XFMR TS3A	50G-3EA/10	BFN-3-50G-211-03EA/010	GE PJC11A	SEIS_1C-4R7
BFN-3-BKR-211-03EA/010	BFN-3-BKR-211-03EA/010, 4KV SD BD 3EA/10, NORMAL FEEDER TO 480V SD BD 3A, XFMR TS3A	Breaker	BFN-3-BKR-211-03EA/010	Wylie-Siemens Type 5-3AK-GEH-250-1200-58	SEIS_1C-4
BFN-3-BKR-211-03EB/009	BFN-3-BKR-211-03EB/009, 4KV SD BD 3EB/9, NOR FDR FOR TRANS TS3E	50G-3EB/9C	BFN-3-50G-211-03EB/009	GE PJC11A	SEIS_1C-5R7
BFN-3-BKR-211-03EB/009	BFN-3-BKR-211-03EB/009, 4KV SD BD 3EB/9, NOR FDR FOR TRANS TS3E	Breaker	BFN-3-BKR-211-03EB/009	Wylie-Siemens Type 5-3AK-GEH-250-1200-58	SEIS_1C-5
BFN-3-BKR-211-03EC/007	BFN-3-BKR-211-03EC/007, 4KV SD BD 3EC/7, NOR FDR FOR TRANS TS3B	50G-3E7C/7	BFN-3-50G-211-03EC/007	GE PJC11A	SEIS_1C-4R7
BFN-3-BKR-211-03EC/007	BFN-3-BKR-211-03EC/007, 4KV SD BD 3EC/7, NOR FDR FOR TRANS TS3B	Breaker	BFN-3-BKR-211-03EC/007	Wylie-Siemens Type 5-3AK-GEH-250-1200-58	SEIS_1C-4
BFN-3-BKR-211-03ED/13	BFN-3-BKR-211-03ED/13, FEEDER TO 480V HVAC BD B TRANSFORMER THB	50G-3ED/13	BFN-3-50GS-211-03ED/13	GE HFC	SEIS_1C-5R7
BFN-3-BKR-211-03ED/13	BFN-3-BKR-211-03ED/13, FEEDER TO 480V HVAC BD B TRANSFORMER THB	Breaker	BFN-3-BKR-211-03ED/13	Wylie-Siemens Type 5-3AK-GEH-250-1200-58	SEIS_1C-5

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-BKR-211-03EA/004	BFN-3-BKR-211-03EA/004, 4KV SD BD 3EA/4 BKR 1726 ALT FDR FROM 4KV BUS TIE BD	86-334	BFN-3-86-211-03EA/07	GE HEA61C218	SEIS_1C-4R4
BFN-3-BKR-211-03EA/004	BFN-3-BKR-211-03EA/004, 4KV SD BD 3EA/4 BKR 1726 ALT FDR FROM 4KV BUS TIE BD	51-334 Phase A	BFN-3-51-211-03EA/07A	GE 12IAC51A101A	SEIS_1C-4R5-1
BFN-3-BKR-211-03EA/004	BFN-3-BKR-211-03EA/004, 4KV SD BD 3EA/4 BKR 1726 ALT FDR FROM 4KV BUS TIE BD	51-334 Phase C	BFN-3-51-211-03EA/07C	GE 12IAC51A101A	SEIS_1C-4R5-1
BFN-3-BKR-211-03EA/004	BFN-3-BKR-211-03EA/004, 4KV SD BD 3EA/4 BKR 1726 ALT FDR FROM 4KV BUS TIE BD	86-726	BFN-3-86-211-03EA/04	12HEA61C218X2	SEIS_1C-4R4
BFN-3-BKR-211-03EA/004	BFN-3-BKR-211-03EA/004, 4KV SD BD 3EA/4 BKR 1726 ALT FDR FROM 4KV BUS TIE BD	51-726 Phase A	BFN-3-51-211-03EA/04A	GE 12IAC51A101A	SEIS_1C-4R5-1
BFN-3-BKR-211-03EA/004	BFN-3-BKR-211-03EA/004, 4KV SD BD 3EA/4 BKR 1726 ALT FDR FROM 4KV BUS TIE BD	51-726 Phase C	BFN-3-51-211-03EA/04C	GE 12IAC51A101A	SEIS_1C-4R5-1
BFN-3-BKR-211-03EA/004	BFN-3-BKR-211-03EA/004, 4KV SD BD 3EA/4 BKR 1726 ALT FDR FROM 4KV BUS TIE BD	86-S3A	BFN-3-86-211-03EA/08	12HEA61C220X2	SEIS_1C-4R5-2
BFN-3-BKR-211-03EA/004	BFN-3-BKR-211-03EA/004, 4KV SD BD 3EA/4 BKR 1726 ALT FDR FROM 4KV BUS TIE BD	86-EAX	BFN-3-86-210-0001A	12HFA51A41F	Trips and Locks out normally open tie board Feeder Breaker. Not a chatter concern.

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-BKR-211-03EA/004	BFN-3-BKR-211-03EA/004, 4KV SD BD 3EA/4 BKR 1726 ALT FDR FROM 4KV BUS TIE BD	R-726	BFN-3-RLY-211-03EA/04C	GE 12HFA51A41H	Trips Tie Board FDR BKR Emergency use only. Not a chatter concern.
BFN-3-BKR-211-03EA/004	BFN-3-BKR-211-03EA/004, 4KV SD BD 3EA/4 BKR 1726 ALT FDR FROM 4KV BUS TIE BD	Breaker	BFN-3-BKR-211-03EA/004	Wylie-Siemens Type 5-3AK-GEH-250-1200-58	Alternate feeder breakers not modeled
BFN-3-BKR-211-03EB/008	BFN-3-BKR-211-03EB/008, 4KV SD BD 3EB/8 BKR 1728 ALT FDR FROM 4KV SD BD 3EA	86-336	BFN-3-86-211-03EB/14	12HEA61C218X2	SEIS_1C-5R4-1
BFN-3-BKR-211-03EB/008	BFN-3-BKR-211-03EB/008, 4KV SD BD 3EB/8 BKR 1728 ALT FDR FROM 4KV SD BD 3EA	51-336 Phase A	BFN-3-51-211-03EB/14A	GE 12IAC51A101A	SEIS_1C-5R5
BFN-3-BKR-211-03EB/008	BFN-3-BKR-211-03EB/008, 4KV SD BD 3EB/8 BKR 1728 ALT FDR FROM 4KV SD BD 3EA	51-336 Phase C	BFN-3-51-211-03EB/14C	GE 12IAC51A101A	SEIS_1C-5R5
BFN-3-BKR-211-03EB/008	BFN-3-BKR-211-03EB/008, 4KV SD BD 3EB/8 BKR 1728 ALT FDR FROM 4KV SD BD 3EA	86-728	BFN-3-86-211-03EB/08	12HEA61C218X2	SEIS_1C-5R4-1
BFN-3-BKR-211-03EB/008	BFN-3-BKR-211-03EB/008, 4KV SD BD 3EB/8 BKR 1728 ALT FDR FROM 4KV SD BD 3EA	51-728 Phase A	BFN-3-51-211-03EB/08A	GE 12IAC51A101A	SEIS_1C-5R5
BFN-3-BKR-211-03EB/008	BFN-3-BKR-211-03EB/008, 4KV SD BD 3EB/8 BKR 1728 ALT FDR FROM 4KV SD BD 3EA	51-728 Phase C	BFN-3-51-211-03EB/08C	GE 12IAC51A101A	SEIS_1C-5R5

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-BKR-211-03EB/008	BFN-3-BKR-211-03EB/008, 4KV SD BD 3EB/8 BKR 1728 ALT FDR FROM 4KV SD BD 3EA	86-S3B	BFN-3-86-211-03EB/07	12HEA61C219X2	SEIS_1C-5R4-1
BFN-3-BKR-211-03EB/008	BFN-3-BKR-211-03EB/008, 4KV SD BD 3EB/8 BKR 1728 ALT FDR FROM 4KV SD BD 3EA	86-EAX	BFN-3-86-210-0001A	12HFA51A41F	Trips and Locks out normally open tie board Feeder Breaker. Not a chatter concern.
BFN-3-BKR-211-03EB/008	BFN-3-BKR-211-03EB/008, 4KV SD BD 3EB/8 BKR 1728 ALT FDR FROM 4KV SD BD 3EA	R-728	BFN-3-RLY-211-03EB/08C	GE 12HFA51A41H	Trips Tie Board FDR BKR Emergency use only. Not a chatter concern.
BFN-3-BKR-211-03EB/008	BFN-3-BKR-211-03EB/008, 4KV SD BD 3EB/8 BKR 1728 ALT FDR FROM 4KV SD BD 3EA	Breaker	BFN-3-BKR-211-03EB/008	Wylie-Siemens Type 5-3AK-GEH-250-1200-58	Alternate feeder breakers not modeled
BFN-3-BKR-211-03EC/003	BFN-3-BKR-211-03EC/003, 4KV SD BD 3EC BKR 1626 ALT FDR FROM 4KV BUS TIE BD	86-338	BFN-3-86-211-03EC/12	12HEA61C218X2	SEIS_1C-4R4
BFN-3-BKR-211-03EC/003	BFN-3-BKR-211-03EC/003, 4KV SD BD 3EC BKR 1626 ALT FDR FROM 4KV BUS TIE BD	51-338 Phase A	BFN-3-51-211-03EC/12A	GE 12IAC51A101A	SEIS_1C-4R5-1
BFN-3-BKR-211-03EC/003	BFN-3-BKR-211-03EC/003, 4KV SD BD 3EC BKR 1626 ALT FDR FROM 4KV BUS TIE BD	51-338 Phase C	BFN-3-51-211-03EC/12C	GE 12IAC51A101A	SEIS_1C-4R5-1
BFN-3-BKR-211-03EC/003	BFN-3-BKR-211-03EC/003, 4KV SD BD 3EC BKR 1626 ALT FDR FROM 4KV BUS TIE BD	86-626	BFN-3-86-211-03EC/03	12HEA61C218X2	SEIS_1C-4R4

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-BKR-211-03EC/003	BFN-3-BKR-211-03EC/003, 4KV SD BD 3EC BKR 1626 ALT FDR FROM 4KV BUS TIE BD	51-626 Phase A	BFN-3-51-211-03EC/03A	GE 12IAC51A101A	SEIS_1C-4R5-1
BFN-3-BKR-211-03EC/003	BFN-3-BKR-211-03EC/003, 4KV SD BD 3EC BKR 1626 ALT FDR FROM 4KV BUS TIE BD	51-626 Phase C	BFN-3-51-211-03EC/03C	GE 12IAC51A101A	SEIS_1C-4R5-1
BFN-3-BKR-211-03EC/003	BFN-3-BKR-211-03EC/003, 4KV SD BD 3EC BKR 1626 ALT FDR FROM 4KV BUS TIE BD	86-S3C	BFN-3-86-211-03EC/011	12HEA61C219X2	SEIS_1C-4R5-2
BFN-3-BKR-211-03EC/003	BFN-3-BKR-211-03EC/003, 4KV SD BD 3EC BKR 1626 ALT FDR FROM 4KV BUS TIE BD	86-ECX	BFN-3-86-210-0001C	12HFA51A41F	Trips and Locks out normally open tie board Feeder Breaker. Not a chatter concern.
BFN-3-BKR-211-03EC/003	BFN-3-BKR-211-03EC/003, 4KV SD BD 3EC BKR 1626 ALT FDR FROM 4KV BUS TIE BD	R-626	BFN-3-RLY-211-03EC/03C	GE 12HFA51A41H	Trips Tie Board FDR BKR Emergency use only. Not a concern.
BFN-3-BKR-211-03EC/003	BFN-3-BKR-211-03EC/003, 4KV SD BD 3EC BKR 1626 ALT FDR FROM 4KV BUS TIE BD	Breaker	BFN-3-BKR-211-03EC/003	Wylie-Siemens Type 5-3AK-GEH-250-1200-58	Alternate feeder breakers not modeled
BFN-3-BKR-211-03ED/001	BFN-3-BKR-211-03ED/001, 4KV SD BD 3ED/1 BKR 1628 ALT FDR FROM 4KV BUS TIE BD	86-342	BFN-3-86-211-03ED/08	12HEA61C218X2	SEIS_1C-5R4-1
BFN-3-BKR-211-03ED/001	BFN-3-BKR-211-03ED/001, 4KV SD BD 3ED/1 BKR 1628 ALT FDR FROM 4KV BUS TIE BD	51-342 Phase A	BFN-3-51-211-03ED/08A	GE 12IAC51A101A	SEIS_1C-5R5

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-BKR-211-03ED/001	BFN-3-BKR-211-03ED/001, 4KV SD BD 3ED/1 BKR 1628 ALT FDR FROM 4KV BUS TIE BD	51-342 Phase C	BFN-3-51-211-03ED/08C	GE 12IAC51A101A	SEIS_1C-5R5
BFN-3-BKR-211-03ED/001	BFN-3-BKR-211-03ED/001, 4KV SD BD 3ED/1 BKR 1628 ALT FDR FROM 4KV BUS TIE BD	86-628	BFN-3-86-211-03ED/01	12HEA61C218X2	SEIS_1C-5R4-1
BFN-3-BKR-211-03ED/001	BFN-3-BKR-211-03ED/001, 4KV SD BD 3ED/1 BKR 1628 ALT FDR FROM 4KV BUS TIE BD	51-628 Phase A	BFN-3-51-211-03ED/01A	GE 12IAC51A101A	SEIS_1C-5R5
BFN-3-BKR-211-03ED/001	BFN-3-BKR-211-03ED/001, 4KV SD BD 3ED/1 BKR 1628 ALT FDR FROM 4KV BUS TIE BD	51-628 Phase C	BFN-3-51-211-03ED/01C	GE 12IAC51A101A	SEIS_1C-5R5
BFN-3-BKR-211-03ED/001	BFN-3-BKR-211-03ED/001, 4KV SD BD 3ED/1 BKR 1628 ALT FDR FROM 4KV BUS TIE BD	86-S3D	BFN-3-86-211-03ED/09	12HEA61C219X2	SEIS_1C-5R4-2
BFN-3-BKR-211-03ED/001	BFN-3-BKR-211-03ED/001, 4KV SD BD 3ED/1 BKR 1628 ALT FDR FROM 4KV BUS TIE BD	86-ECX	BFN-3-86-210-0001C	12HFA51A41F	Trips and Locks out normally open tie board Feeder Breaker. Not a chatter concern.
BFN-3-BKR-211-03ED/001	BFN-3-BKR-211-03ED/001, 4KV SD BD 3ED/1 BKR 1628 ALT FDR FROM 4KV BUS TIE BD	R-628	BFN-3-RLY-211-03ED/01C	GE 12HFA51A41H	Trips Tie Board FDR BKR Emergency use only. Not a chatter concern.
BFN-3-BKR-211-03ED/001	BFN-3-BKR-211-03ED/001, 4KV SD BD 3ED/1 BKR 1628 ALT FDR FROM 4KV BUS TIE BD	Breaker	BFN-3-BKR-211-03ED/001	Wylie-Siemens Type 5-3AK-GEH-250-1200-58	Alternate feeder breakers not modeled

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-PMP-023-0094	BFN-0-PMP-023-0094, RHRSW PUMP D3	27SDX	BFN-0-27-211-000D/11H	12HFA51A41F	SEIS_11-1R1-2
BFN-0-PMP-023-0094	BFN-0-PMP-023-0094, RHRSW PUMP D3	27SDY	BFN-0-27-211-000D/11M	12HFA51A41F	SEIS_11-1R1-2
BFN-0-PMP-023-0094	BFN-0-PMP-023-0094, RHRSW PUMP D3	50G	BFN-0-50G-023-0094	GE PJC-11A	SEIS_11-1R2
BFN-0-PMP-023-0094	BFN-0-PMP-023-0094, RHRSW PUMP D3	Breaker	BFN-0-BKR-023-0094	Wylie-Siemens Type 5-3AK-GEH-250-1200-58	Modeled as EECW pump fails to start
BFN-0-PMP-023-0088	BFN-0-PMP-023-0088, RHRSW PUMP B3	27SCX	BFN-0-27-211-000C/11H	12HFA51A41F	SEIS_11-1R1-1
BFN-0-PMP-023-0088	BFN-0-PMP-023-0088, RHRSW PUMP B3	27SCY	BFN-0-27-211-000C/11M	12HFA51A41F	SEIS_11-1R1-1
BFN-0-PMP-023-0088	BFN-0-PMP-023-0088, RHRSW PUMP B3	50G	BFN-0-50G-023-0088	GE PJC-11A	SEIS_11-1R2
BFN-0-PMP-023-0088	BFN-0-PMP-023-0088, RHRSW PUMP B3	Breaker	BFN-0-BKR-023-0088	Wylie-Siemens Type 5-3AK-GEH-250-1200-58	Modeled as EECW pump fails to start
BFN-0-PMP-023-0085	BFN-0-PMP-023-0085, RHRSW PUMP A3	27S3AX	BFN-3-27-211-03EA/08G	12HFA51A41F	SEIS_11-1R3

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-PMP-023-0085	BFN-0-PMP-023-0085, RHRSW PUMP A3	27S3AY	BFN-3-27-211-03EA/08K	12HFA51A41F	SEIS_11-1R3
BFN-0-PMP-023-0085	BFN-0-PMP-023-0085, RHRSW PUMP A3	50G	BFN-0-50G-023-0085	GE PJC-11A	SEIS_11-1R4
BFN-0-PMP-023-0085	BFN-0-PMP-023-0085, RHRSW PUMP A3	Breaker	BFN-0-BKR-023-0085	Wylie-Siemens Type 5-3AK-GEH-250-1200-58	Modeled as EECW pump fails to start
BFN-0-PMP-023-0091	BFN-0-PMP-023-0091, RHRSW PUMP C3	27S3BX	BFN-3-27-211-03EB/07G	12HFA51A41F	SEIS_11-1R5
BFN-0-PMP-023-0091	BFN-0-PMP-023-0091, RHRSW PUMP C3	27S3BY	BFN-3-27-211-03EB/07K	12HFA51A41F	SEIS_11-1R5
BFN-0-PMP-023-0091	BFN-0-PMP-023-0091, RHRSW PUMP C3	50G	BFN-0-50G-023-0091	GE PJC-11A	SEIS_11-1R6
BFN-0-PMP-023-0091	BFN-0-PMP-023-0091, RHRSW PUMP C3	Breaker	BFN-0-BKR-023-0091	Wylie-Siemens Type 5-3AK-GEH-250-1200-58	Modeled as EECW pump fails to start
BFN-0-FAN-030-0064	BFN-0-FAN-030-0064, DIESEL GEN RM A EXH FAN A	GA	GA*	Cardox - Clark Cat # 5U8	SEIS_17-1R1
BFN-0-FAN-030-0064	BFN-0-FAN-030-0064, DIESEL GEN RM A EXH FAN A	GATD	BFN-0-IC-039-0007	Cardox TM	SEIS_17-1R2

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-FAN-030-0065	BFN-0-FAN-030-0065, DIESEL GEN RM A EXH FAN B	GA	GA*	Cardox - Clark Cat # 5U8	SEIS_17-1R1
BFN-0-FAN-030-0065	BFN-0-FAN-030-0065, DIESEL GEN RM A EXH FAN B	GATD	BFN-0-IC-039-0007	Cardox TM	SEIS_17-1R2
BFN-0-FAN-030-0066	BFN-0-FAN-030-0066, DIESEL GEN RM B EXH FAN A	GB	GB*	Cardox - Clark Cat # 5U8	SEIS_17-1R1
BFN-0-FAN-030-0066	BFN-0-FAN-030-0066, DIESEL GEN RM B EXH FAN A	GBTD	BFN-0-IC-039-0008	Cardox TM	SEIS_17-1R2
BFN-0-FAN-030-0067	BFN-0-FAN-030-0067, DIESEL GEN RM B EXH FAN B	GB	GB*	Cardox - Clark Cat # 5U8	SEIS_17-1R1
BFN-0-FAN-030-0067	BFN-0-FAN-030-0067, DIESEL GEN RM B EXH FAN B	GBTD	BFN-0-IC-039-0008	Cardox TM	SEIS_17-1R2
BFN-0-FAN-030-0068	BFN-0-FAN-030-0068, DIESEL GEN RM C EXH FAN A	GC	GC*	Cardox - Clark Cat # 5U8	SEIS_17-1R1
BFN-0-FAN-030-0068	BFN-0-FAN-030-0068, DIESEL GEN RM C EXH FAN A	GCTD	BFN-0-IC-039-0009	Cardox TM	SEIS_17-1R2
BFN-0-FAN-030-0069	BFN-0-FAN-030-0069, DIESEL GEN RM C EXH FAN B	GC	GC*	Cardox - Clark Cat # 5U8	SEIS_17-1R1

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-FAN-030-0069	BFN-0-FAN-030-0069, DIESEL GEN RM C EXH FAN B	GCTD	BFN-0-IC-039-0009	Cardox TM	SEIS_17-1R2
BFN-0-FAN-030-0070	BFN-0-FAN-030-0070, DIESEL GEN RM D EXH FAN A	GD	GD*	Cardox - Clark Cat # 5U8	SEIS_17-1R1
BFN-0-FAN-030-0070	BFN-0-FAN-030-0070, DIESEL GEN RM D EXH FAN A	GDTD	BFN-0-IC-039-0010	Cardox TM	SEIS_17-1R2
BFN-0-FAN-030-0071	BFN-0-FAN-030-0071, DIESEL GEN RM D EXH FAN B	GD	GD*	Cardox - Clark Cat # 5U8	SEIS_17-1R1
BFN-0-FAN-030-0071	BFN-0-FAN-030-0071, DIESEL GEN RM D EXH FAN B	GDTD	BFN-0-IC-039-0010	Cardox TM	SEIS_17-1R2
BFN-0-FAN-030-0072	BFN-0-FAN-030-0072, DG AUX TRANS TDA RM EXH FAN	EA	EA*	Cardox - Clark Cat # 5U8	SEIS_17-1R1
BFN-0-FAN-030-0072	BFN-0-FAN-030-0072, DG AUX TRANS TDA RM EXH FAN	EATD	BFN-0-IC-039-0005	Cardox TM	SEIS_17-1R2
BFN-0-FAN-030-0073	BFN-0-FAN-030-0073, 480V AUX BD RM B EXH FAN DG TDB	EB	EB*	Cardox - Clark Cat # 5U8	SEIS_17-1R1
BFN-0-FAN-030-0073	BFN-0-FAN-030-0073, 480V AUX BD RM B EXH FAN DG TDB	EBTD	BFN-0-IC-039-0006	Cardox TM	SEIS_17-1R2

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-FAN-030-0230	BFN-3-FAN-030-0230, DIESEL GEN RM 3A EXH FAN A	GA	BFN-3-RLY-039-0038A	Cardox - Clark Cat # 5U8	SEIS_17-1R1
BFN-3-FAN-030-0230	BFN-3-FAN-030-0230, DIESEL GEN RM 3A EXH FAN A	GATD	BFN-3-IC-039-0038	Cardox TM	SEIS_17-1R2
BFN-3-FAN-030-0231	BFN-3-FAN-030-0231, DIESEL GEN RM 3A EXH FAN B	GA	BFN-3-RLY-039-0038A	Cardox - Clark Cat # 5U8	SEIS_17-1R1
BFN-3-FAN-030-0231	BFN-3-FAN-030-0231, DIESEL GEN RM 3A EXH FAN B	GATD	BFN-3-IC-039-0038	Cardox TM	SEIS_17-1R2
BFN-3-FAN-030-0232	BFN-3-FAN-030-0232, DIESEL GEN RM 3B EXH FAN A	GB	BFN-3-RLY-039-0039A	Cardox - Clark Cat # 5U8	SEIS_17-1R1
BFN-3-FAN-030-0232	BFN-3-FAN-030-0232, DIESEL GEN RM 3B EXH FAN A	GBTD	BFN-3-IC-039-0039	Cardox TM	SEIS_17-1R2
BFN-3-FAN-030-0233	BFN-3-FAN-030-0233, DIESEL GEN RM 3B EXH FAN B	GB	BFN-3-RLY-039-0039A	Cardox - Clark Cat # 5U8	SEIS_17-1R1
BFN-3-FAN-030-0233	BFN-3-FAN-030-0233, DIESEL GEN RM 3B EXH FAN B	GBTD	BFN-3-IC-039-0039	Cardox TM	SEIS_17-1R2
BFN-3-FAN-030-0234	BFN-3-FAN-030-0234, DIESEL GEN RM 3C EXH FAN A	GC	BFN-3-RLY-039-0040A	Cardox - Clark Cat # 5U8	SEIS_17-1R1

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-FAN-030-0234	BFN-3-FAN-030-0234, DIESEL GEN RM 3C EXH FAN A	GCTD	BFN-3-IC-039-0040	Cardox TM	SEIS_17-1R2
BFN-3-FAN-030-0235	BFN-3-FAN-030-0235, DIESEL GEN RM 3C EXH FAN B	GC	BFN-3-RLY-039-0040A	Cardox - Clark Cat # 5U8	SEIS_17-1R1
BFN-3-FAN-030-0235	BFN-3-FAN-030-0235, DIESEL GEN RM 3C EXH FAN B	GCTD	BFN-3-IC-039-0040	Cardox TM	SEIS_17-1R2
BFN-3-FAN-030-0236	BFN-3-FAN-030-0236, DIESEL GEN RM 3D EXH FAN A	GD	BFN-3-RLY-039-0041A	Cardox - Clark Cat # 5U8	SEIS_17-1R1
BFN-3-FAN-030-0236	BFN-3-FAN-030-0236, DIESEL GEN RM 3D EXH FAN A	GDTD	BFN-3-IC-039-0041	Cardox TM	SEIS_17-1R2
BFN-3-FAN-030-0237	BFN-3-FAN-030-0237, DIESEL GEN RM 3D EXH FAN B	GD	BFN-3-RLY-039-0041A	Cardox - Clark Cat # 5U8	SEIS_17-1R1
BFN-3-FAN-030-0237	BFN-3-FAN-030-0237, DIESEL GEN RM 3D EXH FAN B	GDTD	BFN-3-IC-039-0041	Cardox TM	SEIS_17-1R2
BFN-3-FAN-030-0243	BFN-3-FAN-030-0243, DIESEL AUX BD RM 3EA FAN	EA	EA*	Cardox - Clark Cat # 5U8	SEIS_17-1R1
BFN-3-FAN-030-0243	BFN-3-FAN-030-0243, DIESEL AUX BD RM 3EA FAN	EATD	BFN-3-IC-039-0036	Cardox TM	SEIS_17-1R2

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-FAN-030-0244	BFN-3-FAN-030-0244, DSL AUX. BD. RM. 3EB & PIPE & ELEC. TUNNEL	EB	EB*	Cardox - Clark Cat # 5U8	SEIS_17-1R1
BFN-3-FAN-030-0244	BFN-3-FAN-030-0244, DSL AUX. BD. RM. 3EB & PIPE & ELEC. TUNNEL	EBTD	BFN-3-IC-039-0037	Cardox TM	SEIS_17-1R2
BFN-0-FCO-031-0088	BFN-0-FCO-031-0088, UNITS 1 & 2 EL 593 AHU A	R1	R1*	Cardox - Clark Cat # 5U8	SEIS_17-1R3
BFN-0-FCO-031-0088	BFN-0-FCO-031-0088, UNITS 1 & 2 EL 593 AHU A	R1TD	BFN-1-IC-039-0015	Cardox TM	SEIS_17-1R4
BFN-0-FCO-031-0088	BFN-0-FCO-031-0088, UNITS 1 & 2 EL 593 AHU A	R2	R2*	Cardox - Clark Cat # 5U8	SEIS_17-1R3
BFN-0-FCO-031-0088	BFN-0-FCO-031-0088, UNITS 1 & 2 EL 593 AHU A	R2TD	BFN-2-IC-039-0017	Cardox TM	SEIS_17-1R4
BFN-0-FCO-031-0089	BFN-0-FCO-031-0089, UNITS 1 & 2 EL 593 AIR HANDLING UNIT 1B	R1	R1*	Cardox - Clark Cat # 5U8	SEIS_17-1R3
BFN-0-FCO-031-0089	BFN-0-FCO-031-0089, UNITS 1 & 2 EL 593 AIR HANDLING UNIT 1B	R1TD	BFN-1-IC-039-0015	Cardox TM	SEIS_17-1R4
BFN-0-FCO-031-0089	BFN-0-FCO-031-0089, UNITS 1 & 2 EL 593 AIR HANDLING UNIT 1B	R2	R2*	Cardox - Clark Cat # 5U8	SEIS_17-1R3

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-FCO-031-0089	BFN-0-FCO-031-0089, UNITS 1 & 2 EL 593 AIR HANDLING UNIT 1B	R2TD	BFN-2-IC-039-0017	Cardox TM	SEIS_17-1R4
BFN-0-AHU-031-0088	BFN-0-AHU-031-0088, U1 & U2 EL 593 AHU A	R1	R1*	Cardox - Clark Cat # 5U8	SEIS_17-1R3
BFN-0-AHU-031-0088	BFN-0-AHU-031-0088, U1 & U2 EL 593 AHU A	R1TD	BFN-1-IC-039-0015	Cardox TM	SEIS_17-1R4
BFN-0-AHU-031-0088	BFN-0-AHU-031-0088, U1 & U2 EL 593 AHU A	R2	R2*	Cardox - Clark Cat # 5U8	SEIS_17-1R3
BFN-0-AHU-031-0088	BFN-0-AHU-031-0088, U1 & U2 EL 593 AHU A	R2TD	BFN-2-IC-039-0017	Cardox TM	SEIS_17-1R4
BFN-0-AHU-031-0089	BFN-0-AHU-031-0089, U1 & U2 EL 593 AHU 1B	R1	R1*	Cardox - Clark Cat # 5U8	SEIS_17-1R3
BFN-0-AHU-031-0089	BFN-0-AHU-031-0089, U1 & U2 EL 593 AHU 1B	R1TD	BFN-1-IC-039-0015	Cardox TM	SEIS_17-1R4
BFN-0-AHU-031-0089	BFN-0-AHU-031-0089, U1 & U2 EL 593 AHU 1B	R2	R2*	Cardox - Clark Cat # 5U8	SEIS_17-1R3
BFN-0-AHU-031-0089	BFN-0-AHU-031-0089, U1 & U2 EL 593 AHU 1B	R2TD	BFN-2-IC-039-0017	Cardox TM	SEIS_17-1R4

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-0-FCO-031-0107	BFN-0-FCO-031-0107, UNIT 3 EL 593.0 AIR HANDLING UNIT 3A	CR3	CR3*	Cardox - Clark Cat # 5U8	SEIS_17-1R3
BFN-0-FCO-031-0107	BFN-0-FCO-031-0107, UNIT 3 EL 593.0 AIR HANDLING UNIT 3A	CR3TD	BFN-3-IC-039-0018	Cardox TM	SEIS_17-1R4
BFN-0-FCO-031-0107	BFN-0-FCO-031-0107, UNIT 3 EL 593.0 AIR HANDLING UNIT 3A	R3	R3*	Cardox - Clark Cat # 5U8	SEIS_17-1R3
BFN-0-FCO-031-0107	BFN-0-FCO-031-0107, UNIT 3 EL 593.0 AIR HANDLING UNIT 3A	R3TD	BFN-3-IC-039-0019	Cardox TM	SEIS_17-1R4
BFN-0-FCO-031-0108	BFN-0-FCO-031-0108, UNIT 3 EL 593.0 AIR HANDLING UNIT 3B	CR3	CR3*	Cardox - Clark Cat # 5U8	SEIS_17-1R3
BFN-0-FCO-031-0108	BFN-0-FCO-031-0108, UNIT 3 EL 593.0 AIR HANDLING UNIT 3B	CR3TD	BFN-3-IC-039-0018	Cardox TM	SEIS_17-1R4
BFN-0-FCO-031-0108	BFN-0-FCO-031-0108, UNIT 3 EL 593.0 AIR HANDLING UNIT 3B	R3	R3*	Cardox - Clark Cat # 5U8	SEIS_17-1R3
BFN-0-FCO-031-0108	BFN-0-FCO-031-0108, UNIT 3 EL 593.0 AIR HANDLING UNIT 3B	R3TD	BFN-3-IC-039-0019	Cardox TM	SEIS_17-1R4
BFN-3-AHU-031-0107	BFN-3-AHU-031-0107, UNIT 3 MECH EQUIP RM AHU 3A	CR3	CR3*	Cardox - Clark Cat # 5U8	SEIS_17-1R3

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-AHU-031-0107	BFN-3-AHU-031-0107, UNIT 3 MECH EQUIP RM AHU 3A	CR3TD	BFN-3-IC-039-0018	Cardox TM	SEIS_17-1R4
BFN-3-AHU-031-0107	BFN-3-AHU-031-0107, UNIT 3 MECH EQUIP RM AHU 3A	R3	R3*	Cardox - Clark Cat # 5U8	SEIS_17-1R3
BFN-3-AHU-031-0107	BFN-3-AHU-031-0107, UNIT 3 MECH EQUIP RM AHU 3A	R3TD	BFN-3-IC-039-0019	Cardox TM	SEIS_17-1R4
BFN-3-AHU-031-0108	BFN-3-AHU-031-0108, UNIT 3 MECH EQUIP RM AHU 3B	CR3	R3*	Cardox - Clark Cat # 5U8	SEIS_17-1R3
BFN-3-AHU-031-0108	BFN-3-AHU-031-0108, UNIT 3 MECH EQUIP RM AHU 3B	CR3TD	BFN-3-IC-039-0018	Cardox TM	SEIS_17-1R4
BFN-3-AHU-031-0108	BFN-3-AHU-031-0108, UNIT 3 MECH EQUIP RM AHU 3B	R3	R3*	Cardox - Clark Cat # 5U8	SEIS_17-1R3
BFN-3-AHU-031-0108	BFN-3-AHU-031-0108, UNIT 3 MECH EQUIP RM AHU 3B	R3TD	BFN-3-IC-039-0019	Cardox TM	SEIS_17-1R4
BFN-1-FCV-073-0002	BFN-1-FCV-073-0002, HPCI STEAM LINE INBD ISOLATION VLV	23A-K37	BFN-1-RLY-073-23A-K37	12HFA151A1F	SEIS_14-1R2-1
BFN-1-FCV-073-0002	BFN-1-FCV-073-0002, HPCI STEAM LINE INBD ISOLATION VLV	23A-K12	BFN-1-RLY-073-23A-K12	12HFA151A1F	SEIS_14-1R1-2

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-1-FCV-073-0002	BFN-1-FCV-073-0002, HPCI STEAM LINE INBD ISOLATION VLV	23A-K27	BFN-1-RLY-073-23A-K27	12HFA151A1F	SEIS_14-1R1-2
BFN-2-FCV-073-0002	BFN-2-FCV-073-0002, HPCI STEAM LINE INBD ISOLATION VLV	23A-K37	BFN-2-RLY-073-23A-K37	12HFA151A1F	SEIS_14-1R2-1
BFN-2-FCV-073-0002	BFN-2-FCV-073-0002, HPCI STEAM LINE INBD ISOLATION VLV	23A-K12	BFN-2-RLY-073-23A-K12	12HFA151A1F	SEIS_14-1R2-1
BFN-2-FCV-073-0002	BFN-2-FCV-073-0002, HPCI STEAM LINE INBD ISOLATION VLV	23A-K27	BFN-2-RLY-073-23A-K27	12HFA151A1F	SEIS_14-1R1-2
BFN-3-FCV-073-0002	BFN-3-FCV-073-0002, HPCI STEAM LINE INBD ISOLATION VLV	23A-K37	BFN-3-RLY-073-23A-K37	12HFA151A1F	SEIS_14-1R2-1
BFN-3-FCV-073-0002	BFN-3-FCV-073-0002, HPCI STEAM LINE INBD ISOLATION VLV	23A-K12A	BFN-3-RLY-073-23A-K12A	12HFA151A1F	SEIS_14-1R2-1
BFN-3-FCV-073-0002	BFN-3-FCV-073-0002, HPCI STEAM LINE INBD ISOLATION VLV	63-73-2	BFN-3-63-073-0002	12HFA151A1F	SEIS_14-1R1-2
BFN-1-FCV-073-0003	BFN-1-FCV-073-0003, HPCI STEAM LINE OUTBD ISOLATION VLV	23A-K37	BFN-1-RLY-073-23A-K37	12HFA151A1F	SEIS_14-1R2-1
BFN-1-FCV-073-0003	BFN-1-FCV-073-0003, HPCI STEAM LINE OUTBD ISOLATION VLV	23A-K12	BFN-1-RLY-073-23A-K12	12HFA151A1F	SEIS_14-1R1-2

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-1-FCV-073-0003	BFN-1-FCV-073-0003, HPCI STEAM LINE OUTBD ISOLATION VLV	23A-K27	BFN-1-RLY-073-23A-K27	12HFA151A1F	SEIS_14-1R1-2
BFN-2-FCV-073-0003	BFN-2-FCV-073-0003, HPCI STEAM LINE OUTBD ISOLATION VLV	23A-K37	BFN-2-RLY-073-23A-K37	12HFA151A1F	SEIS_14-1R2-1
BFN-2-FCV-073-0003	BFN-2-FCV-073-0003, HPCI STEAM LINE OUTBD ISOLATION VLV	23A-K12	BFN-2-RLY-073-23A-K12	12HFA151A1F	SEIS_14-1R2-1
BFN-2-FCV-073-0003	BFN-2-FCV-073-0003, HPCI STEAM LINE OUTBD ISOLATION VLV	23A-K27	BFN-2-RLY-073-23A-K27	12HFA151A1F	SEIS_14-1R1-2
BFN-3-FCV-073-0003	BFN-3-FCV-073-0003, HPCI STEAM LINE OUTBD ISOLATION VLV,	23A-K37	BFN-3-RLY-073-23A-K37	12HFA151A1F	SEIS_14-1R2-1
BFN-3-FCV-073-0003	BFN-3-FCV-073-0003, HPCI STEAM LINE OUTBD ISOLATION VLV,	23A-K12A	BFN-3-RLY-073-23A-K12A	12HFA151A1F	SEIS_14-1R2-1
BFN-3-FCV-073-0003	BFN-3-FCV-073-0003, HPCI STEAM LINE OUTBD ISOLATION VLV,	63-73-2	BFN-3-63-073-0002	12HFA151A1F	SEIS_14-1R1-2
BFN-1-FCV-073-0044	BFN-1-FCV-073-0044, HPCI SYSTEM INBD DISCH VLV	23A-K1	BFN-1-RLY-073-23A-K1	12HFA151A1F	SEIS_14-1R1-2
BFN-1-FCV-073-0044	BFN-1-FCV-073-0044, HPCI SYSTEM INBD DISCH VLV	23A-K3	BFN-1-RLY-073-23A-K3	12HFA151A1F	SEIS_14-1R1-2

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-2-FCV-073-0044	BFN-2-FCV-073-0044, HPCI SYSTEM INBD DISCH VLV	23A-K1	BFN-2-RLY-073-23A-K1	12HFA151A1F	SEIS_14-1R1-2
BFN-2-FCV-073-0044	BFN-2-FCV-073-0044, HPCI SYSTEM INBD DISCH VLV	23A-K3	BFN-2-RLY-073-23A-K3	12HFA151A1F	SEIS_14-1R1-2
BFN-3-FCV-073-0044	BFN-3-FCV-073-0044, HPCI SYSTEM INBD DISCH VLV	23A-K1	BFN-3-RLY-073-23A-K1	12HFA151A1F	SEIS_14-1R1-2
BFN-3-FCV-073-0044	BFN-3-FCV-073-0044, HPCI SYSTEM INBD DISCH VLV	23A-K3	BFN-3-RLY-073-23A-K3	12HFA151A1F	SEIS_14-1R1-2
BFN-1-PMP-074-0005	BFN-1-PMP-074-0005, RESIDUAL HEAT REMOVAL PUMP 1A	10A-K18A	BFN-1-RLY-074-10A-K18A	12HFA151A1F	SEIS_12-1R1
BFN-1-PMP-074-0016	BFN-1-PMP-074-0016, RESIDUAL HEAT REMOVAL PUMP 1C	10A-K18B	BFN-1-RLY-074-10A-K18B	12HFA151A1F	SEIS_12-1R1
BFN-1-PMP-074-0028	BFN-1-PMP-074-0028, RESIDUAL HEAT REMOVAL PUMP 1B	10A-K21A	BFN-1-RLY-074-10A-K21A	12HFA151A1F	SEIS_12-1R1
BFN-1-PMP-074-0039	BFN-1-PMP-074-0039, RESIDUAL HEAT REMOVAL PUMP 1D	10A-K21B	BFN-1-RLY-074-10A-K21B	12HFA151A1F	SEIS_12-1R1
BFN-2-PMP-074-0005	BFN-2-PMP-074-0005, RESIDUAL HEAT REMOVAL PUMP 2A	10A-K18A	BFN-2-RLY-074-10A-K18A	12HFA151A1F	SEIS_12-1R1

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-2-PMP-074-0016	BFN-2-PMP-074-0016, RESIDUAL HEAT REMOVAL PUMP 2C	10A-K18B	BFN-2-RLY-074-10A-K18B	12HFA151A1F	SEIS_12-1R1
BFN-2-PMP-074-0028	BFN-2-PMP-074-0028, RESIDUAL HEAT REMOVAL PUMP 2B	10A-K21A	BFN-2-RLY-074-10A-K21A	12HFA151A1F	SEIS_12-1R1
BFN-2-PMP-074-0039	BFN-2-PMP-074-0039, RESIDUAL HEAT REMOVAL PUMP 2D	10A-K21B	BFN-2-RLY-074-10A-K21B	12HFA151A1F	SEIS_12-1R1
BFN-3-PMP-074-0005	BFN-3-PMP-074-0005, RESIDUAL HEAT REMOVAL PUMP 3A	10A-K18A	BFN-3-RLY-074-10A-K18A	12HFA151A1F	SEIS_12-1R1
BFN-3-PMP-074-0016	BFN-3-PMP-074-0016, RESIDUAL HEAT REMOVAL PUMP 3C	10A-K18B	BFN-3-RLY-074-10A-K18B	12HFA151A1F	SEIS_12-1R1
BFN-3-PMP-074-0028	BFN-3-PMP-074-0028, RESIDUAL HEAT REMOVAL PUMP 3B	10A-K21A	BFN-3-RLY-074-10A-K21A	12HFA151A1F	SEIS_12-1R1
BFN-3-PMP-074-0039	BFN-3-PMP-074-0039, RESIDUAL HEAT REMOVAL PUMP 3D	10A-K21B	BFN-3-RLY-074-10A-K21B	12HFA151A1F	SEIS_12-1R1
BFN-1-FCV-074-0053	BFN-1-FCV-074-0053, RHR SYS I LPCI INBD INJECT VALVE	10A-K67A	BFN-1-RLY-074-10A-K67A	12HGA11A51F	SEIS_14-1R3-1
BFN-2-FCV-074-0053	BFN-2-FCV-074-0053, RHR SYS I LPCI INBD INJECT VALVE	10A-K67A	BFN-2-RLY-074-10A-K67A	12HGA11A51F	SEIS_14-1R3-1

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-FCV-074-0053	BFN-3-FCV-074-0053, RHR SYS I LPCI INBD INJECT VALVE	10A-K67A	BFN-3-RLY-074-10A-K67A	12HGA11A51F	SEIS_14-1R3-1
BFN-1-FCV-074-0067	BFN-1-FCV-074-0067, RHR SYS II LPCI INBD INJECT VALVE	10A-K66B	BFN-1-RLY-074-10A-K66B	12HGA11A51F	SEIS_14-1R3-1
BFN-2-FCV-074-0067	BFN-2-FCV-074-0067, RHR SYS II LPCI INBD INJECT VALVE	10A-K66B	BFN-1-RLY-074-10A-K66B	12HGA11A51F	SEIS_14-1R3-1
BFN-3-FCV-074-0067	BFN-3-FCV-074-0067, RHR SYS II LPCI INBD INJECT VALVE	10A-K66B	BFN-1-RLY-074-10A-K66B	12HGA11A51F	SEIS_14-1R3-1
BFN-1-PMP-075-0005	BFN-1-PMP-075-0005, CORE SPRAY PUMP 1A	14A-K12A	BFN-1-RLY-075-14A-K12A	12HFA151A1F	SEIS_11-1R7
BFN-1-PMP-075-0014	BFN-1-PMP-075-0014, CORE SPRAY PUMP 1C	14A-K14A	BFN-1-RLY-075-14A-K14A	12HFA151A1F	SEIS_11-1R7
BFN-1-PMP-075-0033	BFN-1-PMP-075-0033, CORE SPRAY PUMP 1B	14A-K12B	BFN-1-RLY-075-14A-K12B	12HFA151A1F	SEIS_11-1R7
BFN-1-PMP-075-0042	BFN-1-PMP-075-0042, CORE SPRAY PUMP 1D	14A-K14B	BFN-1-RLY-075-14A-K14B	12HFA151A1F	SEIS_11-1R7
BFN-2-PMP-075-0005	BFN-2-PMP-075-0005, CORE SPRAY PUMP 2A	14A-K12A	BFN-2-RLY-075-14A-K12A	12HFA151A1F	SEIS_11-1R7

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-2-PMP-075-0014	BFN-2-PMP-075-0014, CORE SPRAY PUMP 2C	14A-K14A	BFN-2-RLY-075-14A-K14A	12HFA151A1F	SEIS_11-1R7
BFN-2-PMP-075-0033	BFN-2-PMP-075-0033, CORE SPRAY PUMP 2B	14A-K12B	BFN-2-RLY-075-14A-K12B	12HFA151A1F	SEIS_11-1R7
BFN-2-PMP-075-0042	BFN-2-PMP-075-0042, CORE SPRAY PUMP 2D	14A-K14B	BFN-2-RLY-075-14A-K14B	12HFA151A1F	SEIS_11-1R7
BFN-3-PMP-075-0005	BFN-3-PMP-075-0005, CORE SPRAY PMP 3A, 4KV SD BD 3EA/6	14A-K12A	BFN-3-RLY-075-14A-K12A	12HFA151A1F	SEIS_11-1R7
BFN-3-PMP-075-0014	BFN-3-PMP-075-0014, CORE SPRAY PMP 3C, 4KV SD BD 3EB/5	14A-K14A	BFN-3-RLY-075-14A-K14A	12HFA151A1F	SEIS_11-1R7
BFN-3-PMP-075-0033	BFN-3-PMP-075-0033, CORE SPRAY PMP 3B, 4KV SD BD 3EC/13	14A-K12B	BFN-3-RLY-075-14A-K12B	12HFA151A1F	SEIS_11-1R7
BFN-3-PMP-075-0042	BFN-3-PMP-075-0042, CORE SPRAY PMP 3D, 4KV SD BD 3ED/II	14A-K14B	BFN-3-RLY-075-14A-K14B	12HFA51A41F	SEIS_11-1R7
BFN-1-FCV-075-0037	BFN-1-FCV-075-0037, SYSTEM 2 MIN FLOW ISOL VALVE	FS-75-49	BFN-1-FS-075-0049	Static O Ring	Not a chatter concern. Valve auto opens after chatter event.
BFN-2-FCV-075-0037	BFN-2-FCV-075-0037, SYSTEM 2 MIN FLOW ISOL VALVE	FS-75-49	BFN-2-FS-075-0049	Static O Ring	Not a chatter concern. Valve auto opens after chatter event.

Table 4.1-2: Components of Chatter Concern Requiring Functional Fragility Analysis

Associated Component UNID	Associated Component Description	Relay ID on Circuit Drawing	Relay/Breaker UNID	Component Type	Fragility Group or Disposition
BFN-3-FCV-075-0037	BFN-3-FCV-075-0037, SYSTEM 2 MIN FLOW ISOL VALVE	FS-75-49	BFN-3-FS-075-0049	Static O Ring	Not a chatter concern. Valve auto opens after chatter event.

Note: An asterisk (*) in Relay/Breaker UNID column indicates the component is part of a larger component and was not assigned an individual UNID. The relay ID on the control circuit dwg is used in this column.

4.2 Walkdown Approach

This section provides a summary of the methodology and scope of the seismic walkdowns performed for the SPRA. Walkdowns were performed by personnel with appropriate qualifications as defined in the SPID [2]. The walkdowns were performed on the non-screened components following screening described in Section 4.4.1. Walkdowns of those SSCs included on the SEL were performed as part of the development of the SEL to assess the as-installed condition of these SSCs for use in determining their seismic capacity and performing initial screening.

Walkdowns were performed in accordance with guidance in the SPID, Section 6.5, and the associated requirements in the PRA Standard [7].

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

- Unresolved Safety Issue (USI) A-46/IPEEE [21, 22, 23] – Performed seismic walkdowns and seismic evaluations following the guidelines in EPRI NP-6041-SL [24] for all three units in support of USI A-46 and subsequently for IPEEE. The seismic evaluations for Units 2 and 3 were completed in 1996, and those for Unit 1 were completed in 2005.
- NTTF 2.3 Seismic [25, 26, 27] – Performed in response to NTTF Recommendation 2.3 Seismic to identify and address degraded, nonconforming, or unanalyzed conditions, and to verify the current plant configuration with the current seismic licensing basis.
- Expedited Seismic Evaluation Process (ESEP) [28] – Performed to focus the initial industry efforts on short-term evaluations to demonstrate seismic margin through a review of a subset of the plant equipment that can be relied upon to protect the reactor core following beyond design basis seismic events, including FLEX equipment installations. This included walkdowns and calculations to demonstrate that the high confidence low probability of failure (HCLPF) seismic capacity for the ESEP subset of plant equipment exceeded the Review Level Ground Motion (RLGM). The RLGM was set to 2xSSE (0.4g) for this purpose.

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

Detailed walkdowns were performed for components that had not been previously walked down after the list was reduced by using screening approach described in Section 4.4.1. During a detailed walkdown, the caveats from the Seismic Qualification Utility Group (SQUG) Generic Information Procedure (GIP) [29] were verified and sufficient information was gathered to permit development of a fragility. This included information on anchorage, configuration, weight, dimensions, load path, and other structural information. In addition, the walkdown team focused on potential adverse seismic interaction issues, including the potential for seismically induced fire and flood and seismic II/I concerns, such as masonry block walls in the vicinity of the components.

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

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

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

4.2.1 Significant Walkdown Results and Insights

Consistent with the guidance from EPRI NP-6041-SL [24], no significant findings or adverse conditions were noted during the BFN seismic walkdowns. Observations made during the walkdowns are documented in the walkdown report [30].

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

4.2.2 Seismic Equipment List and Seismic Walkdowns Technical Adequacy

The BFN SPRA SEL development [19] and walkdowns [30] were subjected to an independent peer review against the pertinent requirements in the PRA Standard. The

SEL development and walkdowns were peer reviewed relative to Capability Category II for the full set of SRs in the PRA Standard. After completion of the subsequent independent assessment, the full set of SRs was met, and the SEL and walkdowns were determined to be acceptable for use in the SPRA.

The peer review assessment [6] and subsequent disposition of peer review findings through an independent assessment [16] are further described in Appendix A and establish that the BFN SPRA SEL and seismic walkdowns are suitable for this SPRA application.

4.3 Dynamic Analysis of Structures

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

4.3.1 Fixed-base Analysis

BFN is a firm rock site; SSI was performed for each of the major structures analyzed for the SPRA. Note that fixed-base analyses were performed as a verification step in development of the RB, DGB and IPS SSI models [31].

4.3.2 Soil-Structure Interaction (SSI) Analysis

Multi-case deterministic SSI analyses considering ground motion incoherence were performed for the RB, DGB, and IPS. The SSI between the structures and the surrounding soil medium is considered by System for Analysis for Soil-Structure Interaction (SASSI) at defined interaction nodes. Cutoff frequency for the SSI analyses was chosen to be 50 Hz, and the SSI models were sufficiently refined to transmit frequencies of at least 50 Hz through the soil/rock-foundation interface. RB and DGB SSI analyses utilized the SASSI Modified Subtraction Method (MSM). For RB SSI analysis, the nodes at all four sides of the excavated soil and the nodes within three soil layers (at EL 515 ft, EL 542.63 ft, and EL 565 ft) were considered as interaction node. A sensitivity study was performed to verify that using MSM with three intermediate interaction node layers is adequate to generate accurate In-Structure Response Spectra (ISRS) results. For the DGB, all interaction nodes attached to the excavated soil model directly underneath the DGB units were considered for analysis, while for portions of the soil associated with the RB model, the interaction nodes on the RB basement-soil interface (i.e., RB basement walls and basemat) in addition to the top layer of interaction nodes of the excavated soil were considered for analysis. A sensitivity study was performed to verify the adequacy of the interaction node selection for analysis. IPS SSI analysis utilized the SASSI Direct Method (DM), where all soil layer interfaces and excavated soil nodes are defined as interaction nodes. SSI analyses in the three spatial directions were performed simultaneously.

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

Median soil profiles were defined with hazard-compatible soil properties based on those from the BFN PSHA report [11]. The soil properties include shear-wave velocity (V_S), compression-wave velocity (V_P), corresponding damping (D_S and D_P), and unit weight. These properties and values are provided by the PSHA for a range of hazard levels. The hazard level of interest corresponding to reference earthquake used in SPRA fragility analysis for BFN is 10-5 AFE. The properties and values at AFE of 10-5 represent the hazard-consistent median soil profile for each structure based on its applicable FIRS. The soil layering profiles (i.e., layer thicknesses) for SSI analysis are refined from that of the PSHA to meet passing frequency requirements.

SSI analyses considered soil and structural property variation via use of Best Estimate (BE), Lower Bound (LB), and Upper Bound (UB) structure models and BE, LB, and UB soil models such that five analysis cases were developed: BEsoil-BEstructure, LBsoil-BEstructure, UBsoil-BEstructure, BEsoil-LBstructure, and BEsoil-UBstructure. Ground motion variability was considered via use of five independent sets of time histories (THs) for each analysis case. Soil properties for each layer of each variable soil case (BE, LB, and UB) were defined consistent with the results of the probabilistic site response analysis performed with the PSHA.

The following is a summary of the main steps.

1. Develop BE structural model. A cracking assessment of the structural model is performed at the selected hazard level of interest.
2. Develop median-centered SSI model by considering BE soil strain compatible soil properties.
3. Generate five sets of ground motions by spectrally matching five seed motions to the FIRS at the selected hazard level of interest.
4. Develop BE SSI models using the BE structural models with effective stiffness and damping consistent with 10-5 AFE hazard level ground motion.
5. Uniquely pair each set of ground motions with the five SSI models, generating a total of 25 SSI analyses.
6. Extract results from the 25 SSI analyses and generate results, including the median and 84% ISRS as well as their variability.

For each simulation, structural and soil properties were defined consistent with their response at a hazard level of interest selected via coordination with fragility and PRA analysts. This hazard level of interest was selected to be the 10-5 AFE hazard level based on insights from incremental risk quantifications, especially regarding the relative risk-significance of different acceleration intervals and individual components. A list of structures and description of relevant parameters are provided in Table 4.3-1.

4.3.3 Structure Response Models

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

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

4.3.3.1 Lumped Mass Stick Models

The only LMSM used in the BFN SPRA is the NSSS as part of the RB FEM. Consistent with the design basis reference models, matrix elements were used in some parts of the NSSS LMSM. The NSSS 3D LMSM was developed by combining the horizontal and vertical 2D NSSS models. The 3D NSSS horizontal properties were selected by assigning the 2D NSSS horizontal mass and stiffness properties to both horizontal directions of the 3D model. Similarly, the vertical properties were selected by assigning the vertical 2D NSSS model properties to the vertical direction of the 3D model. Due to the symmetrical shape of the NSSS, a single 3D LMSM adequately represents the dynamic properties of the NSSS.

The LMSM of the NSSS meets or exceeds the seven criteria listed in Section 6.3.1 of the SPID [2] as minimum requirements, which are paraphrased as follows:

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

4.3.3.2 3D Finite Element Models

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

4.3.3.2.1 Reinforced Concrete Walls and Slabs

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

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

4.3.3.2.2 Foundations

The RB and IPS foundation slabs were modeled with concrete shell elements, whereas the DGB foundations were modeled with 3D solid elements due to their significant thickness. This is because the DGB foundations house fuel tanks. The effect of the tanks inside of the DGB foundations is considered through proportionally reducing the elastic modulus of the DGB foundation material.

4.3.3.2.3 Concrete Block Walls

The concrete block walls are considered to crack before the concrete walls and, therefore, not significantly contribute to the stiffness of the structural system. Therefore, the modulus of elasticity of these walls was considered as 10% of the value for concrete where they are explicitly modelled.

4.3.3.3 Structural Damping

Material damping is considered using the guidance of Sections 3.1.2.2 of American Society of Civil Engineers (ASCE) 4-98 [34] as well as Section 3.2.2 of ASCE 4-16 [35], consistent with the damping ratios used in other SPRAs.

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

For steel structures, the median damping ratios are 2% and 4% of critical damping for response level 1 and response level 2, respectively, based on ASCE 4-98 Table 3.1-1.

The justification for the response level used is provided in the SSI model documentation of the applicable structures [31].

For the reinforced-concrete shear walls in the median models, the response levels and corresponding damping ratios were selected based on the in-plane shear stress and out-of-plane bending stress of the wall. If the average shear and/or bending stresses in the walls at any given time step exceed the stress limits provided in Section C3.2.2 of ASCE 4-16 [35], response level 2 is considered and 7% damping ratio is assigned. If the average shear and/or bending stress in the walls at any given time step do not exceed the stress limits that are provided in Section C3.2.2 of ASCE 4-16, response level 1 is considered and 4% damping ratio is assigned. The concrete stress limits for response level and damping determination are $3\sqrt{f'_C}$ for shear and $7.5\sqrt{f'_C}$ for bending.

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

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

4.3.3.4 Concrete cracking

Consistent with the methodology described in ASCE 4-16 [35], the level of concrete cracking was considered based on the state of stress identified from the SSI model. Key lateral-load-resisting concrete features at each major elevation were checked for cracking through evaluation of in-plane shear stress. This assessment considered only seismic loads and did not consider gravity load or dead load. Therefore, no cracking assessment for bending stress was considered. The maximum (over the time series) average shear stresses at the base of the lateral load resisting systems were evaluated to calculate the BE demands.

The BE (median) stiffnesses of concrete structures were determined to be consistent with consistent with the stress state in the structure. This was accomplished by verifying that the stress state in the main load carrying elements (i.e., concrete shear walls) is consistent with the expected response level, as documented for each structure in its corresponding SSI model documentation. Determination of the effective stiffness of the reinforced-concrete members follows the guidance of ASCE 43-05 [36]. The adjustment of the stiffness is achieved by changing the cross-section properties (i.e., thicknesses) rather than the elastic and shear moduli. The changes in cross-section thicknesses are applied to the specific direction that is cracked (i.e., membrane vs. bending). The reduced section thicknesses were not considered in mass calculation and were only used in stiffness calculations.

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

Structure	Foundation Condition	Type of Model	Analysis Method	Comments/Other Information
Reactor Building (RB)	Rock	3D FEM	Multi-case deterministic SSI	Shear-wave velocity ≈7,000 ft/sec at foundation level; SSI analysis performed with incoherence, 5 soil-structure cases used (BE, UB, LB cases for structure and soil), 5 THs for each case.
Diesel Generator Building (DGB)	Compacted earth backfill (12 ft) over crushed stone backfill (38 ft)	3D FEM	Multi-case deterministic SSI	Average shear-wave velocity for the structural backfill ≈740 ft/sec for top ~50 ft. below grade and ~7,000 ft/sec below it; SSI analysis performed with incoherence, 5 soil-structure cases used (BE, UB, LB cases for structure and soil), 5 THs for each case.
Intake Pumping Station (IPS)	Rock	3D FEM	Multi-case deterministic SSI	Shear-wave velocity ≈7,000 ft/sec at foundation level; SSI analysis performed with incoherence, 5 soil-structure cases used (BE, UB, LB cases for structure and soil), 5 THs for each case.

4.3.3.5 Structural Impact between Reactor Building and Turbine Building

An investigation to evaluate the effect of the impact between Turbine Building (TB) Unit 1 and RB Unit 1 through nonlinear building impact analysis was performed using impact elements. The analyses were not intended to provide refined results for local responses, but rather to provide an indication of the effects of such impact on general response of interest. The results show that the effect of the impact on the ISRS results in the RB is negligible. This is because the impacts occur at slow velocities and that the nature of the impacts is not impulsive. In other words, the TB does not bounce and does not apply an impulse load on RB; rather, it gets pushed by the RB at the onset of impact. On the other hand, since the TB foundation is very flexible and has very low stiffness, the impact between RB and TB significantly increases the acceleration response of the TB at even low frequencies since it gets pushed by the RB upon impact. However, the fragility of the RB structure considered the effects of the RB-TB impact. Detailed discussion is documented in [31].

4.3.4 Seismic Structure Response Analysis Technical Adequacy

The BFN SPRA Seismic Structure Response and Soil-Structure Interaction Analysis [31] were subjected to an independent peer review against the pertinent requirements in the PRA Standard. The seismic structure response and SSI were peer reviewed relative to Capability Category II for the full set of requirements in the PRA Standard. After completion of the subsequent independent assessment, the full set of requirements was met, and the seismic structure response and SSI were determined to be acceptable for use in the SPRA.

The peer review assessment [6] and subsequent disposition of peer review findings through an independent assessment [16] are further described in Appendix A.

4.4 SSC Fragility Analysis

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

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

4.4.1 SSC Screening Approach

The BFN SEL, consisting of approximately 6,800 components combined for all three units, was reviewed, analyzed, and then reduced to about 1,600 components prior to walkdowns. The process of reducing the SEL is an iterative and multi-step process as summarized below.

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

Components that are judged inherently rugged were also screened out from needing a walkdown. These items included check valves, manual valves, throttle valves, small safety valves, small relief valves, solenoid valves, temperature elements, hand switches, small in-line strainers, and small in-line filters. These components are driven by the system they are mounted on as they are typically more rugged. Passive valves are small, lightweight, robust, and are typically mounted in line with piping. They do not need to change state during or after an event and have no external vulnerabilities. While the failure of one of these valves can contribute to the results of the SPRA, they are bound by the fragility of the distribution system to which they are attached. No fragility value is specifically developed for passive valves, but fragility for piping is developed. Piping is walked by as part of the distribution system walkdown. This same methodology applies to filters. Dampers are made of robust steel and are typically thick in gauge compared to the duct system to which they are mounted. While they may have to change state after an earthquake, they do not need to change state during the seismic event. As was the case with passive valves, the fragility of the damper is driven by the duct system to which they are mounted. While duct systems are walked down as part of the distribution system walkdown, it is understood that, in general, the failure mode of ducting is usually the supports of that duct. Ducts are either designed to handle tornado vacuum loads, which create more stress in the duct than earthquake loads, or are protected by tornado dampers. The dampers that were not in-line dampers were part of the SEL list. These include the fire dampers in the DG fan room, exhaust dampers in the DGB, the shutdown board room fire dampers in the Unit 1 and 2 RB, and fire dampers in Control Bay (CB) air conditioning room. These dampers were checked for interaction concern during the walkdowns.

The seismic walkdown team reviewed a sample of rugged components on the SEL and did not find any seismic interaction concerns that will invalidate rugged consideration. These components were removed from the SEL by the SPRA systems analysts. Other components were considered to be less rugged but would still have sufficient capacity such that their failures would be unlikely to contribute significantly to the SCDF/SLERF in a SPRA. These components are retained on the SEL. The rugged components retained on the SEL were assigned high HCLPF capacity.

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

4.4.2 SSC Fragility Analysis Methodology

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

> Consistent with the requirements in the PRA Standard, the fragility analysis for the selected SSCs is based on the methodology in EPRI guidelines. The strategy for developing the fragilities for the complete set of SSCs on the SPRA SEL follows the recommendations of EPRI NP-6041-SL [24], EPRI TR-103959 [37], EPRI TR-1019200 [38] and EPRI 3002000709 [18], EPRI 3002012994 [39], and proceeds progressively from using experienced-based capacities to component-specific evaluations. Regardless of the method, the development of fragility estimates uses plant-specific information based on SSC conditions, as confirmed through detailed walkdowns.
>
> Components are first binned into equipment classes according to SQUG classes [29] and then grouped according to similarity and location. Representative samples in each equipment group are then evaluated to obtain fragility estimates for all the items in the group.

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

For the first quantification, site-specific representative fragilities (referred to as 'representative' throughout) were typically developed by scaling from prior work performed for resolution of USI A-46, the Seismic IPEEE at BFN, and existing design basis calculations to account for available margins in the design. This is the margin between allowable values associated with design requirements and values associated with HCLPF evaluations. These margins were used to develop a Safety Factor, which is anchored to the PGA of the 1E-05 UHRS to estimate a HCLPF fragility value. The generic values of aleatory variability and epistemic uncertainty from the SPID were applied to the HCLPF to obtain the median fragility value.

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

Fragilities for the third quantification were developed for the dominant risk contributors (components with F-V greater than 5E-03) as identified during the second SPRA quantification. When beneficial, the fragilities for the final quantification were computed using the Separation of Variable (SoV) approach, where the median capacity and the associated variabilities are calculated rigorously, and then the HCLPF capacity is back-calculated using the median capacity and the variabilities. The SoV approach provides more realistic fragilities.

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

Subsequent to the peer review, additional quantifications were performed to further refine the SPRA model and to respond to peer review findings. These quantifications are described in Section 5 of this report. To support these quantifications, additional refined fragilities were developed using either the CDFM or SoV approach as appropriate. Overall, fragilities for three structures, 124 electrical and mechanical components, 29 block walls and all relays were refined following the Hybrid method, and 11 electrical components, 23 block walls and 20 relays were refined following the SoV approach. The representative fragilities are retained for the SSCs, block walls, and relays that are determined to be non-dominant risk contributors. In some cases, refined fragilities were provided for certain SSCs for use in various sensitivity studies. These refined fragilities were developed based on estimates and maximum potential improvements to determine the impact and benefit of developing more detailed fragilities for these items based on the results of the sensitivity studies. No components were screened out based on capacity, other than the inherently rugged components.

4.4.3 SSC Fragility Analysis Results and Insights

The final set of fragilities for the risk-important contributors to SCDF and SLERF are summarized in Section 5. Refer to Tables 5.4-4 through 5.4-6 for SCDF and Tables 5.5-4 and 5.5-6 for SLERF. Detailed (SoV) calculations have generally been performed for the highest risk-significant SSCs, as well as for selected other components.

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

According to Section 6.4.1, EPRI SPID,

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

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

4.4.4 SSC Fragility Analysis Technical Adequacy

The BFN SPRA SSC Fragility Analysis [40] was subjected to an independent peer review against the pertinent requirements in the PRA Standard. The SSC fragility analysis was peer reviewed relative to Capability Category II for the full set of SRs in the PRA Standard [7]. After completion of the subsequent independent assessment [16], the full set of SRs were met, and the SSC fragility analysis was determined to be acceptable for use in the SPRA.

The peer review assessment [6], and subsequent disposition and closure of peer review findings through an independent assessment [16], are further described in Appendix A.

5.0 Plant Seismic Logic Model

The seismic plant response analysis models the various combinations of structural, equipment, and human failures given the occurrence of a seismic event that could initiate and propagate a seismic core damage or large early release sequence. This model was 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 BFN seismic response model was developed by starting with the BFN internal events at-power Level 1/Level 2 PRA model of record as of February 2018 [62], and then adapting the model in accordance with guidance in the SPID and the PRA Standard. This includes the addition of seismic initiating events (IEs) based on the plant-specific seismic hazard curve and seismic fragility-related basic events to the appropriate portions of the IEPRA, eliminating some parts of the internal events model that do not apply, and adjusting the IEPRA model HRA to account for response during and following a seismic event. This modeling approach leaves the IEPRA system logic intact while incorporating the necessary additions required for the SPRA.

The BFN SPRA model was developed using the EPRI Risk and Reliability Workstation software suite (CAFTA, FRANX, HRA Calculator, ACUBE, SYSIMP and UNCERT). Both random and seismic-induced failures of modeled SSCs were included. Seismic-induced fire and flooding were also evaluated.

5.1.1 Seismic Initiating Event

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

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

The OBE, the strongest earthquake at which the plant is designed to be able to continue normal operation, is defined as 0.1g. Below 0.1g, no significant seismic impacts are expected. The SSE is defined as an acceleration of 0.2g. The plant is seismically designed such that safety-related equipment should not fail given an SSE.

Table 5.1-1: Seismic Hazard Bins

Seismic Bin	Lower Bound (g)	Upper Bound (g)	Bin Mean PGA (g)	Bin Mean Frequency (1/y)	Notes
%G01	0.1	0.2	0.14	2.68E-04	OBE to SSE
%G02	0.2	0.3	0.25	5.56E-05	SSE to 0.3g RLE
%G03	0.3	0.6	0.42	2.98E-05	
%G04	0.6	0.7	0.65	2.17E-06	
%G05	0.7	0.9	0.79	2.19E-06	
%G06	0.9	1.5	1.16	1.82E-06	
%G07	1.5	1.7	1.60	1.80E-07	
%G08	1.7	3.0	2.26	3.32E-07	
%G09	3.0		3.3	5.96E-08	Unbounded bin
				Total=3.6E-04	

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

5.1.2 Accident Sequences

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

The SIET top events include the recommended minimum set of IEs listed in NUREG/CR-4840 [41] except for the initial status of the power conversion system. No credit is taken for non-safety-related equipment such as the power conversion system in the BFN SPRA base case.

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

5.1.3 Loss of Offsite Power

The fragility of seismically induced LOOP resulting from switchyard or grid failures was obtained from Table 6-1 in NUREG/CR-6544 [42]. Seismic-induced LOOP is predicted to occur with a median ground acceleration of 0.3g. The predicted failure mode is failure of ceramic insulators in the switchyard. Use of this fragility for seismically induced

LOOP is a standard industry practice for plants in the eastern portion of the U.S. The path for transmission of offsite power to safety-related equipment and non-safety-related equipment within the plant was considered to be governed by the fragility for seismically induced offsite power, including any paths through the TB. Note that seismically induced LOOP is assumed to fail both switchyards (complete seismic correlation). The SPRA takes no credit for recovery of offsite power.

5.1.4 Very Small LOCA (VSLOCA)

SPRAs need to consider whether a coincident VSLOCA needs to be modeled for other SIET sequences. For other LOCA sequences, which are small LOCA (SLOCA), medium LOCA (MLOCA), large LOCA (LLOCA) and interfacing system LOCA (ISLOCA), the addition of a coincident VSLOCA would have no impact because the other LOCAs modeled are already larger than a VSLOCA. Also, the direct core damage events modeled are not impacted by a VSLOCA because they are assumed to go directly to core damage and early release. Inclusion of a coincident VSLOCA might potentially impact accident progression and success for SIET sequences of general transient (GTRAN), break inside containment, and break outside containment. However, the plant-specific fragility analysis determined that the seismic fragility for the VSLOCA was high (A_m=3.68 g).

5.1.5 SLERF Analysis

The seismic Level 2 PRA analysis includes an accident event progression following core melt that is similar to the event progression initiated by an internal events initiator. The BFN IEPRA developed an SLERF model consisting of containment event trees (CETs) and supporting fault trees. The SPRA used the SLERF model and incorporated the impact of seismic events into it.

The process of performing the containment analysis begins with an evaluation of the BFN SPRA Level 1 sequences. These sequences are categorized in terms of the type of challenge to containment posed by each sequence and the operability of systems that could mitigate these effects. These Plant Damage States (PDSs) are used to assist in the linking of seismic Level 1 sequences to the appropriate SLERF sequences. While each seismic Level 1 accident sequence is explicitly treated in the CAFTA computer model of the BFN plant, the Level 1 sequence logic is transferred into the CETs to take advantage of the similarities in accident challenges from the Level 1 analysis and to streamline the quantification of the core melt progression CETs. The PDS grouping and the CETs identify the general course of the accident sequence, including which systems are operating and the specific phenomena that may occur.

Development of the SLERF model for the seismic sequences was performed in the same manner as for the IEPRA. In the IEPRA, each sequence in the ET that results in core damage is sufficiently subdivided to indicate the type of event, the state of the primary system, and the state of containment protection systems. Each Level 1 end state has a designated PDS. The state of these PDSs determines which CET the sequence is input into in the SLERF analysis. For sequences that can result in a large early release, the SLERF or the Conditional Large Early Release Probability (CLERP) can be determined.

5.1.6 Summary of Resulting Correlated Component Groupings

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

5.1.7 Summary of HRA methodology

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

The list of post-initiator human actions for the internal events model is the starting point of the seismic HRA, and all existing HRAs are analyzed for modification due to seismic effects. The HFEs associated with the existing accident sequence models were retained in the SPRA model. The model was also examined for any potential human actions unique to the seismic analysis, and any new operator actions identified were added to the SPRA. Any new operator actions added to the seismic model are discussed further in the SPRA HRA Notebook [43].

Since the potential earthquakes examined vary in magnitude, as does the on-site acceleration, the level of plant damage varies accordingly due to the impacts of the different seismic events. Post-initiator HFEs retained in the SPRA model were evaluated for seismic impacts. The degree of impact is dependent on the seismic acceleration level. The seismic impacts on every post-initiator HFE in the SPRA models were accounted for by the HFE-specific performance shaping factors and selected minimal values that increase with acceleration as a function of the PDS. Following the EPRI SPRA guideline [18], the seismically adjusted HFEs use the internal events HFE nomenclature, with a suffix of "_Sn," where n ranges from 1 to 4; i.e., four separate seismic acceleration ranges were evaluated for varying seismic impacts. The SPRA HRA Notebook discusses which HFE bins correspond to which seismic acceleration levels. For bin S4 (which includes the highest acceleration seismic initiators), it was conservatively assumed that all post-trip actions are set to failed (1.0).

The use of the same method from the internal events model for the HRA dependency analysis is valid for the SPRA HRA. The SPRA HRA Notebook discusses the method used to assess HFE dependency. The SPRA Quantification Notebook [44] also provides details of how the HRA dependency analysis was performed.

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

5.1.8 Seismic-Fire

Seismic-fire interaction events have the potential to contribute significantly to core damage or large early release. The guidelines in Appendix G of EPRI 3002000709 [18] were followed in the identification and assessment of potential seismic-fire interaction events. That effort included an assessment of fire ignition sources categorized as medium or higher and additional sources identified in the IPEEE and Final Safety Analysis Report (FSAR). The seismic-fire interaction sources and the affected SSCs were walked down by a team of fragility and PRA engineers. The walkdown findings are documented in the walkdown report [30].

Seismic-fire interaction event identification and assessment is included in the BFN SPRA Seismic-Fire Interaction evaluation [46]. This evaluation was identified as a Best Practice by the peer review team. The implementation of the EPRI 3002012980 [47] process for seismic-fire was comprehensive and complete. Every fire source was considered and clearly dispositioned. Qualitative screening was carefully applied. Quantitative screening was done correctly and clearly. Everything was documented in a clear, comprehensive, and traceable manner.

The results of the evaluation indicated no seismically induced fire events need to be included in the SPRA.

5.1.9 Seismic-Flood

Seismic-flood interaction events have the potential to contribute significantly to core damage or large early release. A two-step process was used to identify such events at BFN. The first step was to review internal flood scenarios modeled in the internal flooding portion of the IEPRA [48]. All scenarios from the IEPRA were identified and were subject to further evaluation, including the scenarios that were screened out in the internal events model. The screening in the IEPRA is based on the frequency (SCDF or SLERF), which is the product of the pipe-break frequency and the conditional core damage probability (CCDP) or CLERP. For a seismic event, the pipe-break frequencies directly depend on the occurrence frequency of a certain level of earthquake and the pipe fragility. The flooding scenarios that were screened out due to the very low internal event pipe-break frequency may have considerably high CCDP or CLERP and become seismically risk-significant in combination with the potential seismic failure of other equipment. Therefore, all internal flooding scenarios, screened or not, are included in the seismic-flood interaction evaluation. Second, the scenario with the highest CCDP was then chosen as the seismic flooding scenario for a flooding source that may be from various piping or tanks. The seismic-flood interaction sources and the affected SSCs were walked down by a team of fragility and PRA engineers. The walkdown findings are documented in the walkdown report [30].

There are three types of scenarios in the IEPRA internal flooding analysis: spray, flood, and major flood. The major flood scenario typically has the highest failure probability of the three and is kept for the seismic-flood interaction model. The seismic flooding model is built in FRANX, and no operator flooding recovery actions are credited for the seismic-induced flooding. Finally, the seismic-flooding model was injected into the CAFTA fault tree via the FRANX tool XINIT for the BFN SPRA quantification.

5.2 SPRA Plant Seismic Logic Model Technical Adequacy

The BFN SPRA seismic plant response methodology and analysis were subjected to an independent peer review against the pertinent requirements in the PRA Standard [7]. The seismic plant response methodology and analysis were peer reviewed relative to Capability Category II for the full set of SRs in the PRA Standard. After completion of the subsequent independent assessment, the full set of SRs was met, and the seismic plant response methodology and analysis were determined acceptable for use in the SPRA.

The peer review assessment [6] and subsequent disposition of peer review findings through an independent closure peer review assessment [16] are further described in Appendix A.

5.3 Seismic Risk Quantification

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

5.3.1 SPRA Quantification Methodology

Several ACCESS tables within FRANX were used to define the seismic hazard bins, assign seismic fragilities to basic events with the logic model, calculate fragilities associated with each of the seismic hazard bins, and assign a Human Error Probability (HEP) by seismic bin for each HFE. Note that the seismic HRA module with FRANX 4.2 was not used to determine the HEPs because that module is presently inconsistent with seismic HRA guidelines presented in the EPRI report [18]. The module was developed based on an earlier draft of that report.

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

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

(2) Identify fragilities and HEPs to be refined.

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

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

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

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

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

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

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

(10) Perform uncertainty analysis (UNCERT).

(11) Evaluate sensitivity cases.

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

5.3.2 SPRA Model and Quantification Assumptions

Hazard/fragilities/structures analyses assumptions:

1. Refer to Section 3 of this submittal for a discussion of assumptions and uncertainties associated with the hazard analysis.
2. Refer to Section 4 of this submittal for a discussion of assumptions and uncertainties associated with the fragilities/structures analyses.

Key plant response modeling assumptions:

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

5.4 SCDF Results

5.4.1 Overall SCDF

The SPRA shows that the point estimate SCDF is 6.30E-06 per reactor year (/ry) for Unit 1, 6.40E-06 /ry for Unit 2, and 7.13E-06 /ry for Unit 3.

5.4.2 SCDF as a Function of Hazard Interval

A summary of the SCDF results for each seismic hazard interval is presented in Table 5.4-1 for Unit 1 SCDF, Table 5.4-2 for Unit 2 SCDF, and Table 5.4-3 for Unit 3 SCDF.

Table 5.4-1: Unit 1 SCDF Contribution by Initiating Event

Truncation	Scenario	Description	Earthquake Frequency	CCDP	SCDF	Percent Contribution
1.0E-12	%G01	Seismic Initiating Event (0.1g to <0.2g)	2.68E-04	7.9E-04	2.11E-07	3.4%
1.0E-12	%G02	Seismic Initiating Event (0.2g to <0.3g)	5.56E-05	9.0E-03	5.03E-07	8.0%
1.0E-11	%G03	Seismic Initiating Event (0.3g to <0.6g)	2.98E-05	4.8E-02	1.44E-06	22.8%
1.0E-11	%G04	Seismic Initiating Event (0.6g to <0.7g)	2.17E-06	3.3E-01	7.15E-07	11.3%
3.0E-10	%G05	Seismic Initiating Event (0.7g to <0.9g)	2.19E-06	5.9E-01	1.29E-06	20.5%
3.0E-08	%G06	Seismic Initiating Event (0.9g to <1.5g)	1.82E-06	8.9E-01	1.63E-06	25.8%
3.0E-08	%G07	Seismic Initiating Event (1.5g to <1.7g)	1.80E-07	9.0E-01	1.62E-07	2.6%
2.0E-08	%G08	Seismic Initiating Event (1.7g to <3g)	3.32E-07	9.1E-01	3.03E-07	4.8%
2.0E-08	%G09	Seismic Initiating Event (>3g)	5.96E-08	9.2E-01	5.48E-08	0.9%
				Total SCDF=	**6.30E-06**	

Table 5.4-2: Unit 2 SCDF Contribution by Initiating Event

Truncation	Scenario	Description	Earthquake Frequency	CCDP	SCDF	Percent Contribution
1.0E-12	%G01	Seismic Initiating Event (0.1g to <0.2g)	2.68E-04	8.4E-04	2.26E-07	3.5%
1.0E-12	%G02	Seismic Initiating Event (0.2g to <0.3g)	5.56E-05	9.7E-03	5.42E-07	8.5%
1.0E-11	%G03	Seismic Initiating Event (0.3g to <0.6g)	2.98E-05	5.1E-02	1.52E-06	23.7%
2.0E-11	%G04	Seismic Initiating Event (0.6g to <0.7g)	2.17E-06	3.3E-01	7.10E-07	11.1%
4.0E-10	%G05	Seismic Initiating Event (0.7g to <0.9g)	2.19E-06	5.7E-01	1.26E-06	19.6%
3.0E-08	%G06	Seismic Initiating Event (0.9g to <1.5g)	1.82E-06	8.9E-01	1.63E-06	25.4%
3.0E-08	%G07	Seismic Initiating Event (1.5g to <1.7g)	1.80E-07	9.0E-01	1.62E-07	2.5%
2.0E-08	%G08	Seismic Initiating Event (1.7g to <3g)	3.32E-07	9.1E-01	3.03E-07	4.7%
2.0E-08	%G09	Seismic Initiating Event (>3g)	5.96E-08	9.2E-01	5.48E-08	0.9%
				Total SCDF=	**6.40E-06**	

Table 5.4-3: Unit 3 SCDF Contribution by Initiating Event

Truncation	Scenario	Description	Earthquake Frequency	CCDP	SCDF	Percent Contribution
1.0E-12	%G01	Seismic Initiating Event (0.1g to <0.2g)	2.68E-04	9.8E-04	2.62E-07	4.1%
1.0E-12	%G02	Seismic Initiating Event (0.2g to <0.3g)	5.56E-05	1.0E-02	5.74E-07	9.0%
1.0E-11	%G03	Seismic Initiating Event (0.3g to <0.6g)	2.98E-05	6.2E-02	1.85E-06	28.9%
2.0E-11	%G04	Seismic Initiating Event (0.6g to <0.7g)	2.17E-06	3.9E-01	8.55E-07	13.4%
7.0E-07	%G05	Seismic Initiating Event (0.7g to <0.9g)	2.19E-06	6.5E-01	1.42E-06	22.2%
4.0E-08	%G06	Seismic Initiating Event (0.9g to <1.5g)	1.82E-06	9.0E-01	1.64E-06	25.7%
4.0E-08	%G07	Seismic Initiating Event (1.5g to <1.7g)	1.80E-07	9.0E-01	1.62E-07	2.5%
2.0E-08	%G08	Seismic Initiating Event (1.7g to <3g)	3.32E-07	9.1E-01	3.03E-07	4.7%
2.0E-08	%G09	Seismic Initiating Event (>3g)	5.96E-08	9.2E-01	5.48E-08	0.9%
				Total SCDF=	**7.13E-06**	

5.4.3 Fragility Group Importance for SCDF

The SSCs with the most significant seismic failure contributions to SCDF for Unit 1 are listed in Table 5.4-4, sorted by F-V. The seismic fragilities for each of the significant contributors are also provided in Table 5.4-4, along with the corresponding limiting seismic failure mode and method of fragility calculation. The corresponding measures for Unit 2 and Unit 3 are presented in Tables 5.4-5 and 5.4-6.

Table 5.4-4: Unit 1 SCDF Fragility Group Importance Measures Ranked by F-V

Fragility Group	Description	F-V	A_m (g)	βu	βr	Failure Mode	Fragility Method
SEIS_LOOP	LOOP (Loss of Offsite Power)	0.885	0.30	0.45	0.30	Ceramic insulators	Table 6-1 NUREG/CR-6544
SEIS_12-1P-1	RHRSW pumps based on pipe frag (Pipe Calc)	0.056	2.45	0.61	0.47	Soil (Buried Piping) Failure	SoV
SEIS_12-1P-2	Emergency Equipment Cooling Water (EECW) pumps based on pipe frag calc	0.053	2.48	0.60	0.48	Soil (Buried Piping) Failure	SoV
SEIS_2-1-1	Unit Batteries (Fragility Group 15-03)	0.047	1.32	0.38	0.24	Anchorage	Representative†
SEIS_1B-2-1a	480v BD 219 A, B (U1, U2) (Fragility Group 01-05-01) Anchorage Failure	0.033	1.95	0.56	0.22	Anchorage	SoV
SEIS_1C-4R8	3EA,3EC SDBD Common Accident Signal Relays (CASA)	0.022	0.63	0.32	0.24	Functionality	CDFM
SEIS_1B-2-1f	480v BD 219 A, B (U1, U2) (Fragility Group 01-05-01) Functional Failure	0.022	2.61	0.64	0.24	Functionality	SoV
SEIS_12-1b	RHRSW pumps and EECW Alternate (Fragility Group 06-03)	0.017	1.45	0.32	0.24	Anchorage	CDFM
SEIS_5-2B	Initiation relays and panels	0.016	0.97	0.32	0.24	Anchorage	CDFM
SEIS_3-2	Batt CH 248-1 (Fragility Group 16-06)	0.015	1.47	0.32	0.24	Functionality	CDFM
SEIS_11-1R3	Relay group 3 for group SEIS_11-1 (EECW Pp A3 UV device)	0.013	0.61	0.32	0.24	Functionality	CDFM
SEIS_11-1R1-1	Relay group 1-1 for group SEIS_11-1 (EECW Pp B3 UV device)	0.011	1.31	0.37	0.29	Functionality	SoV
SEIS_BLD-IPS	Intake Pumping Station	0.010	1.90	0.26	0.24	Structural Analysis	CDFM
SEIS_1C-1W28SD	Wall 28 falls towards 480v SD BD 1A and 1B	0.010	2.18	0.54	0.27	Block Wall Failure	SoV

Table 5.4-4: Unit 1 SCDF Fragility Group Importance Measures Ranked by F-V

Fragility Group	Description	F-V	A_m (g)	βu	βr	Failure Mode	Fragility Method
SEIS_4-5	4160-480V TRANSFORMER, CONTROL BAY (Fragility Group 4-12)	0.008	0.60	0.38	0.24	Functionality	Representative†
SEIS_2-1W8	Battery and Wall 8 or 47 interface (Block Wall Group 5)	0.007	2.12	0.41	0.29	Block Wall Failure	SoV
SEIS_11-1R2	Relay group 2 for group SEIS_11-1 (EECW Pp B3&D3 OC device)	0.007	1.29	0.38	0.24	Functionality	CDFM
SEIS_1B-1	480V BD 1A, 2A, 3A, 1B, 2B, 3B (Fragility Group 02-01)	0.007	1.70	0.32	0.24	Functionality	CDFM
SEIS_1C-3R6	B,D SDBD 480 Transformer Trip relays (50G)	0.007	1.32	0.32	0.24	Functionality	CDFM
SEIS_11-1R4	Relay group 4 for group SEIS_11-1 (EECW Pp A3 OC device)	0.005	0.94	0.51	0.27	Functionality	SoV
SEIS_1C-4R4	3EA,3EC SDBD Lockout relays (86)	0.005	0.94	0.32	0.24	Functionality	CDFM
SEIS_1C-4R5-2	3EA,3EC SDBD Lockout relays (86)	0.005	0.94	0.32	0.24	Functionality	CDFM
SEIS_HINST	Seismic failure of Main Control Room instrumentation	0.005	1.96	0.24	0.32	Functionality	CDFM

†See Section 5.7 for additional discussion on these representative fragilities

Table 5.4-5: Unit 2 SCDF Importance Measures Ranked by F-V

Fragility Group	Description	F-V	A_m (g)	βu	βr	Failure Mode	Fragility Method
SEIS_LOOP	LOOP (Loss of Offsite Power)	0.885	0.3	0.45	0.30	Ceramic insulators	Table 6-1 NUREG/CR-6544
SEIS_12-1P-1	RHRSW pumps based on pipe frag (Pipe Calc)	0.056	2.45	0.61	0.47	Soil (Buried Piping) Failure	SoV
SEIS_12-1P-2	EECW pumps based on pipe frag calc	0.053	2.48	0.60	0.48	Soil (Buried Piping) Failure	SoV
SEIS_2-1-1	Unit Batteries (Fragility Group 15-03)	0.047	1.32	0.38	0.24	Anchorage	Representative†
SEIS_1B-2-1a	480v BD 219 A, B (U1, U2) (Fragility Group 01-05-01) Anchorage Failure	0.033	1.95	0.56	0.22	Anchorage	SoV
SEIS_1B-2-1f	480v BD 219 A, B (U1, U2) (Fragility Group 01-05-01) Functional Failure	0.022	2.61	0.64	0.24	Functionality	SoV
SEIS_1C-4R8	3EA,3EC SDBD Common Accident Signal Relays (CASA)	0.019	0.63	0.32	0.24	Functionality	CDFM
SEIS_12-1b	RHRSW pumps and EECW Alternate (Fragility Group 06-03)	0.017	1.45	0.32	0.24	Anchorage	CDFM
SEIS_3-2	Batt CH 248-1 (Fragility Group 16-06)	0.015	1.47	0.32	0.24	Functionality	CDFM
SEIS_11-1R3	Relay group 3 for group SEIS_11-1 (EECW Pp A3 UV device)	0.015	0.61	0.32	0.24	Functionality	CDFM
SEIS_1A-4W64	250v DC Bus A interface with wall 64 (Block Wall Group 3)	0.013	2.18	0.54	0.27	Block Wall Failure	SoV
SEIS_BLD-IPS	Intake Pumping Station	0.010	1.90	0.26	0.24	Structural Analysis	CDFM
SEIS_1C-1W63SD	Wall 63 falls towards 480v SD BD 2A and 2B	0.010	2.18	0.54	0.27	Block Wall Failure	SoV

Table 5.4-5: Unit 2 SCDF Importance Measures Ranked by F-V

Fragility Group	Description	F-V	A_m (g)	βu	βr	Failure Mode	Fragility Method
SEIS_11-1R1-1	Relay group 1-1 for group SEIS_11-1 (EECW Pp B3 UV device)	0.009	1.31	0.37	0.29	Functionality	SoV
SEIS_5-2B	Initiation relays and panels	0.009	0.97	0.32	0.24	Anchorage	CDFM
SEIS_2-1W8	Battery and Wall 8 or 47 interface (Block Wall Group 5)	0.007	2.12	0.41	0.29	Block Wall Failure	SoV
SEIS_1B-1	480V BD 1A, 2A, 3A, 1B, 2B, 3B (Fragility Group 02-01)	0.007	1.70	0.32	0.24	Functionality	CDFM
SEIS_11-1R2	Relay group 2 for group SEIS_11-1 (EECW Pp B3&D3 OC device)	0.006	1.29	0.38	0.24	Functionality	CDFM
SEIS_11-1R4	Relay group 4 for group SEIS_11-1 (EECW Pp A3 OC device)	0.006	0.94	0.51	0.27	Functionality	SoV
SEIS_1C-1W28SD	Wall 28 falls towards 480v SD BD 1A and 1B	0.006	2.18	0.54	0.27	Block Wall Failure	SoV
SEIS_14-1R1-2	Relays for Group SEIS_14-1, Relay Group 1 (HPCI/RCIC Isolations)	0.006	1.62	0.32	0.24	Functionality	CDFM
SEIS_HINST	Seismic failure of Main Control Room instrumentation	0.005	1.96	0.24	0.32	Functionality	CDFM
†See Section 5.7 for additional discussion on these representative fragilities							

Table 5.4-6: Unit 3 SCDF Importance Measures Ranked by F-V

Fragility Group	Description	F-V	A_m (g)	βu	βr	Failure Mode	Fragility Method
SEIS_LOOP	LOOP (Loss of Offsite Power)	0.900	0.30	0.45	0.30	Ceramic insulators	Table 6-1 NUREG/CR-6544
SEIS_1B-2-2	480v BD 219 3EA, 3EB (U3) (Fragility Group 01-05-02)	0.049	1.14	0.32	0.24	Anchorage	CDFM
SEIS_12-1P-1	RHRSW pumps based on pipe frag (Pipe Calc)	0.046	2.45	0.61	0.47	Soil (Buried Piping) Failure	SoV
SEIS_12-1P-2	EECW pumps based on pipe frag calc	0.044	2.48	0.60	0.48	Soil (Buried Piping) Failure	SoV
SEIS_2-1-1	Unit Batteries (Fragility Group 15-03)	0.036	1.32	0.38	0.24	Anchorage	Representative†
SEIS_1C-4R8	3EA,3EC SDBD Common Accident Signal Relays (CASA)	0.032	0.63	0.32	0.24	Functionality	CDFM
SEIS_1B-2-1a	480v BD 219 A, B (U1, U2) (Fragility Group 01-05-01) Anchorage Failure	0.026	1.95	0.56	0.22	Anchorage	SoV
SEIS_1C-4	U3 4kv SD BD EA and EC (Fragility Group 03-03)	0.018	1.09	0.32	0.24	Functionality	CDFM
SEIS_1B-2-1f	480v BD 219 A, B (U1, U2) (Fragility Group 01-05-01) Functional Failure	0.018	2.61	0.64	0.24	Functionality	SoV
SEIS_12-1b	RHRSW Pumps and EECW Alternate (Fragility Group 06-03)	0.012	1.45	0.32	0.24	Anchorage	CDFM
SEIS_3-2	Batt CH 248-1 (Fragility Group 16-06)	0.011	1.47	0.32	0.24	Functionality	CDFM
SEIS_11-1R3	Relay group 3 for group SEIS_11-1 (EECW Pp A3 UV device)	0.009	0.61	0.32	0.24	Functionality	CDFM
SEIS_BLD-IPS	Intake Pumping Station	0.009	1.90	0.26	0.24	Structural Analysis	CDFM
SEIS_5-2B	Initiation relays and panels	0.008	0.97	0.32	0.24	Anchorage	CDFM

Table 5.4-6: Unit 3 SCDF Importance Measures Ranked by F-V

Fragility Group	Description	F-V	A_m (g)	βu	βr	Failure Mode	Fragility Method
SEIS_1C-1W95SD	Wall 95 falls towards 480v SD BD 3A and 3B	0.008	2.18	0.54	0.27	Block Wall Failure	SoV
SEIS_1C-1W95MOV	Wall 95 falls towards 480v RMOV 3A and 250vdc RMOV BD 3A	0.008	2.18	0.54	0.27	Block Wall Failure	SoV
SEIS_1C-4R4	3EA,3EC SDBD Lockout relays (86)	0.006	0.94	0.32	0.24	Functionality	CDFM
SEIS_1C-4R5-2	3EA,3EC SDBD Lockout relays (86)	0.006	0.94	0.32	0.24	Functionality	CDFM
SEIS_1C-4R7	3EA,3EC SDBD 480 Transformer Trip relays (50G)	0.005	0.82	0.32	0.24	Functionality	CDFM
SEIS_2-1W8	Battery and Wall 8 or 47 interface (Block Wall Group 5)	0.005	2.12	0.41	0.29	Block Wall Failure	SoV
[†]See Section 5.7 for additional discussion on these representative fragilities							

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

LOOP represents the most significant contributor, which is consistent with the results of previous SPRA studies across the nuclear industry that have found that extended LOOP events are dominant for seismic risk. The two fragility groups with the highest F-V values excluding LOOP are SEIS_12-1P-2 (EECW pumps based on pipe frag calc) and SEIS_12-1P-1 (RHRSW pumps based on pipe fragility calculation) for Unit 1 and Unit 2. SEIS_12-1P-1 and SEIS_1B-2-2 (480v BD 219 3EA, 3EB (U3) (Fragility Group 01-05-02)) are the most risk-significant groups for Unit 3. SEIS_12-1P-1 is important because the SEIS_12-1P-1 pipe rupture fails the RHRSW system, which is responsible for removing decay heat from the RHR system. SEIS_12-1P-2 fails the EECW pumps that supply emergency cooling water for the EDG engine coolers and Emergency Core Cooling System (ECCS) pump room coolers among other important pieces of equipment. SEIS_1B-2-2 is important because it leads to failure of the Unit 3 EDGs.

5.4.4 SCDF Component Importance (Non-Seismic Failures)

Components were determined to be significant if the component's Risk Achievement Worth (RAW) is greater than or equal to two or its F-V is greater than 0.005, per the definition of significant basic event in Reference [8]. Components are considered risk-significant if the component has a F-V value greater than 0.005 or a RAW greater than or equal to two for either the SCDF or SLERF importance measures.

Table 5.4-7 contains the F-V importance measures for each risk-significant individual component, common-cause, or test and maintenance basic event that appears in the seismic cutsets for the Unit 1 SCDF. Table 5.4-8 and Table 5.4-9 contain similar information for Units 2 and 3.

Table 5.4-7: Unit 1 SCDF Risk-Significant Individual Component Importance by Fussell-Vesely

Basic Event	Description	F-V
DGGFR0EDG_082__DGA	DG A FAILS TO RUN	6.62E-03
DGGFR0EDG_082__DGC	DG C FAILS TO RUN	5.60E-03
TM_0BATA2480000A	SHUTDOWN BATTERY SB-A UNAVAILABLE DUE TO TEST AND MAINTENANCE	5.43E-03
TM_0PNLA2480000A	SHUTDOWN BATTERY BOARD SB-A UNAVAILABLE DUE TO TEST AND MAINTENANCE	5.43E-03
DGGFR0EDG_082__DGB	DG B FAILS TO RUN	5.39E-03

Table 5.4-8: Unit 2 SCDF Risk-Significant Component Importance by Fussell-Vesely

Basic Event	Description	F-V
DGGFR0EDG_082__DGD	DG D FAILS TO RUN	6.90E-03
DGGFR0EDG_082__DGB	DG B FAILS TO RUN	6.65E-03
DGGFR0EDG_082__DGC	DG C FAILS TO RUN	6.33E-03

Table 5.4-9: Unit 3 SCDF Risk-Significant Component Importance by Fussell-Vesely

Basic Event	Description	F-V
TM_0BATA2480003	0-BATA-248-0003 (MAIN BATTERY 3) UNAVAILABLE DUE TO T&M	1.28E-02
DGGFR3EDG_082_DG3C	DG 3C FAILS TO RUN	7.11E-03
DGGFR3EDG_082_DG3A	DG 3A FAILS TO RUN	5.98E-03

Another method of risk ranking components is by RAW. For basic events, RAW is determined by setting the event to TRUE, solving the model, and dividing the recalculated SCDF by the original SCDF. Tables 5.4-10, 5.4-11, and 5.4-12 list the components, common cause groups, and test and maintenance events with a RAW value greater than or equal to two for the Unit 1, 2, and 3 CDF, respectively.

Table 5.4-10: Unit 1 SCDF Risk-Significant Individual Components, Common-Cause Groups, and Test/Maintenance Events' Importance by RAW

Basic Event	Description	RAW
U0_21104CBKFO_ALL	CCF of all components in group 'U0_21104CBKFO'	10.5
U0_08208DGGFD_ALL	CCF of all components in group 'U0_08208DGGFD'	10.5
U0_21104CBKFO_1_2_3	CCF of three components: CBKFO0BKR_211A_003 & CBKFO0BKR_211B_002 & CBKFO0BKR_211C_022	10.5
U0_21104CBKFO_1_2_4	CCF of three components: CBKFO0BKR_211A_003 & CBKFO0BKR_211B_002 & CBKFO0BKR_211D_022	10.5
U0_08208DGGFR_ALL	CCF of all components in group 'U0_08208DGGFR'	10.5
U0_03002FANFR1_1_2	CCF of two components: FANFR0FAN_0300072 & FANFR0FAN_0300073	10.5
U0_03008FANFD_ALL	CCF of all components in group 'U0_03008FANFD'	10.5
U1_CCFMECH	CCF of mechanical equipment causes failure to scram	10.3
U0_03008FANFR_ALL	CCF of all components in group 'U0_03008FANFR'	10.3
U0_08208DGGFR_1_2_3	CCF of three components: DGGFR0EDG_082__DGA & DGGFR0EDG_082__DGB & DGGFR0EDG_082__DGC	10.3
U0_08208DGGFR_1_2_4	CCF of three components: DGGFR0EDG_082__DGA & DGGFR0EDG_082__DGB & DGGFR0EDG_082__DGD	10.3
U0_02308PRHFR_ALL	CCF of all components in group 'U0_02308PRHFR'	10.3
U0_02308PRHFD_ALL	CCF of all components in group 'U0_02308PRHFD'	10.3
U0_08208DGGFD_1_2_3	CCF of three components: DGGFD0EDG_082__DGA & DGGFD0EDG_082__DGB & DGGFD0EDG_082__DGC	10.3
U0_08208DGGFD_1_2_4	CCF of three components: DGGFD0EDG_082__DGA & DGGFD0EDG_082__DGB & DGGFD0EDG_082__DGD	10.3
U0_24804BCHFR_ALL	CCF of all components in group 'U0_24804BCHFR'	7.2
U0_24803BATFR_1_2_3	CCF of three components: BATFR0BATA2480001 & BATFR0BATA2480002 & BATFR0BATA2480003	7.2
U1_07102FANFR_1_2	CCF of two components: FANFR1FAN_0710601 & FANFR1FAN_0710602	6.2
U0_02308CKVFO_ALL	CCF of all components in group 'U0_02308CKVFO'	4.3

Table 5.4-10: Unit 1 SCDF Risk-Significant Individual Components, Common-Cause Groups, and Test/Maintenance Events' Importance by RAW

Basic Event	Description	RAW
FANFR1FAN_0710602	SAI PANEL 9-82 FAN COOLER FAN-71-602 FAILED	3.6
TM_1BDDD2810001A	250V RMOV BD 1A UNAVAILABLE DUE TO TEST AND MAINTENANCE	3.6
BUSFR1BDDD2810001A	250V RMOV BD 1A BUS FAILED	3.5
HFL_1068CCFPTLOPR	LOW RX PRESSURE PERMISSIVE SIGNAL COMMON CAUSE MISCALIBRATION	3.3
U1_00304BISFD3_ALL	CCF of all components in group 'U1_00304BISFD3'	3.2
U0_08208DGGFR_1_2	CCF of two components: DGGFR0EDG_082__DGA & DGGFR0EDG_082__DGB	2.7
U0_21104CBKFO_1_2	CCF of two components: CBKFO0BKR_211A_003 & CBKFO0BKR_211B_002	2.4
U0_21104CBKFO_2_3_4	CCF of three components: CBKFO0BKR_211B_002 & CBKFO0BKR_211C_022 & CBKFO0BKR_211D_022	2.0

Table 5.4-11: Unit 2 SCDF Risk-Significant Individual Components, Common-Cause Groups, and Test/Maintenance Events' Importance by RAW

Basic Event	Description	RAW
U0_03008FANFD_ALL	CCF of all components in group 'U0_03008FANFD'	10.3
U0_03002FANFR1_1_2	CCF of two components: FANFR0FAN_0300072 & FANFR0FAN_0300073	10.3
U0_08208DGGFR_ALL	CCF of all components in group 'U0_08208DGGFR'	10.3
U2_CCFMECH	CCF of mechanical equipment causes failure to scram	10.1
U0_02308PRHFD_ALL	CCF of all components in group 'U0_02308PRHFD'	10.1
U0_02308PRHFR_ALL	CCF of all components in group 'U0_02308PRHFR'	10.1
U0_03008FANFR_ALL	CCF of all components in group 'U0_03008FANFR'	10.1
U0_08208DGGFD_2_3_4	CCF of three components: DGGFD0EDG_082__DGB & DGGFD0EDG_082__DGC & DGGFD0EDG_082__DGD	10.1
U0_08208DGGFD_ALL	CCF of all components in group 'U0_08208DGGFD'	10.1
U0_08208DGGFR_2_3_4	CCF of three components: DGGFR0EDG_082__DGB & DGGFR0EDG_082__DGC & DGGFR0EDG_082__DGD	10.1
U0_21104CBKFO_2_3_4	CCF of three components: CBKFO0BKR_211B_002 & CBKFO0BKR_211C_022 & CBKFO0BKR_211D_022	10.1
U0_21104CBKFO_ALL	CCF of all components in group 'U0_21104CBKFO'	10.1
U0_21104CBKFO_1_2_4	CCF of three components: CBKFO0BKR_211A_003 & CBKFO0BKR_211B_002 & CBKFO0BKR_211D_022	7.3
U0_08208DGGFR_1_2_4	CCF of three components: DGGFR0EDG_082__DGA & DGGFR0EDG_082__DGB & DGGFR0EDG_082__DGD	7.1
U0_24803BATFR_1_2_3	CCF of three components: BATFR0BATA2480001 & BATFR0BATA2480002 & BATFR0BATA2480003	7.1
U0_24804BCHFR_ALL	CCF of all components in group 'U0_24804BCHFR'	7.1

Table 5.4-11: Unit 2 SCDF Risk-Significant Individual Components, Common-Cause Groups, and Test/Maintenance Events' Importance by RAW

Basic Event	Description	RAW
U0_08208DGGFD_1_2_4	CCF of three components: DGGFD0EDG_082__DGA & DGGFD0EDG_082__DGB & DGGFD0EDG_082__DGD	6.9
U2_07102FANFR_1_2	CCF of two components: FANFR2FAN_0710601 & FANFR2FAN_0710602	6.0
U0_02308CKVFO_ALL	CCF of all components in group 'U0_02308CKVFO'	4.2
FANFR2FAN_0710602	SAI PANEL 9-82 FAN COOLER FAN-71-602 FAILED	3.6
TM_2BDDD2810002A	250V RMOV BD 2A UNAVAILABLE DUE TO TEST AND MAINTENANCE	3.6
BUSFR2BDDD2810002A	250V RMOV BD 2A FAILS	3.5
U2_00304BISFD3_ALL	CCF of all components in group 'U2_00304BISFD3'	3.2
U0_08208DGGFR_2_4	CCF of two components: DGGFR0EDG_082__DGB & DGGFR0EDG_082__DGD	2.3
U0_21104CBKFO_2_4	CCF of two components: CBKFO0BKR_211B_002 & CBKFO0BKR_211D_022	2.2
U0_08208DGGFR_2_3	CCF of two components: DGGFR0EDG_082__DGB & DGGFR0EDG_082__DGC	2.1

Table 5.4-12: Unit 3 SCDF Risk-Significant Components, Common-Cause Groups, and Test/Maintenance Events' Importance by RAW

Basic Event	Description	RAW
U0_03008FANFD_ALL	CCF of all components in group 'U0_03008FANFD'	9.3
U3_03008FANFD_ALL	CCF of all components in group 'U3_03008FANFD'	9.3
U0_03002FANFR1_1_2	CCF of two components: FANFR0FAN_0300072 & FANFR0FAN_0300073	9.3
U3_03002FANFR_1_2	CCF of two components: FANFR3FAN_0300243 & FANFR3FAN_0300244	9.3
U0_08208DGGFR_ALL	CCF of all components in group 'U0_08208DGGFR'	9.3
U3_CCFMECH	CCF of mechanical equipment causes failure to scram	9.1
U0_02308PRHFD_ALL	CCF of all components in group 'U0_02308PRHFD'	9.1
U0_02308PRHFR_ALL	CCF of all components in group 'U0_02308PRHFR'	9.1
U0_08208DGGFD_1_2_4	CCF of three components: DGGFD0EDG_082__DGA & DGGFD0EDG_082__DGB & DGGFD0EDG_082__DGD	9.1
U0_08208DGGFD_5_6_7	CCF of three components: DGGFD3EDG_082_DG3A & DGGFD3EDG_082_DG3B & DGGFD3EDG_082_DG3C	9.1
U0_08208DGGFR_1_2_4	CCF of three components: DGGFR0EDG_082__DGA & DGGFR0EDG_082__DGB & DGGFR0EDG_082__DGD	9.1
U0_08208DGGFR_5_6_7	CCF of three components: DGGFR3EDG_082_DG3A & DGGFR3EDG_082_DG3B & DGGFR3EDG_082_DG3C	9.1
U0_03008FANFR_ALL	CCF of all components in group 'U0_03008FANFR'	9.1
U3_03008FANFR_ALL	CCF of all components in group 'U3_03008FANFR'	9.1

Table 5.4-12: Unit 3 SCDF Risk-Significant Components, Common-Cause Groups, and Test/Maintenance Events' Importance by RAW

Basic Event	Description	RAW
U0_21104CBKFO_1_2_4	CCF of three components: CBKFO0BKR_211A_003 & CBKFO0BKR_211B_002 & CBKFO0BKR_211D_022	9.1
U3_21104CBKFO_1_2_3	CCF of three components: CBKFO3BKR_211A_007 & CBKFO3BKR_211B_014 & CBKFO3BKR_211C_012	9.1
U0_08208DGGFD_ALL	CCF of all components in group 'U0_08208DGGFD'	9.1
U0_21104CBKFO_ALL	CCF of all components in group 'U0_21104CBKFO'	9.1
U3_21104CBKFO_ALL	CCF of all components in group 'U3_21104CBKFO'	9.1
U0_24803BATFR_1_2_3	CCF of three components: BATFR0BATA2480001 & BATFR0BATA2480002 & BATFR0BATA2480003	6.4
U0_24804BCHFR_ALL	CCF of all components in group 'U0_24804BCHFR'	6.4
U3_07102FANFR_1_2	CCF of two components: FANFR3FAN_0710601 & FANFR3FAN_0710602	5.4
U0_02308CKVFO_ALL	CCF of all components in group 'U0_02308CKVFO'	3.9
HINST3	Seismic - Instrument Impact on MCR HEPs - G03	3.6
HINST2	Seismic - Instrument Impact on MCR HEPs - G02	3.5
TM_3BDDD2810003A	250V RMOV BD 3A UNAVAILABLE DUE TO TEST AND MAINTENANCE	3.4
FANFR3FAN_0710602	SAI PANEL 9-82 FAN COOLER FAN-71-602 FAILED	3.4
BUSFR3BDDD2810003A	250V RMOV BD 3A FAILS	3.2
U0_08208DGGFR_1_6_7	CCF of three components: DGGFR0EDG_082__DGA & DGGFR3EDG_082_DG3B & DGGFR3EDG_082_DG3C	3.2
HFL_3068CCFPTLOPR	LOW RX PRESSURE PERMISSIVE SIGNAL COMMON CAUSE MISCALIBRATION	3.1
U3_00304BISFD3_ALL	CCF of all components in group 'U3_00304BISFD3'	2.9
TM_0BATA2480003	0-BATA-248-0003 (MAIN BATTERY 3) UNAVAILABLE DUE TO T&M	2.4
U0_21104CBKFO_1_2_3	CCF of three components: CBKFO0BKR_211A_003 & CBKFO0BKR_211B_002 & CBKFO0BKR_211C_022	2.2
FUSSO0FU2_2803_111	FUSED SWITCH 111 FAILS	2.2
U0_08208DGGFD_1_6_7	CCF of three components: DGGFD0EDG_082__DGA & DGGFD3EDG_082_DG3B & DGGFD3EDG_082_DG3C	2.2
U3_21104CBKFO_1_3_4	CCF of three components: CBKFO3BKR_211A_007 & CBKFO3BKR_211C_012 & CBKFO3BKR_211D_008	2.2
BATFR0BATA2480003	BATTERY 3 FAILS DURING OPERATION INCLUDING COMMON CAUSE	2.0
BUSFR0BDDD2800003	BATTERY BD. 3 FAILS	2.0
BUSFR3BDBB2680003B	480V RMOV BD 3B FAILS	2.0

5.4.5 Significant Human Failure Events

Significant post-initiator operator actions are defined as those operator action basic events that have an F-V value greater than 0.005 or a RAW greater than or equal to two. Note that the common methods of calculating RAW for basic events will not yield useful results for the HRA events, due to the processing of combination events. This is because the events are set to one during the quantification process and a recovery event representing the combination or single event is appended to the cutset. Therefore, setting the event to one to determine the RAW value has no effect on CDF. The F-V values of each operator action were determined in SYSIMP by defining groups where each particular operator action appearing in each seismic HRA bin (S1, S2, S3, and S4) was simultaneously set to FALSE in the combined cutset file to determine the combined importance across all seismic bins. Tables 5.4-13, 5.4-14, and 5.4-15 list the operator actions that were determined to be risk-significant for SCDF in the seismic model for Unit 1, Unit 2, and Unit 3, respectively.

Table 5.4-13: Risk-Significant Operator Actions for Unit 1 SCDF

Operator Action	Description	F-V
HFA_OPS_FLEXN2ALIGN	Operator fails to align N2 FLEX backup to Drywell Control Air (Safety Relief Valves (SRVs))	2.07E-01
HFA_0084CADALIGN	Operator fails to align Containment Air Dilution (CAD) backup to Drywell Control Air (DCA)	1.82E-01
HFA_0074HPSPC2	Failure to align RHR for suppression pool cooling (Anticipated Transient Without Scram (ATWS) or Inadvertent Opening of One Relief Valve(IOORV)) **	1.08E-01
HFA_0074HPSPC1	Failure to align RHR for suppression pool cooling (non-ATWS/IOORV)	6.59E-02
HFA_0023ALIGNEECW_L	Operator fails to align backup EECW pump	6.23E-02
HFA_OPS_4KVSDBDRESET	Operator reset of 4kV Shutdown Board lockout relays (seismic)	5.40E-02
HFA_0SBDALTDC	Operator fails to isolate SD BD and align alternate DC	5.35E-02
HFA_0248ALNALTCHG	Failure to align alternate battery charger	3.08E-02
HFA_0280ALNALTBBD	Operator fails to align alternate feeder	2.82E-02
HFA_0033HVACDOOR	Failure to open doors and install fans after HVAC failure	2.11E-02
HFA_0031STARTHVAC	Failure to start standby Control Bay HVAC	1.83E-02
HFA_0231480SDBTIE	Failure to transfer 480V shutdown board to alternate source	1.49E-02
HFA_0268480CRSTIE	Failure to transfer deenergized 480v board to alternate supply	1.13E-02
HFA_0001HPRVD1_L	Failure to initiate reactor-vessel depressurization (transient or ATWS)	8.90E-03
HFA_0073MANLEVEL	Operator fails to manually control level with High Pressure Coolant Injection (HPCI)	8.30E-03
HFA_HINST*	Seismic failure of Main Control Room instrumentation	5.10E-03

Table 5.4-13: Risk-Significant Operator Actions for Unit 1 SCDF

Operator Action	Description	F-V
*Although this is not an operator action but an equipment failure, it has an effect similar to that of multiple operator action failures.		

Table 5.4-14: Risk-Significant Operator Actions for Unit 2 SCDF

Operator Action	Description	F-V
HFA_OPS_FLEXN2ALIGN	Operator fails to align N2 FLEX backup to Drywell Control Air (SRVs)	2.27E-01
HFA_0084CADALIGN	Operator fails to align CAD backup to DCA	1.88E-01
HFA_0074HPSPC2	Failure to align RHR for suppression pool cooling (ATWS or IORV) **	1.12E-01
HFA_0074HPSPC1	Failure to align RHR for suppression pool cooling (non-ATWS/IORV)	7.10E-02
HFA_0023ALIGNEECW_L	Operator fails to align backup EECW pump	5.83E-02
HFA_0SBDALTDC	Operator fails to isolate SD BD and align alternate DC	4.71E-02
HFA_OPS_4KVSDBDRESET	Operator reset of 4kV Shutdown Board lockout relays (seismic)	3.70E-02
HFA_0248ALNALTCHG	Failure to align alternate battery charger	3.02E-02
HFA_0280ALNALTBBD	Operator fails to align alternate feeder	2.73E-02
HFA_0033HVACDOOR	Failure to open doors and install fans after HVAC failure	2.40E-02
HFA_0231480SDBTIE	Failure to transfer 480V shutdown board to alternate source	1.84E-02
HFA_0031STARTHVAC	Failure to start standby Control Bay HVAC	1.82E-02
HFA_0073MANLEVEL	Operator fails to manually control level with HPCI	1.76E-02
HFA_0268480CRSTIE	Failure to transfer deenergized 480v board to alternate supply	1.24E-02
HFA_0001HPRVD1	Failure to initiate reactor-vessel depressurization (transient or ATWS)	1.04E-02
HFA_0001HPRVD1_L	Failure to initiate reactor-vessel depressurization (transient or ATWS)	8.60E-03
HFA_HINST*	Seismic failure of Main Control Room instrumentation	5.10E-03
*Although this is not an operator action but an equipment failure, it has an effect similar to that of multiple operator action failures.		

Table 5.4-15: Risk-Significant Operator Actions for Unit 3 SCDF

Operator Action	Description	F-V
HFA_OPS_FLEXN2ALIGN	Operator fails to align N2 FLEX backup to Drywell Control Air (SRVs)	1.95E-01
HFA_0084CADALIGN	Operator fails to align CAD backup to DCA	1.53E-01

Table 5.4-15: Risk-Significant Operator Actions for Unit 3 SCDF

Operator Action	Description	F-V
HFA_0074HPSPC2	Failure to align RHR for suppression pool cooling (ATWS or IORV)	1.00E-01
HFA_0074HPSPC1	Failure to align RHR for suppression pool cooling (non-ATWS/IORV)	7.05E-02
HFA_OPS_4KVSDBDRESET	Operator reset of 4kV Shutdown Board lockout relays (seismic)	6.78E-02
HFA_0SBDALTDC	Operator fails to isolate SD BD and align alternate DC	4.78E-02
HFA_0231480SDBTIE	Failure to transfer 480V shutdown board to alternate source	4.53E-02
HFA_0023ALIGNEECW_L	Operator fails to align backup EECW pump	4.01E-02
HFA_0248ALNALTCHG	Failure to align alternate battery charger	1.50E-02
HFA_0268480CRSTIE	Failure to transfer deenergized 480v board to alternate supply	1.33E-02
HFA_0073MANLEVEL	Operator fails to manually control level with HPCI	1.08E-02
HFA_0280ALNALTBBD	Operator fails to align alternate feeder	1.07E-02
HFA_0001HPRVD1_L	Failure to initiate reactor-vessel depressurization (transient or ATWS)	7.70E-03
HFA_0033HVACDOOR	Failure to open doors and install fans after HVAC failure	6.30E-03
HFA_0001HPRVD1	Failure to initiate reactor-vessel depressurization (transient or ATWS)	5.70E-03
HFA_0031STARTHVAC	Failure to start standby Control Bay HVAC	5.00E-03

5.4.5.1 Summary of the Approach used to Evaluate Human Error Probabilities

The approach used to evaluate HEPs is based on EPRI 3002008093 [49]. Each HFE that was determined to be feasible for the SPRA was subject to a detailed analysis.

5.4.5.2 Detailed Analysis for HEPs

All HFEs in the BFN SPRA were analyzed with detailed HRA [42], in accordance with the guidance in EPRI 3002008093 [49].

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

5.4.5.3 Operator action credit for FLEX

The FLEX operator actions to align backup nitrogen to drywell control air for the SRVs were included in the model. No other FLEX systems were credited in the model.

5.4.6 Significant SCDF Accident Sequences

The SPRA evaluated the GTRAN, Direct Core Damage, LLOCA, MLOCA, IOORV, ISLOCA, and ATWS ETs that were the same ET categories considered in the IEPRA, except for Direct Core Damage, which is unique to the SPRA. The results were dominated by seismically induced LOOP transient scenarios, which are treated in the GTRAN ET. Since FRANX eliminates events that are set TRUE in its final processing, it is difficult to obtain accurate results of the exact percentages due to each accident sequence since the sequence flags are absent from the FRANX group cutset results. However, an examination of the results of a consolidated CAFTA cutset file without ACUBE processing and with the flags intact should show a relatively accurate representation of the percentages that each accident sequence contributes. The results show that the dominant sequences are GTRAN-005, GTRAN-006, GTRAN-002, and GTRAN-012 for Unit 1. Unit 2 is dominated by GTRAN-005, GTRAN-006, GTRAN-012, GTRAN-002 and GTRAN-011. Unit 3 is dominated by GTRAN-005, GTRAN-006, GTRAN-002, GTRAN-012 and GTRAN-001A. Figure 5.4-1, Figure 5.4-2, and Figure 5.4-3 show the relative percentage contribution for each dominant accident sequence based on a consolidated CAFTA cutset file for Unit 1, Unit 2, and Unit 3, respectively. Significant accident sequences for SCDF are discussed in Table 5.4-16. The differences in results between units are primarily the result of power dependency differences for the SPRA.

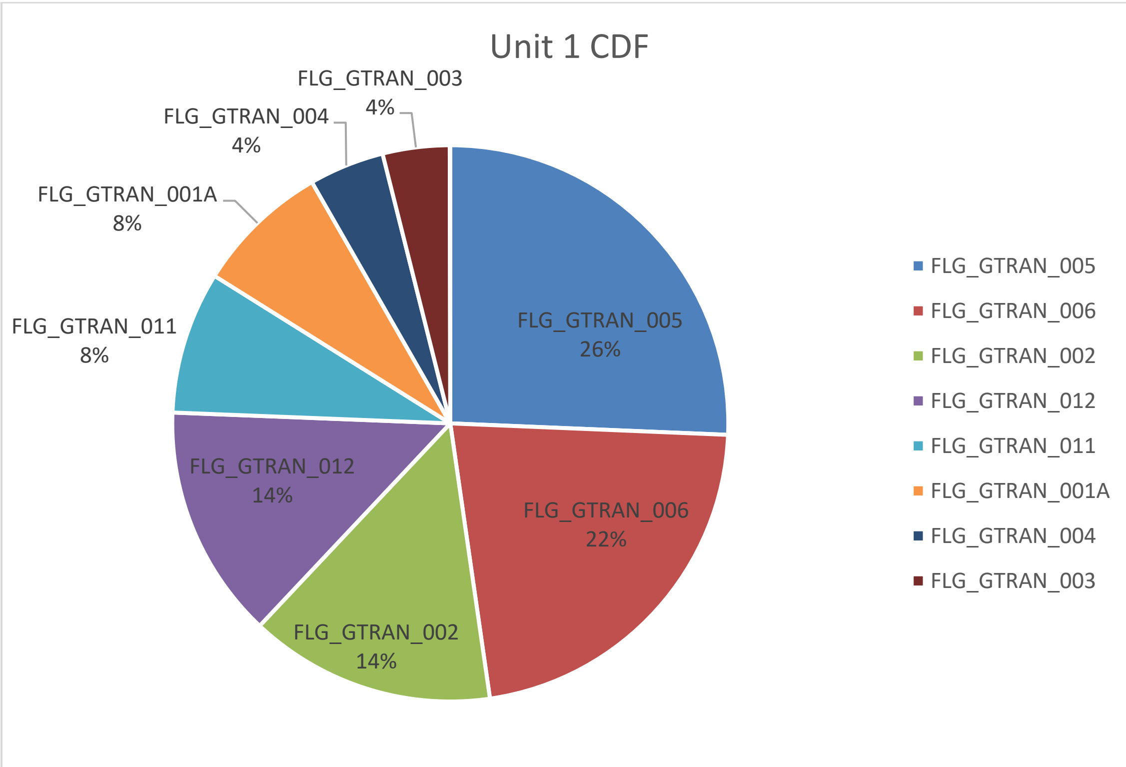

Figure 5.4-1: Unit 1 SCDF Accident Sequence Contribution

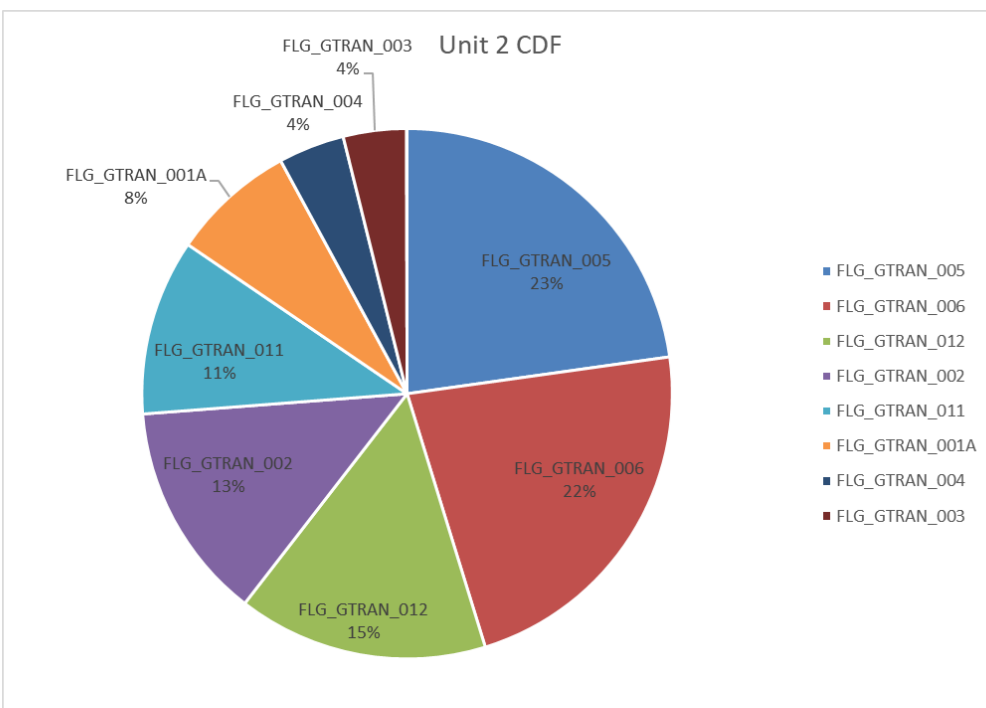

Figure 5.4-2: Unit 2 SCDF Accident Sequence Contribution

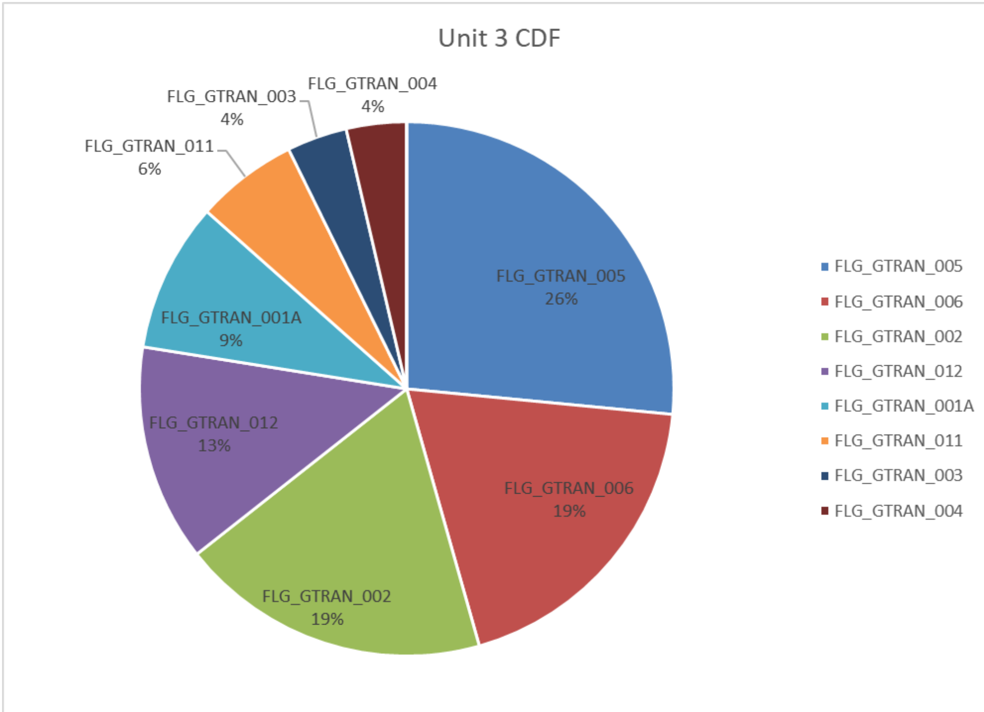

Figure 5.4-3: Unit 3 SCDF Accident Sequence Contribution

Table 5.4-16: Dominant SCDF Accident Sequences in SPRA

Accident Sequence	Description	Discussion
GTRAN-001A	The scram successfully occurs, the Power Conversion Systems (PCS) fails, and there are no breaks outside containment or stuck-open relief valves. HPCI or RCIC is successful for at least 4 hours. Alternate shutdown cooling fails. Control Rod Drive (CRD) fails. Low-pressure injection via core spray or Low Pressure Coolant Injection (LPCI) fail.	The PCS is assumed failed for all seismic scenarios. The CST is assumed failed for all seismic scenarios, which fails CRD injection. Top contributors to this sequence involve common accident signal relays fragility group (SEIS_1C-4R8) failing, which fails the diesels, and initiation relays and panels (SEIS_5-2B) failing, which fails standby coolant injection, LPCI, core spray, and shutdown cooling. The shutdown boards fail because of relay chatter, and the operator action to reset the relays also fails.
GTRAN-002	The scram successfully occurs, the PCS fails, and there are no breaks outside containment or stuck-open relief valves. HPCI or RCIC is successful for at least 4 hours. Early suppression pool cooling is not successful or initiated in time to prevent exceeding Heat Capacity Temperature Limit (HCTL) and 190 deg F in the suppression pool. Therefore, long-term HPCI or RCIC is not successful. The CRD and High Pressure Make-Up (HPMU) systems fail, and manual depressurization is challenged and initiated successfully with 2 SRVs at HCTL about 4 hours after the scram. After depressurization, low-pressure injection by RHR in the LPCI mode or Core Spray (CS) is successful. Alternate Shutdown Cooling (ASDC) is unsuccessful. Late suppression pool cooling fails, allowing the primary containment to pressurize to PCPL in about 13 hours, but the hardened wetwell vent successfully establishes Decay Heat Removal (DHR). After the successful vent, the suppression pool no longer provides a successful suction source for RHR or CS, and late injection is not re-established when all the available injection sources fail. These sources are condensate injection, standby coolant injection, shutdown cooling and RHR or CS with suction on the CST. Core damage is caused by loss of injection and the Reactor Pressure Vessel (RPV) is at low pressure.	The PCS is assumed failed for all seismic scenarios. The CST is assumed failed for all seismic scenarios, which fails CRD injection. HPCI and RCIC fail after 4 hours due a failure to align suppression pool cooling. Late suppression pool cooling is not successful because the operator fails to align it, and drywell sprays are also not available because of operator failure to align it. All low-pressure injection sources have failed. Shutdown cooling is not available due to operator alignment failure. LPCI has failed because it cannot take suction from the CST, which is assumed failed for seismic scenarios. Core spray is failed for the same reason. Standby coolant injection has failed because operators have failed to align it.
GTRAN-005	Sequence GTRAN-005 is the same as sequence GTRAN-002 except all low-pressure injections fail. core damage occurs about 1.5 hours after scram with the RPV at low pressure.	For Unit 1 and Unit 2, the driving seismic consideration is generally failure of fragility group SEIS_5-2A2-2-C-G06 (SEISMIC FRAGILITY FOR %G06: Initiation relays and panels

	A GTRAN-005 cutset is the dominant cutset for Units 1, 2, and 3 CDF.	(20-08)), which fails accident initiation signals for multiple systems. For Unit 3, it is failure of the 480V shutdown board via fragility group SEIS_1B-2-2-C-G06, while cutsets involving fragility group SEIS_5-2A2-2-C-G06 are also top contributors similar to Units 1 and 2.
GTRAN-006	Sequence GTRAN-006 is the same as sequence GTRAN-002 except depressurization fails. Without depressurization, there are no other available injection sources, and core damage occurs. CD occurs about 1.5 hours after scram with the RPV at high pressure. GTRAN-006 cutsets are in the top 10 most significant for all three units.	For all units, the dominant cutsets for GTRAN-006 are not directly related to seismic failures (other than systems like the CST that are assumed to be failed for seismic initiators and, therefore, do not appear in the cutsets), but rather the fact that all operator actions are assumed to be failed for seismic bin %G07-%G09. Top cutsets for Unit 1 involve simultaneous failure to initiate depressurization via operator intervention and to align RHR for suppression pool cooling, or simultaneous failure to establish alternate FLEX nitrogen supply for controlling the SRVs during emergency depressurization, along with the failure to align RHR for suppression pool cooling. For Unit 2 and Unit 3, these scenarios are also dominant.
GTRAN-011	Sequence GTRAN-011 is the same as sequence GTRAN-007 except all low-pressure injections fail. CD occurs in about 30 to 40 minutes with the RPV at low pressure. In sequence GTRAN-007, the scram successfully occurs, the PCS fails, and there are no breaks outside containment or stuck-open relief valves. Early HPCI or RCIC are unsuccessful as an initial injection source. A cooldown is initiated but HPMU is unsuccessful. CRD may be available but is not challenged because it lacks sufficient capacity to be used as an initial injection source. When RPV level drops to Top of Active Fuel (TAF), manual depressurization is successful. Low-pressure injection by RHR in the LPCI mode or CS is successful as an initial injection source but must be initiated within 30 minutes. ASDC is unsuccessful. Late suppression pool cooling and drywell spray are unsuccessful. Hardened Containment Vent (HCV) successfully accomplishes DHR. Without SPC, PCPL is reached in approximately 10 hours. Successful venting fails the suppression pool as a suction source, and RHR and CS with suction on the suppression pool are lost. A post-vent injection source of either RHR with suction on the CST, CS with suction on the CST, shutdown cooling, standby coolant injection,	The PCS is assumed failed for all seismic scenarios. The CST and the condensate system are assumed failed by the seismic event. HPCI fails early because it receives an erroneous isolation signal due to the failure of fragility group SEIS_14-1R1 (HPCI/RCIC isolations relays). This also fails RCIC. Early low-pressure injection fails because core spray and LPCI both fail. Core spray loop 1 fails because of loss of pump power from both 4kV shutdown boards A and B from a LOOP event and a loss of the diesels due to EECW failure due to RHRSW pump failure from fragility group SEIS_11-1R1 (EECW pump B&D relays). Core spray loop 2 is lost for similar reasons. LPCI loop 1 fails for several reasons, both because the loop 1 injection path fails and because the RHR loop 1 supply is not available. The LPCI loop 1 injection path fails because of reactor motor-operated valve (RMOV) board failure caused by loss of the normal and alternate shutdown board supplies due to a LOOP along with diesel generator failure as a result of EECW failure due again to fragility group SEIS_11-1R1. LPCI loop 2 fails for similar reasons.

	condensate injection, or CRD (1 pump) is unsuccessful. CD occurs due to loss of injection, and the RPV fails at low pressure.	
GTRAN-012	Sequence GTRAN-012 is the same as sequence GTRAN-007 except depressurization fails. Without depressurization, there are no other available injection sources, and core damage occurs. Core damage occurs in about 30 to 40 minutes with the RPV at high pressure.	The PCS is assumed failed for all seismic scenarios. RCIC and HPCI both fail. HPCI and RCIC both fail due to steam supply path failures because the RCIC steam supply line outboard isolation valve fails due to fragility group SEIS_14-1R1-2 (HPCI/RCIC isolations). Emergency depressurization fails due to operator failure to initiate depressurization.

5.5 SLERF Results

5.5.1 Overall SLERF

The baseline total LERF is 3.00E-6 /ry for Unit 1, 3.10E-6 /ry for Unit 2, and 3.31E-6 /ry for Unit 3.

5.5.2 SLERF as a Function of Hazard Interval

A summary of the SLERF results for each seismic hazard interval is presented in Table 5.5-1 for Unit 1 SLERF, Table 5.5-2 for Unit 2 SLERF, and Table 5.5-3 for Unit 3 SLERF.

Table 5.5-1: Unit 1 SLERF Contribution by Initiating Event

Truncation	Scenario	Description	Earthquake Frequency	CLERP	SLERF	Percent Contribution
1.0E-12	%G01	Seismic Initiating Event (0.1g to <0.2g)	2.68E-04	2.8E-05	7.50E-09	0.2%
1.0E-12	%G02	Seismic Initiating Event (0.2g to <0.3g)	5.56E-05	9.1E-04	5.05E-08	1.7%
2.0E-12	%G03	Seismic Initiating Event (0.3g to <0.6g)	2.98E-05	8.5E-03	2.52E-07	8.4%
7.0E-11	%G04	Seismic Initiating Event (0.6g to <0.7g)	2.17E-06	9.9E-02	2.15E-07	7.2%
2.0E-09	%G05	Seismic Initiating Event (0.7g to <0.9g)	2.19E-06	2.5E-01	5.57E-07	18.5%
6.0E-08	%G06	Seismic Initiating Event (0.9g to <1.5g)	1.82E-06	7.7E-01	1.40E-06	46.7%
5.0E-08	%G07	Seismic Initiating Event (1.5g to <1.7g)	1.80E-07	9.0E-01	1.61E-07	5.4%
2.0E-08	%G08	Seismic Initiating Event (1.7g to <3g)	3.32E-07	9.1E-01	3.03E-07	10.1%
2.0E-08	%G09	Seismic Initiating Event (>3g)	5.96E-08	9.2E-01	5.48E-08	1.8%
				Total SLERF=	**3.00E-06**	

Table 5.5-2: Unit 2 SLERF Contribution by Initiating Event

Truncation	Scenario	Description	Earthquake Frequency	CLERP	SLERF	Percent Contribution
1.0E-12	%G01	Seismic Initiating Event (0.1g to <0.2g)	2.68E-04	2.8E-05	7.62E-09	0.2%
5.0E-11	%G02	Seismic Initiating Event (0.2g to <0.3g)	5.56E-05	8.7E-04	4.86E-08	1.6%
5.0E-12	%G03	Seismic Initiating Event (0.3g to <0.6g)	2.98E-05	9.0E-03	2.68E-07	8.6%
2.0E-10	%G04	Seismic Initiating Event (0.6g to <0.7g)	2.17E-06	9.8E-02	2.12E-07	6.9%
2.0E-09	%G05	Seismic Initiating Event (0.7g to <0.9g)	2.19E-06	2.7E-01	5.88E-07	19.0%
5.0E-08	%G06	Seismic Initiating Event (0.9g to <1.5g)	1.82E-06	8.0E-01	1.45E-06	46.9%
4.0E-08	%G07	Seismic Initiating Event (1.5g to <1.7g)	1.80E-07	9.0E-01	1.61E-07	5.2%
2.0E-08	%G08	Seismic Initiating Event (1.7g to <3g)	3.32E-07	9.1E-01	3.03E-07	9.8%
2.0E-08	%G09	Seismic Initiating Event (>3g)	5.96E-08	9.2E-01	5.48E-08	1.8%
				Total SLERF=	**3.10E-06**	

Table 5.5-3: Unit 3 SLERF Contribution by Initiating Event

Truncation	Scenario	Description	Earthquake Frequency	CLERP	SLERF	Percent Contribution
1.0E-12	%G01	Seismic Initiating Event (0.1g to <0.2g)	2.68E-04	2.9E-05	7.72E-09	0.2%
5.0E-11	%G02	Seismic Initiating Event (0.2g to <0.3g)	5.56E-05	9.0E-04	5.00E-08	1.6%
4.0E-12	%G03	Seismic Initiating Event (0.3g to <0.6g)	2.98E-05	1.1E-02	3.17E-07	10.2%
4.0E-10	%G04	Seismic Initiating Event (0.6g to <0.7g)	2.17E-06	1.1E-01	2.43E-07	7.8%
4.0E-09	%G05	Seismic Initiating Event (0.7g to <0.9g)	2.19E-06	3.0E-01	6.57E-07	21.2%
5.0E-08	%G06	Seismic Initiating Event (0.9g to <1.5g)	1.82E-06	8.3E-01	1.52E-06	49.0%
5.0E-08	%G07	Seismic Initiating Event (1.5g to <1.7g)	1.80E-07	9.0E-01	1.61E-07	5.2%
2.0E-08	%G08	Seismic Initiating Event (1.7g to <3g)	3.32E-07	9.1E-01	3.03E-07	9.8%
2.0E-08	%G09	Seismic Initiating Event (>3g)	5.96E-08	9.2E-01	5.48E-08	1.8%
				Total SLERF=	**3.31E-06**	

5.5.3 Fragility Group Importance for SLERF

The SSCs with the most significant seismic failure contributions to SLERF for Unit 1 are listed in Table 5.5-4, sorted by F-V. The seismic fragilities for each of the significant contributors are also provided in Table 5.5-4, along with the corresponding limiting seismic failure mode and method of fragility calculation. The corresponding measures for Unit 2 and Unit 3 are presented in Table 5.5-5 and 5.5-6.

Table 5.5-4: Unit 1 SLERF Fragility Group Importance Measures Ranked by F-V

Fragility Group	Description	F-V	A_m (g)	βu	βr	Failure Mode	Fragility Method
SEIS_LOOP	LOOP (Loss of Offsite Power)	0.820	0.30	0.45	0.30	Ceramic insulators	Table 6-1 NUREG/CR-6544
SEIS_5-2B	Initiation relays and panels	0.210	0.97	0.32	0.24	Anchorage	CDFM
SEIS_2-1-1	Unit Batteries (Fragility Group 15-03)	0.183	1.32	0.38	0.24	Anchorage	Representative†
SEIS_5-8	REACTOR PROTECTION & NSS PNL (18-02)	0.056	1.22	0.32	0.24	Functionality	Representative†
SEIS_12-1P-2	EECW pumps based on pipe frag calc	0.042	2.48	0.60	0.48	Soil (Buried Piping) Failure	SoV
SEIS_2-1W8	Battery and Wall 8 or 47 interface (Block Wall Group 5)	0.035	2.12	0.41	0.29	Block Wall Failure	SoV
SEIS_1B-2-1a	480v BD 219 A, B (U1, U2) (Fragility Group 01-05-01) Anchorage Failure	0.034	1.95	0.56	0.22	Anchorage	SoV
SEIS_BLD-IPS	Intake Pumping Station	0.021	1.90	0.26	0.24	Structural Analysis	CDFM
SEIS_1B-2-1f	480v BD 219 A, B (U1, U2) (Fragility Group 01-05-01) Functional Failure	0.021	2.61	0.64	0.24	Functionality	SoV
SEIS_12-1b	RHRSW pumps and EECW Alternate (Fragility Group 06-03)	0.016	1.45	0.32	0.24	Anchorage	CDFM
SEIS_1B-1	480V BD 1A, 2A, 3A, 1B, 2B, 3B (Fragility Group 02-01)	0.015	1.70	0.32	0.24	Functionality	CDFM
SEIS_12-1P-1	RHRSW pumps based on pipe frag (Pipe Calc)	0.012	2.45	0.61	0.47	Soil (Buried Piping) Failure	SoV
SEIS_11-1R1-1	Relay group 1-1 for group SEIS_11-1 (EECW Pp B3 UV device)	0.011	1.31	0.37	0.29	Functionality	SoV
SEIS_14-1R1-2	Relays for Group SEIS_14-1, Relay Group 1 (HPCI/RCIC Isolations)	0.011	1.62	0.32	0.24	Functionality	CDFM

Table 5.5-4: Unit 1 SLERF Fragility Group Importance Measures Ranked by F-V

Fragility Group	Description	F-V	A_m (g)	βu	βr	Failure Mode	Fragility Method
SEIS_11-1R3	Relay group 3 for group SEIS_11-1 (EECW Pp A3 UV device)	0.011	0.61	0.32	0.24	Functionality	CDFM
SEIS_1C-1W28SD	Wall 28 falls towards 480v SD BD 1A and 1B	0.011	2.18	0.54	0.27	Block Wall Failure	SoV
SEIS_5-4	Panels group 4-1 Control Room panels Lower Fragility (Fragility Group 20-02 and 20-03)	0.010	2.04	0.38	0.24	Anchorage	Representative†
SEIS_11-1R2	Relay group 2 for group SEIS_11-1 (EECW Pp B3&D3 OC device)	0.009	1.29	0.38	0.24	Functionality	CDFM
SEIS_1C-4R8	3EA,3EC SDBD Common Accident Signal Relays (CASA)	0.006	0.63	0.32	0.24	Functionality	CDFM
SEIS_5-2A2-2	Initiation relays and panels (20-08)	0.006	1.30	0.58	0.23	Anchorage	SoV
SEIS_1C-1W28MOV	Wall 28 falls toward 480v RMOV 1A and 4kv SD BD A	0.006	2.18	0.54	0.27	Blockwall	SoV
SEIS_12-1a	EECW pumps (Fragility Group 06-03-01)	0.005	1.34	0.32	0.24	Anchorage	CDFM

†See Section 5.7 for additional discussion on these representative fragilities

Table 5.5-5: Unit 2 SLERF Fragility Group Importance Measures Ranked by F-V

Fragility Group	Description	F-V	A_m (g)	βu	βr	Failure Mode	Fragility Method
SEIS_LOOP	LOOP (Loss of Offsite Power)	0.813	0.30	0.45	0.30	Ceramic insulators	Table 6-1 NUREG/CR-6544
SEIS_5-2B	Initiation relays and panels	0.189	0.97	0.32	0.24	Anchorage	CDFM
SEIS_2-1-1	Unit Batteries (Fragility Group 15-03)	0.184	1.32	0.38	0.24	Anchorage	Representative†
SEIS_5-8	REACTOR PROTECTION & NSS PNL (18-02)	0.070	1.22	0.32	0.24	Functionality	Representative†
SEIS_12-1P-2	EECW Pumps based on pipe frag calc	0.044	2.48	0.60	0.48	Soil (Buried Piping) Failure	SoV
SEIS_1B-2-1a	480v BD 219 A, B (U1, U2) (Fragility Group 01-05-01) Anchorage Failure	0.039	1.95	0.56	0.22	Anchorage	SoV
SEIS_2-1W8	Battery and Wall 8 or 47 interface (Block Wall Group 5)	0.039	2.12	0.41	0.29	Block Wall Failure	SoV
SEIS_14-1R1-2	Relays for Group SEIS_14-1, Relay Group 1 (HPCI/RCIC Isolations)	0.036	1.62	0.32	0.24	Functionality	CDFM
SEIS_12-1b	RHRSW Pumps and EECW Alternate (Fragility Group 06-03)	0.033	1.45	0.32	0.24	Anchorage	CDFM
SEIS_5-2A2-2	Initiation relays and panels (20-08)	0.031	1.30	0.58	0.23	Anchorage	SoV
SEIS_11-1R1-1	Relay group 1-1 for group SEIS_11-1 (EECW Pp B3 UV device)	0.031	1.31	0.37	0.29	Functionality	SoV
SEIS_11-1R2	Relay group 2 for group SEIS_11-1 (EECW Pp B3&D3 OC device)	0.030	1.29	0.38	0.24	Functionality	CDFM
SEIS_11-1R3	Relay group 3 for group SEIS_11-1 (EECW Pp A3 UV device)	0.030	0.61	0.32	0.24	Functionality	CDFM
SEIS_BLD-IPS	Intake Pumping Station	0.025	1.90	0.26	0.24	Structural Analysis	CDFM
SEIS_1C-4R8	3EA,3EC SDBD Common Accident Signal Relays (CASA)	0.024	0.63	0.32	0.24	Functionality	CDFM

Table 5.5-5: Unit 2 SLERF Fragility Group Importance Measures Ranked by F-V

Fragility Group	Description	F-V	A_m (g)	βu	βr	Failure Mode	Fragility Method
SEIS_11-1R4	Relay group 4 for group SEIS_11-1 (EECW Pp A3 OC device)	0.023	0.94	0.51	0.27	Functionality	SoV
SEIS_1B-2-1f	480v BD 219 A, B (U1, U2) (Fragility Group 01-05-01) Functional Failure	0.022	2.61	0.64	0.24	Functionality	SoV
SEIS_1C-6R6	C SDBD 480 Transformer Trip relays (50G)	0.021	1.29	0.38	0.24	Functionality	CDFM
SEIS_1C-6R2	C DG BKR Trip Relays (CAR, OTX, RI, VRL, VRR)	0.021	1.43	0.32	0.24	Functionality	CDFM
SEIS_1C-4R7	3EA,3EC SDBD 480 Transformer Trip relays (50G)	0.020	0.82	0.32	0.24	Functionality	CDFM
SEIS_1C-4R4	3EA,3EC SDBD Lockout relays (86)	0.020	0.94	0.32	0.24	Functionality	CDFM
SEIS_1C-4R5-2	3EA,3EC SDBD Lockout relays (86)	0.020	0.94	0.32	0.24	Functionality	CDFM
SEIS_5-2A2-1	Initiation relays and panels (20-07)	0.020	1.96	0.32	0.24	Functionality	CDFM
SEIS_1B-2-2	480v BD 219 3EA, 3EB (U3) (Fragility Group 01-05-02)	0.020	1.14	0.32	0.24	Anchorage	CDFM
SEIS_1C-4	U3 4kv SD BD EA and EC (Fragility Group 03-03)	0.020	1.09	0.32	0.24	Functionality	CDFM
SEIS_1A-4W64	250v DC Bus A interface with wall 64 (Block Wall Group 3)	0.019	2.18	0.54	0.27	Block Wall Failure	SoV
SEIS_12-1a	EECW Pumps (Fragility Group 06-03-01)	0.018	1.34	0.32	0.24	Anchorage	CDFM
SEIS_12-1P-1	RHRSW Pumps based on pipe frag (Pipe Calc)	0.018	2.45	0.61	0.47	Soil (Buried Piping) Failure	SoV
SEIS_1B-1	480V BD 1A, 2A, 3A, 1B, 2B, 3B (Fragility Group 02-01)	0.017	1.7	0.32	0.24	Functionality	CDFM
SEIS_HINST	Seismic failure of Main Control Room instrumentation	0.016	1.96	0.24	0.32	Functionality	CDFM

Table 5.5-5: Unit 2 SLERF Fragility Group Importance Measures Ranked by F-V

Fragility Group	Description	F-V	A_m (g)	βu	βr	Failure Mode	Fragility Method
SEIS_1C-6R8	C SDBD Common Accident Signal Relays (CASA)	0.013	1.62	0.38	0.24	Functionality	CDFM
SEIS_1C-3R8	B,D SDBD Common Accident Signal Relays (CASA)	0.013	1.62	0.38	0.24	Functionality	CDFM
SEIS_1C-1W63SD	Wall 63 falls towards 480v SD BD 2A and 2B	0.012	2.18	0.54	0.27	Blockwall	SoV
SEIS_5-4	Panels group 4-1 Control Room panels Lower Fragility (Fragility Group 20-02 and 20-03)	0.011	2.04	0.38	0.24	Functionality	Representative†
SEIS_1C-4R3	3A,3C DG BKR Trip relay (R3)	0.008	1.26	0.32	0.24	Functionality	CDFM
SEIS_1C-1R6	A SDBD 480 Transformer Trip relays (50G)	0.008	1.29	0.38	0.24	Functionality	CDFM
SEIS_1C-1W63MOV	Wall 63 falls toward 480v RMOV 2A and 4kv SD BD C	0.007	2.18	0.54	0.27	Blockwall	SoV
SEIS_19-4	CAD NITROGEN STRG TNK (084) (Fragility Group 21-06)	0.007	1.22	0.26	0.24	Anchorage	CDFM
SEIS_9-1	EDG (Fragility Group 17-01)	0.006	2.03	0.32	0.24	Functional	Representative†
SEIS_1A-2	250v DC bus B (Fragility Group 01-02)	0.006	1.89	0.32	0.24	Functional	Representative†
SEIS_5-9-1	HPCI/RCIC PNLS (925-0058&63)	0.005	1.82	0.32	0.24	Anchorage	Representative†

†See Section 5.7 for additional discussion on these representative fragilities

Table 5.5-6: Unit 3 SLERF Fragility Group Importance Measures Ranked by F-V

Fragility Group	Description	F-V	A_m (g)	βu	βr	Failure Mode	Fragility Method
SEIS_LOOP	LOOP (Loss of Offsite Power)	0.813	0.30	0.45	0.30	Ceramic insulators	Table 6-1 NUREG/CR-6544
SEIS_5-2B	Initiation relays and panels	0.195	0.97	0.32	0.24	Anchorage	CDFM
SEIS_2-1-1	Unit Batteries (Fragility Group 15-03)	0.135	1.32	0.38	0.24	Anchorage	Representative†
SEIS_1B-2-2	480v BD 219 3EA, 3EB (U3) (Fragility Group 01-05-02)	0.066	1.14	0.32	0.24	Anchorage	CDFM
SEIS_5-8	REACTOR PROTECTION & NSS PNL (18-02)	0.054	1.22	0.32	0.24	Functionality	Representative†
SEIS_12-1P-2	EECW Pumps based on pipe frag calc	0.031	2.48	0.60	0.48	Soil (Buried Piping) Failure	SoV
SEIS_1B-2-1a	480v BD 219 A, B (U1, U2) (Fragility Group 01-05-01) Anchorage Failure	0.029	1.95	0.56	0.22	Anchorage	SoV
SEIS_2-1W8	Battery and Wall 8 or 47 interface (Block Wall Group 5)	0.025	2.12	0.41	0.29	Block Wall Failure	SoV
SEIS_11-1R3	Relay group 3 for group SEIS_11-1 (EECW Pp A3 UV device)	0.024	0.61	0.32	0.24	Functionality	CDFM
SEIS_1C-4	U3 4kv SD BD EA and EC (Fragility Group 03-03)	0.017	1.09	0.32	0.24	Functionality	CDFM
SEIS_1B-2-1f	480v BD 219 A, B (U1, U2) (Fragility Group 01-05-01) Functional Failure	0.017	2.61	0.64	0.24	Functionality	SoV
SEIS_BLD-IPS	Intake Pumping Station	0.016	1.90	0.26	0.24	Structural Analysis	CDFM
SEIS_1C-4R8	3EA,3EC SDBD Common Accident Signal Relays (CASA)	0.015	0.63	0.32	0.24	Functionality	CDFM
SEIS_14-1R1-2	Relays for Group SEIS_14-1, Relay Group 1 (HPCI/RCIC Isolations)	0.010	1.62	0.32	0.24	Functionality	CDFM

Table 5.5-6: Unit 3 SLERF Fragility Group Importance Measures Ranked by F-V

Fragility Group	Description	F-V	A_m (g)	βu	βr	Failure Mode	Fragility Method
SEIS_11-1R4	Relay group 4 for group SEIS_11-1 (EECW Pp A3 OC device)	0.010	0.94	0.51	0.27	Functionality	SoV
SEIS_5-2A2-1	Initiation relays and panels (20-07)	0.010	1.96	0.32	0.24	Functionality	CDFM
SEIS_1C-1W95MOV	Wall 95 falls towards 480v RMOV 3A and 250vdc RMOV BD 3A	0.010	2.18	0.54	0.27	Block Wall Failure	SoV
SEIS_1B-1	480V BD 1A, 2A, 3A, 1B, 2B, 3B (Fragility Group 02-01)	0.008	1.7	0.32	0.24	Functionality	CDFM
SEIS_1C-1W95SD	Wall 95 falls towards 480v SD BD 3A and 3B	0.008	2.18	0.54	0.27	Block Wall Failure	SoV
SEIS_12-1P-1	RHRSW pumps based on pipe frag (Pipe Calc)	0.007	2.45	0.61	0.47	Soil (Buried Piping) Failure	SoV
SEIS_12-1b	RHRSW pumps and EECW Alternate (Fragility Group 06-03)	0.006	1.45	0.32	0.24	Anchorage	CDFM

†See Section 5.7 for additional discussion on these representative fragilities.

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

LOOP represents the most significant contributor, which is consistent with the results of previous SPRA studies across the nuclear industry that have found that extended LOOP events are dominant for seismic risk. The most important fragility groups for Unit 1 are SEIS_5-2B (Initiation relays and panels) and SEIS_2-1-1 (Unit Batteries (Fragility Group 15-03)). SEIS_5-2B is important because its loss affects many accident initiation signal relays. SEIS_2-1-1 is important because it affects the availability of DC power from the 250V battery boards. For Unit 2 and Unit 3, the most important fragility groups are also SEIS_5-2B and SEIS_2-1-1.

5.5.4 SLERF Component Importance (Non-Seismic Failures)

Components were determined to be risk significant if the component's RAW is greater than two or its F-V is greater than 0.005. Components are considered risk-significant if the component has a F-V value greater than 0.005 or a RAW greater than two for either the SCDF or SLERF importance measures.

One individual component-related basic event related to a main battery test and maintenance term (TM_0BATA2480001="0-BATA-248-0001 (MAIN BATTERY 1) UNAVAILABLE DUE TO T&M") was risk-significant for Unit 1. Table 5.5-7 contains the risk-significant RAW importance measures for each individual component, common cause group, and test and maintenance basic event that appears in the seismic cutsets for Unit 1 SLERF.

One individual component-related basic event related to a main battery test and maintenance term (TM_0BATA2480002="0-BATA-248-0002 (MAIN BATTERY 2) UNAVAILABLE DUE TO T&M") was risk-significant for Unit 2.

Table 5.5-8 contains the risk-significant RAW importance measures for each component, common cause group, and test and maintenance basic event that appears in the seismic cutsets for Unit 2 SLERF.

One individual component-related basic event related to a main battery test and maintenance term (TM_0BATA2480003="0-BATA-248-0003 (MAIN BATTERY 3) UNAVAILABLE DUE TO T&M") was risk-significant for Unit 3. Table 5.5-9 contains the risk-significant RAW importance measures for each component, common cause group, and test and maintenance basic event that appears in the seismic cutsets for Unit 3 SLERF.

Table 5.5-7: Unit 1 SLERF Risk-Significant Individual Component and Common-Cause Group Importance by RAW

Basic Event	Description	RAW
U1_CCFMECH	CCF of mechanical equipment causes failure to scram	21.4
U0_24803BATFR_1_2_3	CCF of three components: BATFR0BATA2480001 & BATFR0BATA2480002 & BATFR0BATA2480003	13.6
FANFR1FAN_0710602	SAI PANEL 9-82 FAN COOLER FAN-71-602 FAILED	4.6
TM_1BDDD2810001A	250V RMOV BD 1A UNAVAILABLE DUE TO TEST AND MAINTENANCE	4.5
TM_0BATA2480001	0-BATA-248-0001 (MAIN BATTERY 1) UNAVAILABLE DUE TO T&M	4.5
BUSFR1BDDD2810001A	250V RMOV BD 1A BUS FAILED	4.4
FUSSO0FU2_2801_111	FUSED SWITCH 111 FAILS	4.4
BATFR0BATA2480001	BATTERY 1 FAILS	4.3
BUSFR0BDDD2800001	BATTERY BD. 1 FAILS	4.3
CBKXO0BKR_2801_110	BREAKER 110 TRANSFERS OPEN	4.2
CBKXO0BKR_2801_202	FEEDER BRK 202 TRANSFERS OPEN DURING OPERATION.	4.2
CBKXO1BKR_2811A_2D	BREAKER 2D TRANSFERS OPEN	4.2
U0_08208DGGFR_ALL	CCF of all components in group 'U0_08208DGGFR'	3.4
U1_07102FANFR_1_2	CCF of two components: FANFR1FAN_0710601 & FANFR1FAN_0710602	3.3
U0_21104CBKFO_ALL	CCF of all components in group 'U0_21104CBKFO'	3.3
U0_03008FANFD_ALL	CCF of all components in group 'U0_03008FANFD'	3.3
U0_03002FANFR1_1_2	CCF of two components: FANFR0FAN_0300072 & FANFR0FAN_0300073	3.3
U0_08208DGGFD_ALL	CCF of all components in group 'U0_08208DGGFD'	3.3
U0_21104CBKFO_1_2_3	CCF of three components: CBKFO0BKR_211A_003 & CBKFO0BKR_211B_002 & CBKFO0BKR_211C_022	3.3
U0_21104CBKFO_1_2_4	CCF of three components: CBKFO0BKR_211A_003 & CBKFO0BKR_211B_002 & CBKFO0BKR_211D_022	3.3
U0_08208DGGFR_1_2_3	CCF of three components: DGGFR0EDG_082__DGA & DGGFR0EDG_082__DGB & DGGFR0EDG_082__DGC	3.0
U0_08208DGGFR_1_2_4	CCF of three components: DGGFR0EDG_082__DGA & DGGFR0EDG_082__DGB & DGGFR0EDG_082__DGD	3.0
U0_03008FANFR_ALL	CCF of all components in group 'U0_03008FANFR'	3.0
U0_08208DGGFD_1_2_3	CCF of three components: DGGFD0EDG_082__DGA & DGGFD0EDG_082__DGB & DGGFD0EDG_082__DGC	2.3
U0_08208DGGFD_1_2_4	CCF of three components: DGGFD0EDG_082__DGA & DGGFD0EDG_082__DGB & DGGFD0EDG_082__DGD	2.3

Table 5.5-8: Unit 2 SLERF Risk-Significant Individual Component and Common-Cause Group Importance by RAW

Basic Event	Description	RAW
U2_CCFMECH	CCF of mechanical equipment causes failure to scram	14.7
U0_24803BATFR_1_2_3	CCF of three components: BATFR0BATA2480001 & BATFR0BATA2480002 & BATFR0BATA2480003	7.4
FANFR2FAN_0710602	SAI PANEL 9-82 FAN COOLER FAN-71-602 FAILED	4.4
TM_0BATA2480002	0-BATA-248-0002 (MAIN BATTERY 2) UNAVAILABLE DUE TO T&M	4.3
TM_2BDDD2810002A	250V RMOV BD 2A UNAVAILABLE DUE TO TEST AND MAINTENANCE	4.3
BUSFR2BDDD2810002A	250V RMOV BD 2A FAILS	3.9
FUSSO0FU2_2802_111	FUSED SWITCH 111 FAILS	3.9
BATFR0BATA2480002	BATTERY 2 FAILS INCLUDING COMMON CAUSE	3.9
BUSFR0BDDD2800002	BATTERY BD. 2 FAILS.	3.9
U0_08208DGGFR_ALL	CCF of all components in group 'U0_08208DGGFR'	2.6
U0_03008FANFD_ALL	CCF of all components in group 'U0_03008FANFD'	2.5
U0_03002FANFR1_1_2	CCF of two components: FANFR0FAN_0300072 & FANFR0FAN_0300073	2.5
U0_08208DGGFD_ALL	CCF of all components in group 'U0_08208DGGFD'	2.4
U0_21104CBKFO_1_2_4	CCF of three components: CBKFO0BKR_211A_003 & CBKFO0BKR_211B_002 & CBKFO0BKR_211D_022	2.4
U0_03008FANFR_ALL	CCF of all components in group 'U0_03008FANFR'	2.3
U2_07102FANFR_1_2	CCF of two components: FANFR2FAN_0710601 & FANFR2FAN_0710602	2.2

Table 5.5-9: Unit 3 SLERF Risk-Significant Individual Component and Common-Cause Group Importance by RAW

Basic Event	Description	RAW
U3_CCFMECH	CCF of mechanical equipment causes failure to scram	13.8
U0_24803BATFR_1_2_3	CCF of three components: BATFR0BATA2480001 & BATFR0BATA2480002 & BATFR0BATA2480003	7.0
TM_0BATA2480003	0-BATA-248-0003 (MAIN BATTERY 3) UNAVAILABLE DUE TO T&M	4.3
FANFR3FAN_0710602	SAI PANEL 9-82 FAN COOLER FAN-71-602 FAILED	4.2
TM_3BDDD2810003A	250V RMOV BD 3A UNAVAILABLE DUE TO TEST AND MAINTENANCE	4.2
FUSSO0FU2_2803_111	FUSED SWITCH 111 FAILS	3.8
BUSFR0BDDD2800003	BATTERY BD. 3 FAILS	3.8
BATFR0BATA2480003	BATTERY 3 FAILS DURING OPERATION INCLUDING COMMON CAUSE	3.8
BUSFR3BDDD2810003A	250V RMOV BD 3A FAILS	3.7
U0_08208DGGFR_ALL	CCF of all components in group 'U0_08208DGGFR'	2.6
U3_07102FANFR_1_2	CCF of two components: FANFR3FAN_0710601 & FANFR3FAN_0710602	2.3
U0_08208DGGFD_ALL	CCF of all components in group 'U0_08208DGGFD'	2.3
CBKXO0BKR_2803_110	BREAKER 110 TRANSFERS OPEN	2.3
CBKXO3BKR_2803_203	FEEDER BREAKER 203 TRANSFERS OPEN DURING OPERATION	2.3
CBKXO3BKR_2813A_2D	BREAKER 2D TRANSFERS OPEN DURING OPERATION	2.3

5.5.5 Significant Human Failure Events

Significant post-initiator operator actions are defined as those operator action basic events that have an F-V value greater than 0.005 or a RAW greater than two. Note that the common methods of calculating RAW for basic events will not yield useful results for the HRA events due to the processing of combination events. This is because the events are set to one during the quantification process, and a recovery event representing the combination or single event is appended to the cutset. Therefore, setting the event to one to determine the RAW value has no effect on SLERF. The F-V values of each operator action were determined in SYSIMP by defining groups where each particular operator action appearing in each seismic HRA bin (S1, S2, S3 and S4) were simultaneously set to FALSE in the combined cutset file to determine the combined importance across all seismic bins. Tables 5.5-10, 5.5-11, and 5.5-12 list the operator actions that were determined to be risk-significant in the seismic model for Unit 1, Unit 2 and Unit 3 SLERF, respectively.

Table 5.5-10: Risk-Significant Operator Actions for Unit 1 SLERF

Operator Action	Description	F-V
HFA_0280ALNALTBBD	Operator fails to align alternate feeder	3.52E-02
HFA_0001HPRVD1	Failure to initiate reactor-vessel depressurization (transient or ATWS)	3.44E-02
HFA_0073MANLEVEL	Operator fails to manually control level with HPCI	3.40E-02
HFA_0001HPRVD2	Operator Fails to Initiate Depressurization (SLERF)	2.67E-02
HFA_0TD2_HPI	Operator fails to manually initiate injection into drywell after core damage	2.67E-02
HFA_OPS_4KVSDBDRESET	Operator reset of 4kV Shutdown Board lockout relays (seismic)	9.30E-03
HFA_0IR2_LPI	Operator fails to manually initiate injection for in-vessel recovery	7.90E-03
HFA_0TD2_LPI	Operator fails to manually initiate injection into drywell after core damage	5.10E-03

Table 5.5-11: Risk-Significant Operator Actions for Unit 2 SLERF

Operator Action	Description	F-V
HFA_0073MANLEVEL	Operator fails to manually control level with HPCI	4.86E-02
HFA_0001HPRVD1	Failure to initiate reactor-vessel depressurization (transient or ATWS)	4.34E-02
HFA_0TD2_HPI	Operator fails to manually initiate injection into drywell after core damage	3.27E-02
HFA_0001HPRVD2	Operator Fails to Initiate Depressurization (SLERF)	3.23E-02
HFA_0280ALNALTBBD	Operator fails to align alternate feeder	3.22E-02
HFA_HINST*	Seismic failure of Main Control Room instrumentation	1.65E-02
HFA_0IR2_LPI	Operator fails to manually initiate injection for in-vessel recovery	8.90E-03
HFA_0TD2_LPI	Operator fails to manually initiate injection into drywell after core damage	5.80E-03
HFA_OPS_4KVSDBDRESET	Operator reset of 4kV Shutdown Board lockout relays (seismic)	5.00E-03
*Although this is not an operator action but an equipment failure, it has an effect similar to that of multiple operator action failures.		

Table 5.5-12: Risk-Significant Operator Actions for Unit 3 SLERF

Operator Action	Description	F-V
HFA_0073MANLEVEL	Operator fails to manually control level with HPCI	3.93E-02
HFA_0001HPRVD1	Failure to initiate reactor-vessel depressurization (transient or ATWS)	3.46E-02
HFA_0280ALNALTBBD	Operator fails to align alternate feeder	3.24E-02
HFA_0TD2_HPI	Operator fails to manually initiate injection into drywell after core damage	2.99E-02
HFA_0001HPRVD2	Operator Fails to Initiate Depressurization (SLERF)	2.78E-02
HFA_OPS_4KVSDBDRESET	Operator reset of 4kV Shutdown Board lockout relays (seismic)	2.52E-02

Operator Action	Description	F-V
HFA_0231480SDBTIE	Failure to transfer 480V shutdown board to alternate source	2.34E-02
HFA_0IR2_LPI	Operator fails to manually initiate injection for in-vessel recovery	7.50E-03
HFA_0TD2_LPI	Operator fails to manually initiate injection into drywell after core damage	5.00E-03

5.5.6 Significant SLERF Accident Sequences

The SPRA evaluated the GTRAN, Direct Core Damage, LLOCA, MLOCA, IOORV, ISLOCA, and ATWS ETs, which were the same ET categories considered in the IEPRA, except for direct core damage, which is unique to the SPRA. The results were dominated by seismically induced LOOP scenarios, which are treated in the GTRAN ET.

An examination of the individual bin results in FRANX shows that the results are dominated by Level 1 accident sequences GTRAN-005, GTRAN-012, GTRAN-011, and GTRAN-001A. The sequence descriptions are given in Table 5.5-13.

Figures 5.5-1, 5.5-2, and 5.5-3 show the relative percentage contribution for each dominant accident sequence based on a consolidated CAFTA cutset file for Unit 1, Unit 2, and Unit 3, respectively. The differences in results between units are primarily the result of power dependency differences for the SPRA.

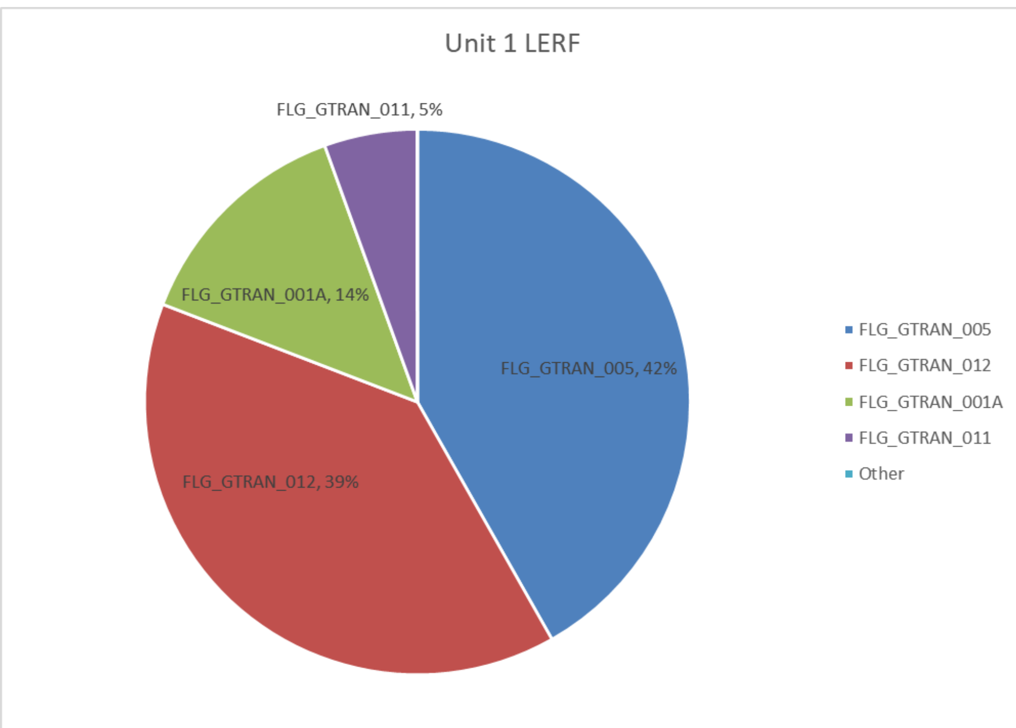

Figure 5.5-1: Unit 1 SLERF Accident Sequence Contribution

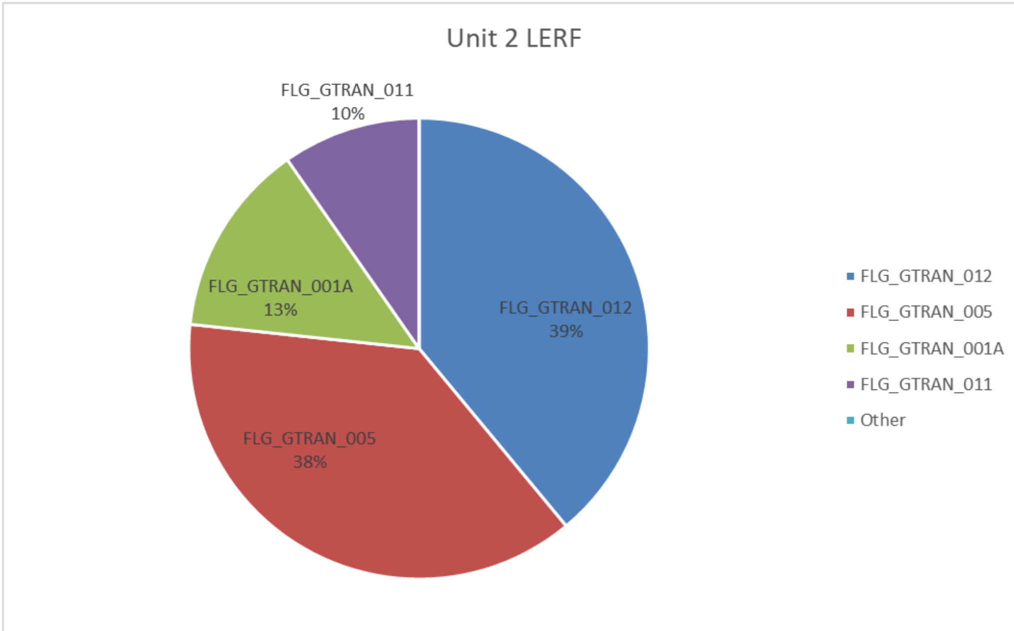

Figure 5.5-2: Unit 2 SLERF Accident Sequence Contribution

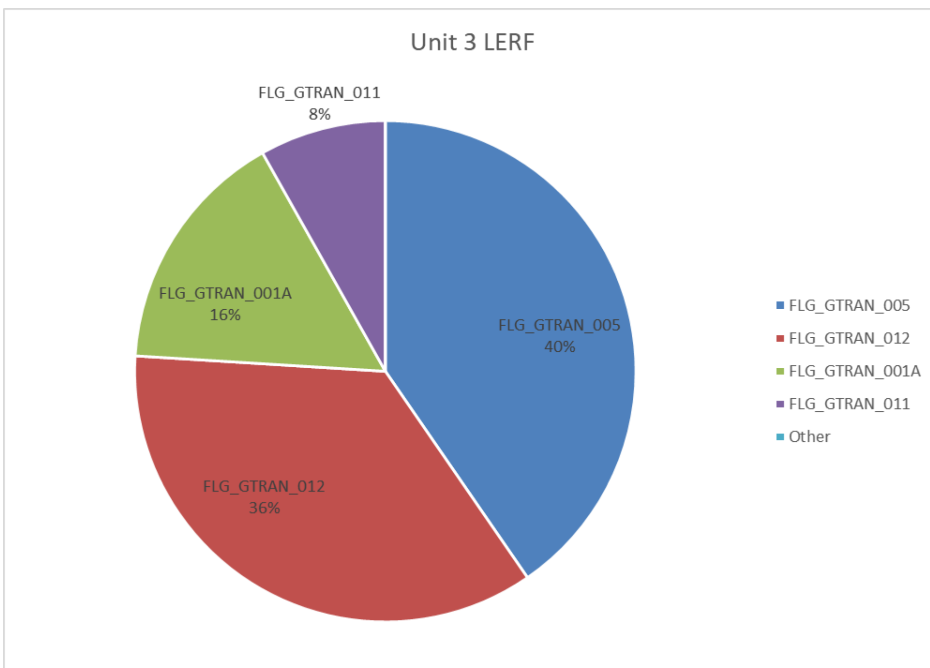

Figure 5.5-3: Unit 3 SLERF Accident Sequence Contribution

Table 5.5-13: Significant SLERF Accident Sequences for Units 1, 2, and 3

Accident Sequence	Description	Discussion
GTRAN-005/ CET1-008	Sequence GTRAN-005 is the same as sequence GTRAN-002 except all low-pressure injections fail. core damage occurs about 1.5 hours after scram with the RPV at low pressure. In sequence GTRAN-002, the scram successfully occurs, the PCS fails, and there are no breaks outside containment or stuck-open relief valves. HPCI or RCIC is successful for at least 4 hours. Early suppression pool cooling is not successful or initiated in time to prevent exceeding HCTL and 190 deg F in the suppression pool. Therefore, long-term HPCI or RCIC is not successful. The CRD and HPMU systems fail, and manual depressurization is challenged and initiated successfully with 2 SRVs at HCTL about 4 hours after the scram. After depressurization, low-pressure injection by RHR in the LPCI mode or CS is successful. ASDC is unsuccessful. Late suppression pool cooling fails allowing the primary containment to pressurize to PCPL in about 13 hours, but the hardened wetwell vent successfully establishes DHR. After the successful vent, the suppression pool no longer provides a successful suction source for RHR or CS, and late injection is not re-established when all the available injection sources fail. These sources are condensate injection, standby coolant injection, shutdown cooling and RHR or CS with suction on the CST. Core damage is caused by loss of injection, and the RPV is at low pressure.	The dominant cutset involves a GTRAN-005 sequence along with a containment isolation failure of 3 inches or greater (CET 1, release state 8 in the ET). This could be caused by both inboard and outboard Main Steam Isolation Valves (MSIVs) failing to close in any of the main steamlines. In the case of the dominant sequence, this is due to failure of fragility group SEIS_5-2B (initiation relays and panels). It could also be caused by failure of high drywell trip signals coupled with an operator action failure to manually close the primary containment isolation valves along with a low reactor trip signal failure.
GTRAN-012/ CET1-CILRT-16	Sequence GTRAN-012 is the same as sequence GTRAN-007 except depressurization fails. Without depressurization, there are no other available injection sources, and core damage occurs. Core damage occurs in about 30 to 40 minutes with the RPV at high pressure. In sequence GTRAN-007, the scram successfully occurs, the PCS fails, and there are no breaks outside containment or stuck-open relief valves. Early HPCI or RCIC are unsuccessful as an initial injection source. A cooldown is initiated but HPMU	The dominant cutset involves a GTRAN-012 sequence. The operator fails to depressurize the RPV because the SRVs fail as a result of loss of nitrogen supply to the X-50 drywell penetration. Also, in-vessel recovery fails due to loss of HPCI and RCIC because the suppression pool cooling path is unavailable. Finally, RPV/drywell injection after core damage fails because of loss of drywell sprays and loss of low-pressure injection. The drywell spray flow path fails because of fragility group SEIS_5-2A2-2 (initiation relays and panels (20-08)) or SEIS_5-2B (initiation relays and panels). Low-pressure

	is unsuccessful. CRD may be available but is not challenged because it lacks sufficient capacity to be used as an initial injection source. When RPV level drops to TAF, manual depressurization is successful. Low-pressure injection by RHR in the LPCI mode or CS is successful as an initial injection source but must be initiated within 30 minutes. ASDC is unsuccessful. Late suppression pool cooling and drywell spray are unsuccessful. HWWV successfully accomplishes DHR. Without SPC, PCPL is reached in approximately 10 hours. Successful venting fails the suppression pool as a suction source, and RHR and CS with suction on the SP are lost. A post-vent injection source of either RHR with suction on the CST, CS with suction on the CST, shutdown cooling, standby coolant injection, condensate injection, or CRD (1 pump) is unsuccessful. CD occurs due to loss of injection, and the RPV fails at low pressure.	injection fails because core spray and LPCI are both lost. Core spray can be lost because the low reactor pressure permissive signal fails (SEIS_5-2B) or the pumps could be lost due to fragility group SEIS_5-2A2-2 (Initiation relays and panels (20-08)). LPCI can be lost because of pump failures due to fragility group SEIS_5-2A2-2 or the reactor low pressure permissive signal fails (SEIS_5-2B).
GTRAN-011/ CET1-CILRT-16	Sequence GTRAN-011 is the same as sequence GTRAN-007 except all low-pressure injections fail. CD occurs in about 30 to 40 minutes with the RPV at low pressure.	Depressurization of the RPV fails because of failure of operator action to initiate depressurization. Also, in-vessel recovery fails due to loss of HPCI and RCIC, which both fail because the suppression pool cooling path is unavailable. The suppression pool cooling is not available because of pump failures due to loss of fragility group SEIS_5-2B (initiation relays and panels). It could also be due to loss of fragility group SEIS_5-2A2-2 (initiation relays and panels (20-08)). There is a failure to inject into the RPV or drywell after core damage occurs because low-pressure injection and drywell sprays both fail. The two loops of drywell spray fail because of either fragility group SEIS_5-2B or SEIS_5-2A2-2. Core spray loop 1 and 2 also fail because of fragility group SEIS_5-2A2-2 or SEIS_5-2B. Both loops of LPCI also fail due to the loss of components in fragility groups SEIS_5-2B or SEIS_5-2A2-2.
GTRAN-001A/CET1	The scram successfully occurs, the PCS fails, and there are no breaks outside containment or stuck-open relief valves. HPCI or RCIC is successful for at least 4 hours. ASDC fails. CRD fails. Low-pressure injection via core spray or LPCI fail.	This sequence involves a GTRAN-001A sequence along with a containment isolation failure of >=3 inches. Core damage occurs for the same reasons as those discussed in the SCDF descriptions for this sequence. The containment isolation failure occurs in the most dominant sequence because of MSIV failures. This is primarily due to initiation relay failures from fragility group SEIS_5-2B.

5.6 SPRA Quantification Uncertainty Analysis

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

NRC RG 1.200 [45] states that an important aspect in understanding the PRA results is knowing the sources of uncertainty and assumptions and understanding their potential impact. They include: (1) parameter uncertainties; (2) model uncertainties and related assumptions; (3) completeness uncertainties; and (4) assumptions related to scope and level of detail.

The scope of the SPRA was limited to the base PRA results and sources of uncertainty for the at-power, Level 1 PRA plus SLERF for seismic events. The focus was also on *epistemic uncertainty* that results from incompleteness; it is noted that a PRA also includes *aleatory uncertainty* that results from randomness. The requirements of PRA applications will be evaluated separately for each application to determine whether sources of uncertainties and assumptions are acceptable. Uncertainties and sensitivities in the IEPRA base model are documented in the Quantification, Uncertainty and Sensitivity Analysis Notebook [44].

5.6.1 Parameter Uncertainty

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

SCDF uncertainty analysis results are summarized in Tables 5.6-1 through 5.6-3 and Figures 5.6-1 through 5.6-3. The uncertainty analysis was performed with UNCERT 4.0, using Monte Carlo sampling with 20,000 samples and ACUBE processing of 200 cutsets. Reference [50] shows that the uncertainty bounds are not especially sensitive to the number of ACUBE cutsets processed.

The uncertainty mean for Unit 1 SCDF is 1.51E-05/ry, compared with the point estimate mean of 6.69E-06/ry. The uncertainty mean is typically larger than the point estimate mean for these types of analyses. The 5th percentile is 7.53E-06/ry, the median is 1.36E-05/ry, and the 95th percentile is 2.77E-05/ry.

The uncertainty mean for Unit 2 SCDF is 1.57E-05/ry, compared with the point estimate mean of 6.78E-06/ry. The uncertainty mean is typically larger than the point estimate mean for these types of analyses. The 5th percentile is 7.72E-6/ry, the median is 1.42E-05/ry, and the 95th percentile is 2.89E-05/ry.

The uncertainty mean for Unit 3 SCDF is 1.72E-05/ry, compared with the point estimate mean of 7.50E-06/ry. The uncertainty mean is typically larger than the point estimate mean for these types of analyses. The 5th percentile is 8.57E-6/ry, the median is 1.58E-05/ry, and the 95th percentile is 3.01E-05/ry.

SLERF uncertainty analysis results are summarized in Tables 5.6-4 through 5.6-6 and Figures 5.6-4 through 5.6-6. The uncertainty analysis was performed with UNCERT 4.0, using Monte Carlo sampling with 20,000 samples and ACUBE processing of 200 cutsets.

The uncertainty mean for Unit 1 SLERF is 6.74E-06/ry, compared with the point estimate mean of 3.20E-06/ry. The uncertainty mean is typically larger than the point estimate mean for these types of analyses. The 5th percentile is 3.39E-06/ry, the median is 6.10E-06/ry, and the 95th percentile is 1.21E-05/ry.

The uncertainty mean for Unit 2 SLERF is 7.27E-06/ry, compared with the point estimate mean of 3.51E-06/ry. The uncertainty mean is typically larger than the point estimate mean for these types of analyses. The 5th percentile is 3.95E-06/ry, the median is 6.64E-06/ry, and the 95th percentile is 1.28E-05/ry.

The uncertainty mean for Unit 3 SLERF is 8.04E-06/ry, compared with the point estimate mean of 4.01E-06/ry. The uncertainty mean is typically larger than the point estimate mean for these types of analyses. The 5th percentile is 4.42E-06/ry, the median is 7.38E-06/ry, and the 95th percentile is 1.39E-05/ry.

The UNCERT analysis included distributions for seismic IEs, seismic fragility estimates, seismic HEPs and combinations, and IEPRA basic events. The seismic bin frequency distributions are presented in Table 5.6-7. Those distributions were generated by the FRANX code assuming lognormal distributions and estimating error factors (EFs) from the various seismic hazard curves input into the code (16th, median, and 84th).

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

Table 5.6-1: Unit 1 SCDF Uncertainty Results

Parameter	Estimate	Confidence Range
Point Est	6.690E-06	
Samples	20000	
Mean	1.513E-05	[1.5E-05 , 1.5E-05]
5%	7.527E-06	[7.4E-06 , 7.6E-06]
Median	1.358E-05	[1.3E-05 , 1.4E-05]
95%	2.773E-05	[2.7E-05 , 2.8E-05]
StdDev	6.964E-06	
Skewness	2.443	
Smp Size @ 10%	81	
Smp Size @ 2%	2034	

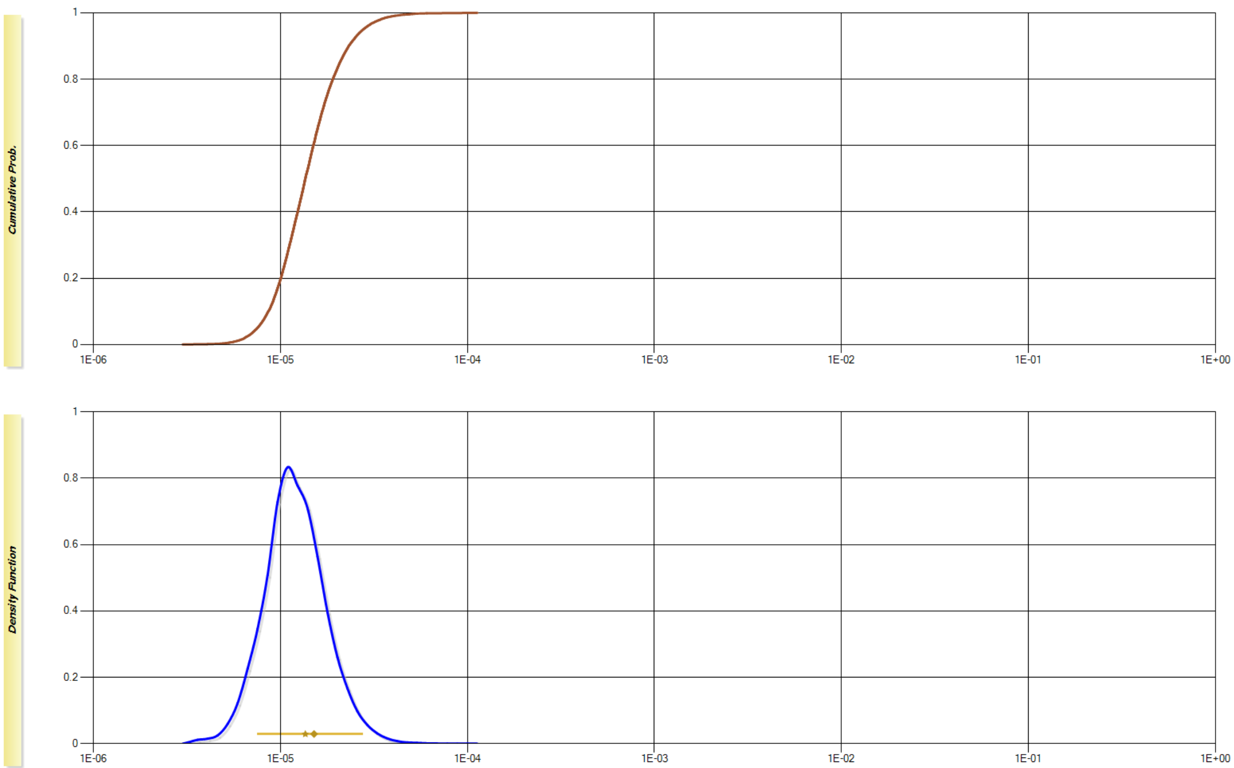

Figure 5.6-1: Unit 1 SCDF Uncertainty Results

Table 5.6-2: Unit 2 SCDF Uncertainty Results

Parameter	Estimate	Confidence Range
Point Est	6.776E-06	
Samples	20000	
Mean	1.573E-05	[1.6E-05 , 1.6E-05]
5%	7.724E-06	[7.6E-06 , 7.8E-06]
Median	1.417E-05	[1.4E-05 , 1.4E-05]
95%	2.886E-05	[2.9E-05 , 2.9E-05]
StdDev	7.122E-06	
Skewness	2.035	
Smp Size @ 10%	79	
Smp Size @ 2%	1968	

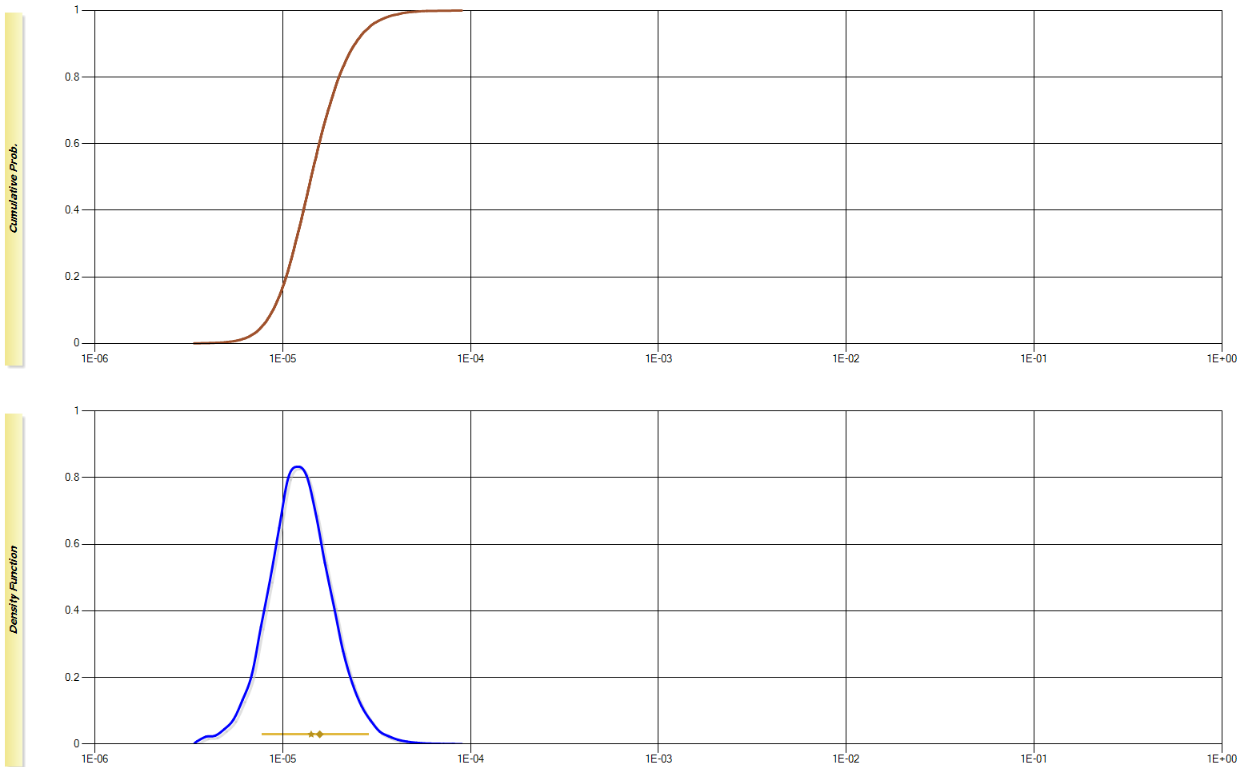

Figure 5.6-2: Unit 2 SCDF Uncertainty Results

Table 5.6-3: Unit 3 SCDF Uncertainty Results

Parameter	Estimate	Confidence Range
Point Est	7.500E-06	
Samples	20000	
Mean	1.715E-05	[1.7E-05 , 1.7E-05]
5%	8.574E-06	[8.5E-06 , 8.7E-06]
Median	1.575E-05	[1.6E-05 , 1.6E-05]
95%	3.014E-05	[3.0E-05 , 3.1E-05]
StdDev	7.344E-06	
Skewness	2.119	
Smp Size @ 10%	70	
Smp Size @ 2%	1761	

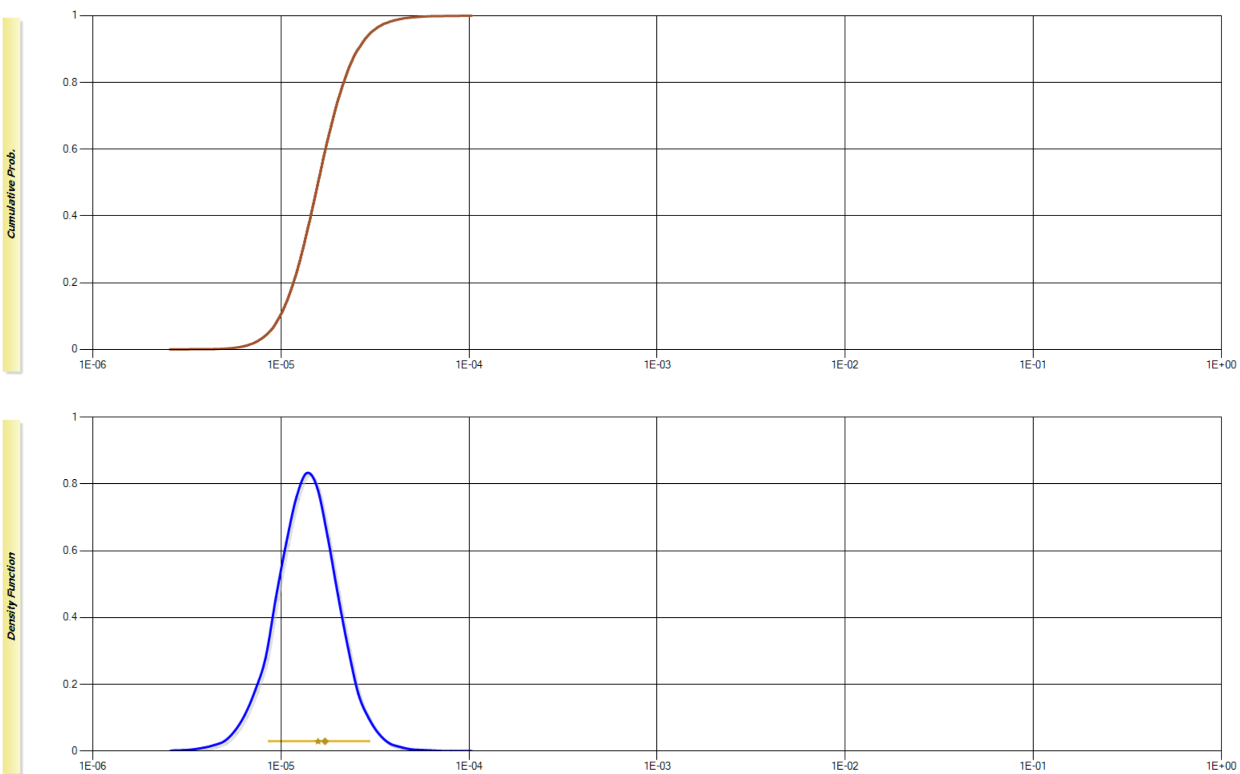

Figure 5.6-3: Unit 3 SCDF Uncertainty Results

Table 5.6-4: Unit 1 SLERF Uncertainty Results

Parameter	Estimate	Confidence Range
Point Est	3.195E-06	
Samples	20000	
Mean	6.741E-06	[6.7E-06 , 6.8E-06]
5%	3.394E-06	[3.4E-06 , 3.4E-06]
Median	6.102E-06	[6.1E-06 , 6.1E-06]
95%	1.205E-05	[1.2E-05 , 1.2E-05]
StdDev	2.991E-06	
Skewness	2.295	
Smp Size @ 10%	76	
Smp Size @ 2%	1891	

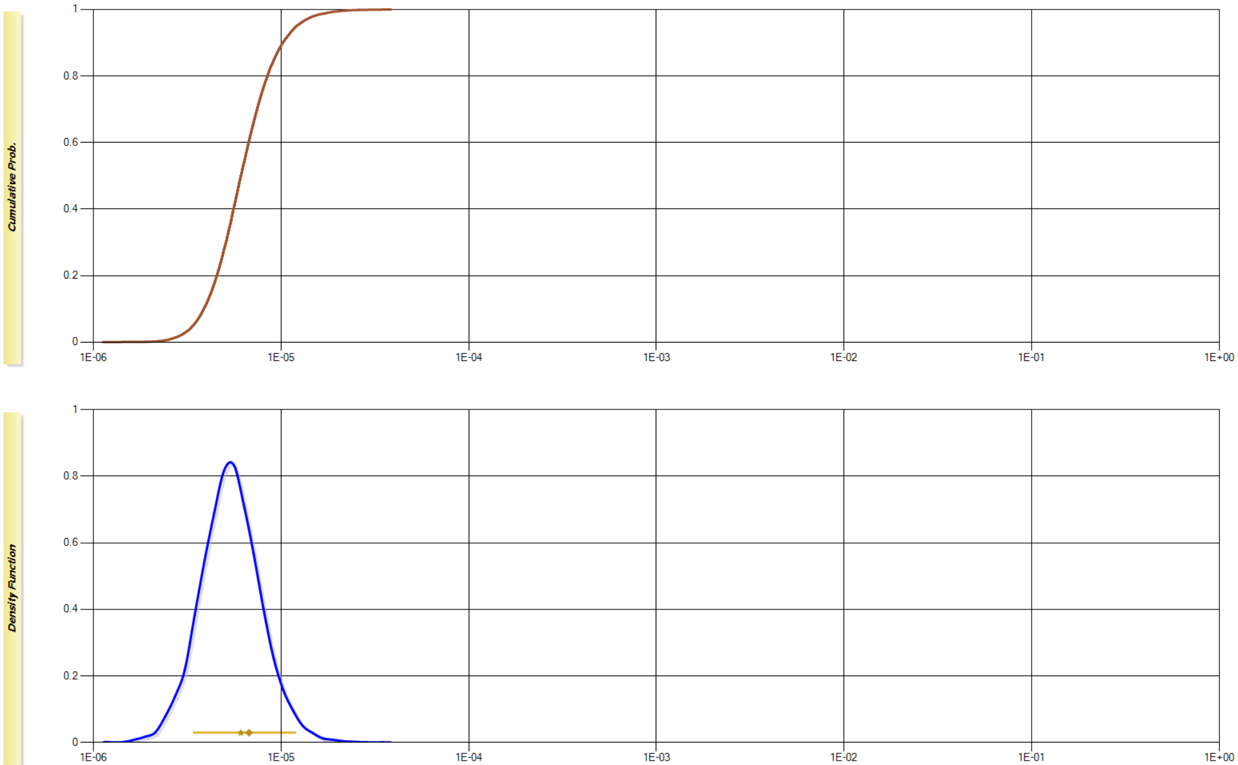

Figure 5.6-4: Unit 1 SLERF Uncertainty Results

Table 5.6-5: Unit 2 SLERF Uncertainty Results

Parameter	Estimate	Confidence Range
Point Est	3.512E-06	
Samples	20000	
Mean	7.268E-06	[7.2E-06 , 7.3E-06]
5%	3.945E-06	[3.9E-06 , 4.0E-06]
Median	6.635E-06	[6.6E-06 , 6.7E-06]
95%	1.275E-05	[1.3E-05 , 1.3E-05]
StdDev	2.971E-06	
Skewness	2.396	
Smp Size @ 10%	64	
Smp Size @ 2%	1605	

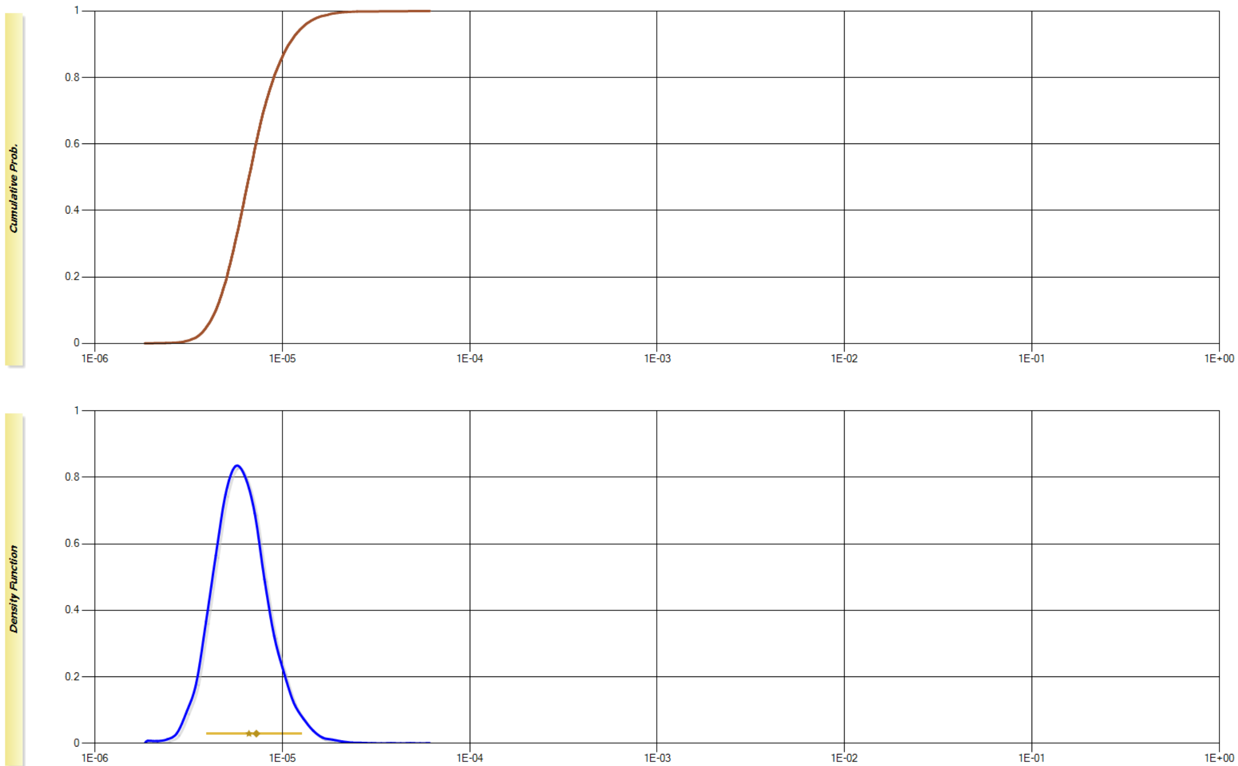

Figure 5.6-5: Unit 2 SLERF Uncertainty Results

Table 5.6-6: Unit 3 SLERF Uncertainty Results

Parameter	Estimate	Confidence Range
Point Est	4.005E-06	
Samples	20000	
Mean	8.043E-06	[8.0E-06 , 8.1E-06]
5%	4.418E-06	[4.4E-06 , 4.5E-06]
Median	7.383E-06	[7.3E-06 , 7.4E-06]
95%	1.391E-05	[1.4E-05 , 1.4E-05]
StdDev	3.164E-06	
Skewness	1.929	
Smp Size @ 10%	59	
Smp Size @ 2%	1486	

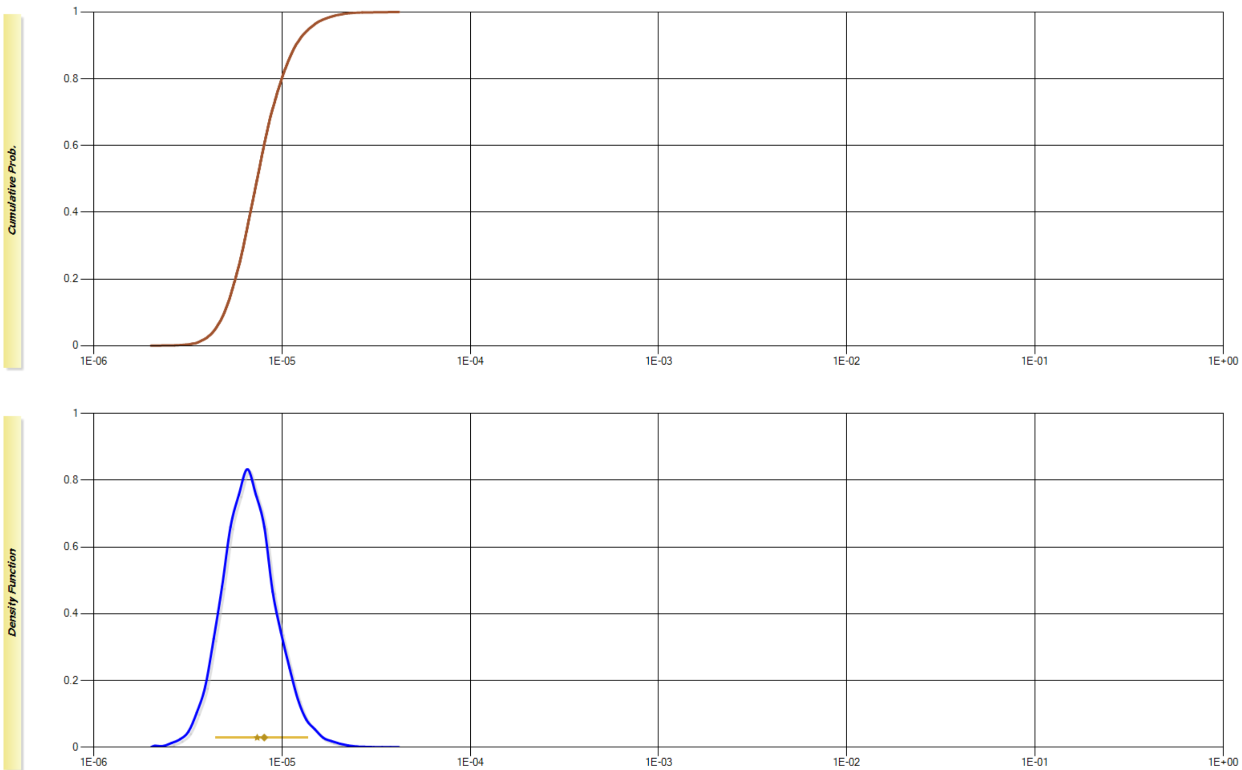

Figure 5.6-6: Unit 3 SLERF Uncertainty Results

Table 5.6-7: Seismic Bin Frequency Distributions

Seismic Bin	Bin PGA (g)	Mean Frequency (/ry)	Error Factor (EF)
%G01	0.14	2.68E-4	9.08
%G02	0.24	5.56E-5	5.60
%G03	0.42	2.98E-5	4.93
%G04	0.65	2.17E-6	4.77
%G05	0.79	2.19E-6	4.97
%G06	1.16	1.82E-6	5.71
%G07	1.60	1.80E-7	6.96
%G08	2.26	3.32E-7	9.13
%G09	3.3	5.96E-8	68.08
Note: Uncertainty in the bin PGA is assumed to be covered in the bin frequency distribution.			

5.6.2 Model Uncertainty

Model uncertainty arises because different approaches exist to represent plant response. A source of model uncertainty is one related to an issue in which no consensus approach or model exists, and where the choice of approach or model is known to have an effect on the SPRA. These uncertainties are typically addressed by making assumptions; e.g., the approach to address CCFs, or the approach to identify and quantify HFEs. In general, model uncertainties are addressed through sensitivity studies using different models or assumptions.

The guidance provided in EPRI 1016737, *Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments* [51], was used to address sources of model uncertainty and related assumptions. It provides a framework for the pragmatic treatment of uncertainty characterization to support risk-informed applications and decision-making. The process includes identification and characterization of sources of model uncertainty and related assumptions; the following sections summarize those in the Level 1 PRA.

5.6.3 Completeness Uncertainty

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

No completeness uncertainties were identified for the BFN SPRA, based on the PRA Standard.

5.6.4 Truncation Study

A truncation study was performed on the Units 1, 2, and 3 SPRA models (SCDF and SLERF) to ensure that sufficient cutsets were generated to result in an accurate estimate for SCDF and SLERF. The truncation study is more complex than typically performed for the IEPRA CDF because of several reasons:

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

Therefore, the truncation study is multi-dimensional. Results of the truncation study are presented in Reference [52], which shows that the BFN SPRA model uses a truncation such that SCDF and SLERF converge for Units 1, 2, and 3.

5.7 SPRA Quantification Sensitivity Analysis

This section presents the results of several sensitivity studies chosen to determine the effects of changing various variables, parameters, or assumptions, most of which are particular to the seismic analysis. Most of these sensitivity studies were done for the Revision 0 (Peer Review) model and were not re-performed for the post-Peer Review model. Sensitivity study #2 is not applicable to the post-Peer Review model since late sequences above %G04 are not considered to be early in the current model.

One new sensitivity was run for the post-Peer Review model (sensitivity #6). In the pre-Peer Review model, SLERF sequences that are normally considered late were considered early for earthquakes in bin %G04 and above. In the post-Peer Review model, the late sequences were not considered early. Sensitivity #6 shows the results for Unit 1 SLERF if late sequences are considered early for seismic bins %G04 and above. The sensitivity cases presented below were not re-run subsequent to the Closure Review.

An additional sensitivity study (Sensitivity #7) is documented in Reference [53] that discusses the effects of increasing the A_m values of certain fragility groups from their original representative non-detailed (i.e., conservative) fragility values since these were risk-significant fragility groups. The fragility groups changed in the sensitivity study analyzed in Reference [53] were SEIS_2-1-1, SEIS_4-5, SEIS_5-4, and SEIS_5-8. These four were selected for improved fragility evaluations due to their importance in SLERF. The additional representative fragilities that show up on the top risk contributor list for U2 SLERF (SEIS_9-1, SEIS_1A-2, SEIS_5-9-1) were not included in this evaluation because these were on the lower end of importance and improvements were judged to not provide significant benefit to the results.

The uncertainties in the assessment of the seismic hazard curve, and of SSC fragilities, are captured in the parameters that define these intermediate results; i.e., by the family of seismic hazard exceedance curves and the parameters for each of the SSC fragilities (A_m, β_r, and β_u). Section 5.6.1 presents the SCDF and SLERF results of the uncertainty analysis, which captured the variation in these risk metrics accounting for these uncertainties.

Sensitivity studies described below are used to investigate other sources of uncertainty, which impact the modeling of seismic impacts and the quantification methods used.

The following areas were investigated:

- Modeling of Seismic Impacts
- Impact of Modeled FLEX HEPs
- Relay Chatter
- HEP Seismic Bin Divisions
- HEP Recovery Actions Added for Seismic

The results for each of the seismic-related sensitivity cases are summarized in Table 5.7-1.

Table 5.7-1: Sensitivity Studies

Case #		Base	Sensitivity	Delta %	Notes
1-No FLEX nitrogen backup to hardened wetwell vent	U1CDF=	5.07E-06	7.23E-06	43%	This sensitivity assumed there was no credit for the backup nitrogen supply to the hardened wetwell vent and the FLEX nitrogen supply to drywell control air by assuming the FLEX nitrogen storage bottles had a high likelihood of failure (SEIS_HLF) and the operator action to align them failed (HFA_OPS_FLEXN2ALIGN=1). It resulted in a relatively large increase in SCDF and a small increase in SLERF.
	U1LERF=	3.53E-06	3.61E-06	2%	
2- Late LERF not early above bin %G04	U1CDF=	5.07E-06	N/A	---	This sensitivity does not affect CDF. It results in a large reduction in SLERF. The gate U1S_LLR_TOP1 was removed from the model.
	U1LERF=	3.53E-06	2.45E-06	-30%	
3-84% hazard curve exceedance values	U1CDF=	5.07E-06	5.14E-06	1%	This sensitivity set the hazard curve mean exceedance frequencies to their 84% percentile values rather than their mean values. There was a small increase in both SCDF and SLERF.
	U1LERF=	3.53E-06	3.59E-06	2%	
4-Redefine break point for EPRI HRA bin S3	U1CDF=	5.07E-06	5.07E-06	-0.03%	This sensitivity re-defined the bin 7 and 8 break points so that the EPRI HRA bin S3 break point with bin S4 could more closely match those used in the Sequoyah SPRA. This resulted in a very small reduction in SCDF and SLERF.
	U1LERF=	3.53E-06	3.48E-06	-2%	
5-No Credit for relay chatter recovery actions	U1CDF=	5.07E-06	8.30E-06	64%	The model was modified to take no credit for operator actions HFA_OPS_4KVSDBDRESET_S* or HFA_OPS_EDGRESET_S* (they were set to 1 for all bins). This resulted in a relatively large increase in SCDF and SLERF.
	U1LERF=	3.53E-06	5.26E-06	49%	

Case #		Base	Sensitivity	Delta %	Notes
6-Late sequences considered early for bin %G04 and above. (post-Peer Review Model)	U1CDF=	6.72E-06	N/A	--	The flag file was modified such that FLG_LATE_SEIS_EARLY was set to 1 rather than FALSE. Also, since there was difficulty in quantifying bin %G07, all the basic events associated with fragility group SEIS_LOOP were set to TRUE for bin 7.
	U1LERF=	3.29E-06	4.16E-06	27%	
7-Sensitivity Study for Representative Fragilities (post-Closure Review Model)	U1CDF=	6.30E-06	6.30E-06	-2.65%	The purpose of this evaluation is to document the sensitivity study and to determine the impact of increasing fragilities of the risk-significant components that have representative fragilities. The increased fragilities are taken from Appendix H of the fragility report [40]. The study showed that an increase in fragilities did not result in a significant increase in SCDF or SLERF.
	U1LERF=	3.00E-06	2.58E-06	-14.24%	

5.8 SPRA Logic Model and Quantification Technical Adequacy

The BFN SPRA risk quantification and results interpretation methodology [44] were subjected to an independent peer review against the pertinent requirements in the PRA Standard. The risk quantification and results interpretation methodology were peer reviewed relative to Capability Category II for the full set of SRs in the PRA Standard. After completion of the subsequent independent assessment, the full set of SRs was met, and the seismic hazard analysis was determined to be acceptable for use in the SPRA.

The peer review assessment and subsequent disposition of peer review findings through an independent assessment are further described in Appendix A, and references [6] and [16].

6.0 Conclusions

A SPRA has been performed for BFN in accordance with the guidance in the PRA Standard and the SPID [2]. The point estimate results of the BFN SPRA are summarized below:

	Unit 1	**Unit 2**	**Unit 3**
Core Damage Frequency	6.30E-06	6.40E-06	7.13E-06
Large Early Release Frequency	3.00E-06	3.10E-06	3.31E-06

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

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

7.0 References

1. NRC Letter to All Power Reactor Licensees et al., "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3 and 9.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," March 12, 2012, ML12056A046.
2. EPRI 1025287, "Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," Electric Power Research Institute, Palo Alto, CA: February 2013.
3. TVA Letter to NRC, "Tennessee Valley Authority's Seismic Hazard and Screening Report (CEUS Sites), Response to NRC Request for Information Pursuant to 10CFR50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," March 31, 2014, ML14098A478.
4. TVA Letter to NRC, "Spent Fuel Pool Evaluation Supplemental Report for Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Response to NRC Request for Information Pursuant to Title 10 of the Code of Federal regulations 50.54(f) Regarding Recommendation 2.1 of the Near-Term, Task Force Review of Insights from the Fukushima Dai-ichi Accident," December 21, 2016, ML16356A596.
5. NRC Letter to TVA, "Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Staff Review of Spent Fuel Pool Evaluation Associated with Reevaluated Seismic Hazard Implementing Near-Term Task Force Recommendation 2.1 (CAC Nos. MF3764, MF3765 and MF3766)," January 27, 2017, ML17024A164.
6. BWR Owners Group, "Browns Ferry Nuclear Plant Seismic PRA Peer Review Report Using the PRA Standard Requirements," Revision 0, June 2019.
7. ASME/ANS RA-S CASE 1, "Case for ASME/ANS RA-Sb-2013 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME and the American Nuclear Society, November 22, 2017.
8. ASME/ANS RA-Sb-2013, "Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME and the American Nuclear Society, June 2013.
9. NEI-12-13, "External Hazards PRA Peer Review Process Guidelines," Revision 0, Nuclear Energy Institute, Washington, DC, August 2012.
10. NRC Letter to NEI, "U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X Guidance 05-04, 07-12, and 12-13, Close-Out of Facts and Observations (F&Os)," ML17079A427, May 3, 2017.

11. Fugro PR No.160029-PR-01, Revision 4, "Probabilistic Seismic Hazards Analysis for TVA Browns Ferry Nuclear Plant PSHA Results Report," October 2019.
12. NUREG-2115, EPRI Product ID 3002005288, "Central and Eastern United States Seismic Source Characterization for Nuclear Facilities," U.S. Nuclear Regulatory Commission. Office of Nuclear Regulatory Research. Washington DC 20555, Sep 29, 2015.
13. NRC Regulatory Guide 1.208, Revision 0, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion," U.S. Nuclear Regulatory Commission, March 2007.
14. NUREG/CR-6728, "Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard and Risk-Consistent Ground Motion Spectra Guidelines," October 2001.
15. EPRI Report 3002004396, "High Frequency Program, Application Guidance for Functional Confirmation and Fragility Evaluation," Electric Power Research Institute, July 2015.
16. Jensen-Hughes Report 006069-RPT-01, Revision 0, "Brown Ferry Seismic PRA Fact and Observation Closure Independent Assessment," November 2019.
17. Report CJC-BFN-C-001, Revision 1, "Updated Soil Failure and Fragility Analysis for the Browns Ferry Nuclear Plant (BFN)," September 2019.
18. EPRI 3002000709, "Seismic PRA Implementation Guide," Electric Power Research Institute, Palo Alto, CA, December 2013.
19. TVA Calculation MDN0009992019000269, Revision 1, "BFN Seismic PRA Seismic Equipment List," September 2019.
20. ENERCON Report TVAEBFN062-REPT-003, Revision 2, "Browns Ferry Units 1-3 Seismic PRA Chatter Analysis Report," September 2019.
21. TVA Letter to NRC, "Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - Generic Letter (GL) 87-02, Supplement 1, Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue A- 46 and GL 88-20, Supplement 4, Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - Submittal of Seismic Evaluation Reports (TAC Nos. M69431, M69432, M83596 and M83589)," June 28, 1996.
22. TVA Letter to NRC "Browns Ferry Nuclear Plant (BFN) - Unit 1 Response to NRC GL 88-20, Supplement 4, Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - Submittal of Browns Ferry Nuclear Plant Unit 1 Seismic and Internal Fires IPEEE," January 14, 2005.
23. Tennessee Valley Authority, CDQ0999960096, Revision 9, "Resolution of A-46/IPEEE Seismic Programs at Browns Ferry Nuclear Plant," February 2016.

24. EPRI NP 6041-SL, Revision 1, “A Methodology for Assessment of Nuclear Power Plant Seismic Margin,” Electric Power Research Institute, Palo Alto, CA, August 1991.
25. TVA Letter to NRC, L44 121127 001, “TVA Response to NRC Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding the Browns Ferry Nuclear Plant Seismic Walkdown Results of Recommendation 2.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident,” November 27, 2012.
26. TVA Letter to NRC, L44 150831 002, “Supplemental Response to NRC Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding the Browns Ferry Nuclear Plant, Unit 3 Seismic Walkdown Results of Recommendation 2.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident,” September 1, 2015.
27. TVA Letter to NRC, L44 130628 001, “Supplemental Response to NRC Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding the Browns Ferry Nuclear Plant, Unit 2 Seismic Walkdown Results of Recommendation 2.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident,” June 28, 2013.
28. TVA Letter to NRC, CNL-14-210, “Tennessee Valley Authority’s Browns Ferry Nuclear Plant Expedited Seismic Evaluation Process Report (CEUS Sites) Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident,” December 22, 2014.
29. SQUG/GIP, Revision 3A, “SQUG Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment,” December 2001.
30. ENERCON Report TVAEBFN062-REPT-001, Revision 1, “Browns Ferry Nuclear Plant Seismic Walkdowns,” October 2019.
31. SC Solutions Report BFN-17-001, Revision 2, “Browns Ferry Nuclear Plant Seismic Probabilistic Risk Assessment: Structural Response Analysis,” September 2019.
32. ASCE/SEI 4-98., “Seismic Analysis of Safety-Related Nuclear Structures,” American Society of Civil Engineers, 1998.
33. ASCE/SEI Standard 4-16, “Seismic Analysis of Safety-Related Nuclear Structures,” American Society of Civil Engineers, 2016.
34. ASCE/SEI 43-05, Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities, American Society of Civil Engineers and Structural Engineering Institute, 2005.
35. EPRI TR-103959, “Methodology for Developing Seismic Fragilities,” Electric Power Research Institute, Palo Alto, CA, June 1994.

36. EPRI TR-1019200, “Seismic Fragility Application Guide Update,” Electric Power Research Institute, Palo Alto, CA, December 2009.
37. EPRI 3002012994, “Seismic Fragility and Seismic Margin Guidance for Seismic Probabilistic Risk Assessment,” Electric Power Research Institute, Palo Alto, CA, September 2018.
38. ENERCON Report TVAEBFN062-REPT-002, Revision 2, “Browns Ferry Nuclear Components and Structures Fragility Evaluation,” December 2019.
39. NUREG/CR-4840, “Procedure for the External Event Core Damage Frequency Analyses for NUREG-1150,” November 1990.
40. NUREG/CR-6544, “A Methodology for Analyzing Precursors to Earthquake-Initiated and Fire-Initiated Accident Sequences,” April 1998.
41. TVA Calculation MDN0009992019000266, Revision 1, “BFN Seismic PRA Human Reliability Analysis,” October 2019.
42. TVA Calculation MDN0009992019000258, Revision 1, “BFN Seismic PRA Quantification, Sensitivity and Uncertainty Notebook,” November 2019.
43. Regulatory Guide 1.200, Revision 2, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” U.S. Nuclear Regulatory Commission, March 2009.
44. PRA Evaluation BFN-0-18-116, Revision 1, “BFN Seismic PRA Seismic-Fire Interaction,” March 2019.
45. EPRI 3002012980, “Methodology for Seismically Induced Internal Fire and Flood Probabilistic Risk Assessment,” June 2018.
46. TVA Calculation NDN00099920070031, Revision 2, “BFN Probabilistic Risk Assessment – Internal Flooding Analysis,” October 2018.
47. EPRI 3002008093, “An Approach to Human Reliability Analysis for External Events with a Focus on Seismic,” December 2016.
48. PRA Evaluation BFN-0-19-065, Revision 0. “BFN SPRA Uncertainty Analysis to Address F&O 25-3,” May 2019.
49. EPRI 1016737, “Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments,” Electric Power Research Institute, Palo Alto, CA, December 2008.
50. PRA Evaluation BFN-0-19-022, Revision 1. “Convergence of the BFN Seismic PRA Model,” July 2019.
51. PRA Evaluation BFN-0-19-115, Revision 0. “Updated Fragility Estimates for Selected Components,” November 2019.
52. BWR Owners Group Report “BROWNS FERRY UNITS 1,2,3 PRA Peer Review Report Using ASME PRA Standard Requirements,” August 2009.

53. NEI 05-04, “Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard,” Revision 2, Nuclear Energy Institute, Washington, DC, November 2008.

54. ABS Consulting, “Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard,” October 2009.

55. ASME/ANS RA-Sa-2009, “Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,” ASME and the American Nuclear Society, 2009.

56. P3146-0001-02 Revision 0, Focused Scope Peer Review of the Browns Ferry Nuclear Power Plant (BFN) Internal Flood PRA Model Against the ASME PRA Standard Requirements, October 2018.

57. P3146-1000-01, Revision 0, “F&O Closeout by Independent Assessment Report for the Browns Ferry Nuclear Plant (BFN) Internal Events PRA Model Against the ASME PRA Standard Requirements and NEI 05-04 Appendix X,” November 2018.

58. P0132150002-5175, Revision 0, “BFN PRA Focused Scope Peer Review Final Report,” August 31, 2015.

59. NUREG-1855, Revision 1, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making,” March 2017.

60. TVA Calculation NDN00099920100001, Revision 8, “BFN Probabilistic Risk Assessment – Summary Document,” February 2018.

8.0 Acronyms and Abbreviations

AFE	Annual Frequency of Exceedance
Am	Median Acceleration Capacities
ANS	American Nuclear Society
AOV	Air-Operated Valve
ASCE	American Society of Civil Engineers
ASDC	Alternate Shutdown Cooling
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BE	Best Estimate
BFN	Browns Ferry Nuclear Plant
BWROG	Boiling Water Reactor Owners' Group
CAD	Containment Air Dilution CB Control Bay
CCDP	Conditional Core Damage Probability
CCF	Common Cause Failure
CDF	Core Damage Frequency
CDFM	Conservative Deterministic Failure Model
CET	Containment Event Tree
CEUS	Central and Eastern United States
CEUS-SSC	Central and Eastern United States Seismic Source Characterization
CG	Center of Gravity
CLERP	Conditional Large Early Release Probability
CR	Center of Rigidity
CRD	Control Rod Drive
CS	Core Spray
CST	Condensate Storage Tank
DCA	Drywell Control Air
DGB	Diesel Generator Building
DHR	Decay Heat Removal
DM	Direct Method
DOE	Department of Energy
Dp	Compression-wave Damping

Ds	Shear-wave Damping
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EECW	Emergency Equipment Cooling Water
EF	Error Factor
EL	Elevation
EPRI	Electric Power Research Institute
ESEP	Expedited Seismic Evaluation Process
ET	Event Tree
FEM	Finite Element Model
F&O	Facts and Observations
FIRS	Foundation Input Response Spectra
FLEX	Diverse and Flexible Coping Strategies
FPIE	Full Power Internal Events
FSAR	Final Safety Analysis Report
FSPR	Focused-Scope Peer Review
F-V	Fussell-Vesely
GERS	Generic Equipment Ruggedness Spectra
GIP	Generic Information Procedure
GMC	Ground Motion Characterization
GMPE	Ground Motion Prediction Equation
GMRS	Ground Motion Response Spectra
GTRAN	General Transient
HEP	Human Error Probability
HCLPF	High Confidence of Low Probability of Failure
HCTL	Heat Capacity Temperature Limit
HCV	Hardened Containment Vent
HF	High Frequency
HFE	Human Failure Event
HLR	High-Level Requirements
HPMU	High Pressure Make-Up
HRA	Human Reliability Analysis
HVAC	Heating, Ventilating, and Air Conditioning

IE	Initiating Event
IEPRA	Internal Events Probabilistic Risk Assessment
IOORV	Inadvertent Opening of One Relief Valve
IPEEE	Individual Plant Examination for External Events
IPS	Intake Pumping Station
ISLOCA	Interfacing System LOCA
ISRS	In-Structure Response Spectra
JCNRM	Joint Committee on Nuclear Risk Management
LAR	License Amendment Request
LB	Lower Bound
LERF	Large Early Release Frequency
LF	Low Frequency
LLOCA	Large LOCA
LOCA	Loss-of-Coolant Accident
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection
LMSM	Lumped Mass Stick Model
MAFE	Mean Annual Frequency of Exceedance
MCC	Motor Control Center
MLOCA	Medium LOCA
MOV	Motor-Operated Valve
MSIV	Main Steam Isolation Valve
MSL	Mean Sea Level
MSM	Modified Subtraction Method
NEI	Nuclear Energy Institute
NRC	United States Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NTTF	Near Term Task Force
OBE	Operating Basis Earthquake
PCS	Power Conversion Systems
PDS	Plant Damage States
PGA	Peak Ground Acceleration
PRA	Probabilistic Risk Assessment

PSHA	Probabilistic Seismic Hazard Analysis
RAW	Risk Achievement Worth
RB	Reactor Building
RE	Reference Earthquake
RG	Regulatory Guide
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RLE	Review Level Earthquake
RLGM	Review Level Ground Motion
RLME	Repeated Large Magnitude Earthquakes
RMOV	Reactor Motor-Operated Valve
SASSI	System for Analysis for Soil-Structure Interaction
SAF	Site Amplification Factor
SEL	Seismic Equipment List
SEWS	Screening and Evaluation Walkdown Sheets
SFR	Seismic Fragility Element Within ASME/ANS PRA Standard
SHA	Seismic Hazard Analysis Element Within ASME/ANS PRA Standard
SIET	Seismic Initiating Event Tree
SLOCA	Small LOCA
SoV	Separation of Variables
SPID	Screening, Prioritization and Implementation Details
SPR	Seismic PRA Modeling Element Within ASME/ANS PRA Standard
SPRA	Seismic PRA
SQUG	Seismic Qualification Utility Group
SR	Supporting Requirement
SRT	Seismic Review Team
SRV	Safety Relief Valve
SSC	Structures, Systems and Components
SSE	Safe Shutdown Earthquake
SSEL	Safe Shutdown Equipment List
SSHAC	Senior Seismic Hazard Analysis Committee
SSI	Soil-Structure Interaction
TAF	Top of Active Fuel

TB	Turbine Building
TH	Time History
TVA	Tennessee Valley Authority
UB	Upper Bound
UHRS	Uniform Hazard Response Spectrum
USI	Unresolved Safety Issue
V/H	Vertical over Horizontal
Vp	Compression-wave Velocity
Vs	Shear-wave Velocity
VSLOCA	Very Small LOCA

Appendix A

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

A.1 Introduction

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

A.2 Peer Review of BFN SPRA

The BFN SPRA was performed in three phases. A pre-visit review, a one-week onsite review, and a post-review phase. The pre-visit consisted of an independent review of SPRA documentation with and exchange of questions and answers between the BFN SPRA team and the peer reviewers. The onsite phase of the BFN SPRA Peer Review was performed May 6 through May 10, 2019, at the Browns Ferry Nuclear Plant (BFN) training center in Athens, AL. As part of the peer review, a walkdown of portions of BFN Units 1, 2, and 3 was performed on May 7, 2019, by several members of the peer review team who have the appropriate SQUG training. The post-review time period was used to resolve comments and perform technical editing on the final peer review report.

The information presented here establishes that the SPRA has been peer reviewed by a team with adequate credentials to perform the assessment, establishes that the peer review process followed meets the intent of the peer review characteristics and attributes in Table 16 of RG 1.200 Rev. 2 [45] and the requirements in Section 1-6 of the PRA Standard [8], and presents the significant results of the peer review.

A.2.1 Summary of the BFN Seismic PRA Peer Review Process

The BFN SPRA Peer Review was performed using the NEI 12-13 process and the PRA Standard (ASME/ANS RA-Sb-2013, Code Case 1). The purpose of this review was to establish the technical adequacy of the BFN SPRA (Units 1, 2, and 3) for the spectrum of potential risk-informed plant applications for which the SPRA may be used. The 2019 BFN SPRA Peer Review was a full-scope review of all the technical elements of the BFN at-power SPRA (model files provided in April 2019) against all technical elements in ASME/ANS RA-Sb-2013, Code Case 1.

The peer review team consisted of eight team members with extensive qualifications in all areas of SPRA as required by NEI 12-13. The team members' experience averaged more than 20 years in PRA, seismic hazard and fragility analyses, with extensive experience in SPRA and the SPRA Section of the PRA Standard. Team member experience is discussed further in Section A.2.2, with resumes provided in Appendix D of the peer review report [6].

The peer review was performed against the requirements in the PRA Standard (ASME/ANS RA-Sb-2013, Code Case 1), using the peer review process defined in NEI 12-13 [9]. The review was conducted over a four-day period, with a summary and exit meeting on the morning of the fifth day.

The SPRA peer review process defined in NEI 12-13 involves an examination by each reviewer of their assigned PRA technical elements against the requirements in the PRA Standard to ensure the robustness of the model relative to all the requirements.

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

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

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

A.2.2 Peer Review Team Qualifications

The review was conducted by Mr. Paul Amico of Jensen Hughes, Mr. Russell Childs of Duke Energy, Mr. Eddie Guerra of ARUP, Mr. Jeffrey Kimball of Rizzo International, Mr. Lawrence Lee of Jensen Hughes, Dr. Glenn Rix of Geosyntec, Mr. Habib Shtaih of Energy Northwest, and Mr. Philip Tarpinian of Exelon. In addition, support was provided by Dr. Mayasandra Ravindra of MK Ravindra Consulting. The working observer was Ms. Coreen McDonald of American Electric Power (12 years of experience (7 years fragility)), who supported SFR.

Mr. Paul Amico, the team lead, is a nuclear engineer with over 40 years of experience in the performance and management of domestic and international programs involving risk and reliability technology and its application to the design and operation of nuclear plants. He has been involved in Seismic PRA since 1981 and is currently very active in Seismic PRA standards development and the performance of Seismic PRAs.

Dr. Glenn Rix was the lead for the Seismic Hazard Analysis (SHA) technical element. He has 30 years of experience in geotechnical earthquake engineering and engineering seismology (particularly for the eastern and central U.S.), and seismic hazard assessment and risk mitigation for civil infrastructure. Dr. Rix is an expert in site characterization; site response analysis; and secondary hazards such as liquefaction-induced ground deformation. He was assisted by Mr. Jeffrey Kimball. Mr. Kimball has 39 years of experience with the evaluation and characterization of natural phenomena hazards and the design of critical facilities to resist these hazards. He led the preparation of Department of Energy (DOE) standards and guides to define requirements and procedures to complete assessment of natural phenomena hazards and has full knowledge of a wide range of nuclear facility regulations, regulatory guides, standards, manuals, guides and review plans associated with nuclear facility design and evaluation. Mr. Kimball is a recognized expert in site characterization; ground motion modeling including site response and probabilistic seismic hazard analysis (PSHA) including guidance for completing PSHA.

Mr. Eddie Guerra was the lead for the seismic fragility analysis (SFR) technical element. Mr. Guerra has nine years of experience in seismic engineering, including seismic risk assessments for nuclear plants. This includes Seismic Equipment List (SEL) development, building analysis, equipment fragility, and walkdowns. He was assisted by Mr. Rusty Childs. Mr. Childs is the Lead Nuclear Engineer for the Oconee Nuclear Station and has over 30 years of experience in civil engineering and seismic qualification projects in the nuclear industry. Mr. Childs's experience in support of Seismic PRA and margin assessments includes the fragility and walkdown scopes for A46/IPEEE program, NTTF 2.3 Walkdowns, ESEP and 2.1 NTTF submittal for Duke Energy's Oconee Nuclear Station. Mr. Childs is the Chairman of the EPRI Seismic Qualification Utility Group (SQUG), where he oversees the application of the SQUG database and methodology for U.S. utilities as well as international vendors. Assistance was also provided Dr. Mayasandra Ravindra, who was originally scheduled to be the SFR lead. He completed the pre-visit review but was unable to attend.

Mr. Lawrence Lee was the lead for the Seismic Plant Response (SPR) technical element. Mr. Lee has a degree in Mechanical Engineering and over 27 years of experience in the risk assessment area. Mr. Lee has contributed to, led and reviewed numerous nuclear power risk assessments (Level 1, Level 2, all modes, internal and external events), as well as numerous other risk-related projects. He has experience in the performance and peer review of SPRAs. He is a Principal Engineer with Jensen Hughes. Mr. Lee was assisted by Mr. Habib Shtaih and Mr. Philip Tarpinian. Mr. Shtaih has 12 years of experience, including the performance of most major elements of Level 1 internal events PRA as well as various applications such as MSPI, SDP and MOV ranking. He is currently the technical lead for the SPRA for the Columbia Generating Station. Mr. Tarpinian has 36 years of experience in the nuclear field in engineering and PRA positions, all at the Limerick Station and Exelon corporate nuclear offices. His work since 2012 has been focused on Exelon's post-Fukushima (10CFR50.54f) evaluations regarding seismic, external flooding, FLEX and containment venting.

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

A.2.3 Summary of the Peer Review

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

A.2.3.1 Seismic Hazard Analysis (SHA)

As required by the PRA Standard, the frequency of occurrence of earthquake ground motions at the site was based on a probabilistic seismic hazard analysis (PSHA). The seismic source characterization inputs to the PSHA are based on the Central and Eastern U.S. (CEUS) regional seismic source characterization model published in NUREG-2115 (i.e., the "CEUS-SSC" model). The ground motion characterization (GMC) inputs to the PSHA are based on an updated CEUS ground motion model published by EPRI (Reference 8). The seismic hazard analysis for BFN also accounts for the effects of local site response for those structures, systems, and components that are not founded on CEUS reference rock (i.e., "hard rock"); site response analyses were performed to calculate Ground Motion Response Spectra (GMRS) and Foundation Input Response Spectra (FIRS) at several elevations (EL): EL 515 ft (GMRS/FIRS1, base of reactor building (RB)); EL 556 ft (FIRS2, base of the diesel generator building (DGB)); EL 515 ft (FIRS3, base of the intake pumping station (IPS)); and EL 565 ft (FIRS4, base of the Yard equipment).

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

The PRA Standard requires compilation of an up-to-date database, including regional geological, seismological, geophysical data, local site topography, and information on surficial geologic and geotechnical site properties. These data include a catalog of relevant historical, instrumental, and paleoseismic information. The CEUS-SSC study involved an extensive data collection effort that satisfies the requirements of the PRA Standard as it relates to developing a regional-scale seismic source model. In the implementation of the CEUS-SSC model for the BFN site, all distributed (i.e., background) and repeated large-magnitude earthquake (RLME) seismic sources in the

CEUS-SSC model were included in the PSHA calculations. By including these seismic sources in the analysis, the contributions of all credible sources of potentially damaging earthquakes to ground motions at the BFN site were considered. Likewise, the EPRI (Reference 8) GMC study involved an extensive data collection effort that satisfies the requirements of the PRA Standard as it relates to developing a GMC model to estimate the range of vibratory ground motion that may occur at the BFN site.

The CEUS-SSC and EPRI regional models discussed above are existing models, and the seismicity database that underpins significant aspects of the CEUS-SSC only includes earthquakes through 2008. According to the PRA Standard, if an existing model is used, a data collection and evaluation effort should be conducted to determine (1) whether new information has become available since the data was compiled for the existing model and, if so, (2) whether any new information challenges the validity of the technical basis of the existing study. It is not the case that identification of new data automatically requires an update to the PSHA existing model. Rather, an evaluation of the new data determines whether the existing model is appropriate for its continued use in the intended application. In the case of the PSHA for the BFN site, the analysts developed an updated seismicity catalog that was quantitatively assessed to ensure that (1) assumptions regarding the distribution of the maximum magnitude are not violated and (2) no new data exists that undermines the earthquake recurrence parameters of sources in the CEUS-SSC model important to the seismic hazard at the BFN site. The analysts also performed a literature review to confirm that no new (since 2012) information is available that may impact the CEUS-SSC model.

The PRA Standard requires consideration of all sources that can potentially cause important vibratory ground motion at the BFN site, including non-tectonic, human-induced earthquakes. The CEUS-SSC model used to assess vibratory ground motions explicitly removes non-tectonic earthquakes, which is appropriate because the underlying causation is different from tectonic earthquakes and is non-stationary (i.e., it may change over relatively short time periods). However, human-induced and other non-tectonic seismicity (e.g., earthquakes from wastewater injection and reservoir-induced seismicity) can produce damaging ground motions in some cases. For the BFN site, a separate seismicity catalog of non-tectonic earthquakes was compiled and evaluated via a screening process. Because of the very low magnitude of non-tectonic earthquakes within 400 km of the site, it was concluded that no revisions to the CEUS-SSC model were required to account for nontectonic seismicity.

The PSHA results are provided over an appropriately wide range of spectral frequencies and annual frequencies of exceedances (AFEs). Uncertainties on the reference rock hazard are quantified, analyzed and reported, as required in the standard. The lower-bound magnitude chosen for the analysis is consistent with standard practice. The results include fractile and mean hazard curves, median and mean uniform hazard response spectra (UHRS), and deaggregation results by magnitude and distance and by seismic source.

The SHA for the BFN site included a site response analysis for structures, systems and components not founded on reference hard rock. As noted previously, GMRS and FIRS were developed for several elevations within the plant. As part of the characterization of the site, an extensive geophysical study was performed in 2016 to evaluate the shear-wave velocity of the rock material below EL 515 ft at the site to inform the site response analysis. The study utilized state-of-the-art methods, including active and passive surface wave methods, vertical-over-horizontal (V/H) spectral ratios, first arrival information, and sonic logs to constrain the interpretation of the data via joint inversion, resulting in a Best Practice. The site response analysis included the effects of site topography, surficial geologic deposits, and site geotechnical properties on ground motions at the site. However, the data and information related to the various site materials above the sedimentary rock (EL 515 ft) were not compiled in an integrated effort, leading to inconsistencies between analyses for site response, soil-structure interaction (SSI), and soil failure.

Both the aleatory and epistemic uncertainties have been addressed in characterizing the seismic sources, ground motion models, and site response analyses. For the SSC and GMC, the characterization of uncertainties is included in NUREG-2115 (Reference 9) and EPRI (Reference 8), respectively. For the site response component, epistemic uncertainty is represented by three shear-wave velocities and two sets of modulus reduction and damping curves for firm rock. Aleatory variability is represented by 60 random realizations of each profile, including random variations in shear-wave velocity and modulus reduction and damping curves. In general, the parameters selected to model each type of uncertainty are consistent with values recommended in EPRI (Reference 8). Correlation between properties is modeled when appropriate.

The reference hard rock hazard calculations are based on the CEUS-SSC and EPRI GMC models. During the development of these models, uncertainties in the seismic sources and ground motion prediction equations were included, and appropriate sensitivity analyses were performed to demonstrate the sensitivity of the results to uncertainties in key model parameters. These sensitivity analyses are also documented in the associated reports. In addition, the PSHA analysts for BFN performed site-specific sensitivity analyses for the ground motion prediction equations (GMPEs), magnitude completeness, earthquake recurrence rate, and maximum magnitude. A sensitivity analysis was performed to evaluate the potential impacts of implementing the Next Generation Attenuation-East ground motion model at BFN. Sensitivity analyses were also performed to evaluate the sensitivity of the calculated soil amplification factors and/or surface hazard curves to variations in site response parameters, including alternative profile randomization schemes and the presence/absence of a soft layer near the bottom of the Fort Payne formation.

The PRA Standard requires that spectral shapes be based on a site-specific evaluation that considers the contributions of deaggregated magnitude-distance results of the PSHA. The horizontal UHRS used in the SPRA is based on site-specific results and incorporates analysis results for spectral frequencies ranging from 0.5 to 100 Hz (i.e., PGA). A Finding under SHA-G1 was developed due to the lack of a sufficient technical explanation and justification for how the high-frequency portion (25 to 100 Hz) of the reference rock spectrum was calculated. Appropriate V/H ratios were used to calculate vertical response spectra.

SR SHA-I1 addresses the bases and methodology used for any screening of the seismic hazards other than vibratory ground motion. The purpose of SHAI1 is to ensure that a screening analysis is performed using a structured approach to ensure that all possible secondary (or “other” seismic-related) hazards are identified and appropriately dispositioned. Many secondary hazards screened out for BFN, but soil failure modes related to liquefaction-induced settlement and lateral spreading are evaluated further as part of addressing SR SHA-I2, using an approach to estimate the magnitude of displacement and associated uncertainty that is based on local soil conditions. A Finding was developed regarding the need to more robustly model uncertainties with respect to liquefaction-induced settlement and lateral spreading to provide more confidence in the resulting fragilities.

The PRA Standard requires that documentation of the PSHA that supports the PRA applications, peer review and potential future upgrades of the SHA be provided. This requirement establishes a high standard for documentation of the PSHA that allows for examination of the PSHA methodology, its implementation, and the PSHA results to evaluate whether the approach is appropriate, the analyses were performed correctly, and the results are reasonable. Overall, the PSHA documentation is very good and fulfills these requirements. Items that must be addressed to improve the documentation on specific topics and items that should be addressed to improve the readability of the documentation are included in Findings linked to SHA-J1.

A.2.3.2 Seismic Fragility Analysis (SFR)

The SFR assessment of BFN SPRA covered three principal elements of the fragility analysis: building seismic response analysis, plant walkdowns, and fragility analysis calculations. These three elements are briefly summarized below.

The building seismic response analyses of the BFN structures that feed into the fragility evaluations are based on input response spectra corresponding to the control point (top of rock) UHRS with AFE of 10-5 anchored to a horizontal PGA of 0.52g. BFN developed soil-structure interaction (SSI) models for all the buildings and structures included in the PRA using the computer code SC-SASSI. The modeling was done following the recommendations of ASCE/SEI 4-16. The model configurations and properties were based on available drawings, seismic analysis reports, and other documents. The peer review team concurs that the structural models are realistic.

BFN performed deterministic SSI analyses using five (5) combinations of soil-structural properties varying from a median-centered (Best Estimate) model. Median and 84th percentile structural responses were obtained from multiple deterministic analyses. The analyses were performed for three (3) structures: RB, IPS and the DGB. Approximations were made to model a single representative RB SSI and two separate DGB+RB models in lieu of a combined 2XDGB+3XRB model. The selection of the representative Unit 1 RB structure among all three RB units is well-justified based on seismic response sensitivity studies.

As part of the building response analysis task, the BFN team investigated the effect of potential impact between the Turbine Building (TB) and the RB structure. A coupled finite element model was developed to assess the impact effects between TB and RB structures, capturing the differing foundation conditions of both the RB (rock-founded) and TB (pile- supported in soft soil), and with nonlinear impact elements between the RB and TB. The analysis results demonstrated that the impact would not result in significant building structural damage, nor meaningful shock response across floors in the RB that would propagate into cabinets housing vibration-sensitive devices such as relays. Nominal impact-induced vibration effects were included in subsequent fragility evaluations.

The walkdown effort included representatives from the fragility team and the systems modeling team. During the walkdowns, the team verified that SSCs met the high-capacity and inherently rugged capacity criteria.

- Inherently rugged items were judged by the Seismic Review Team (SRT) to have sufficiently high seismic capacity such that they should have a negligible contribution to seismic risk.
- High-capacity items were judged by the SRT to have a HCLPF capacity of at least 2g. A fragility curve with 2g HCLPF capacity convolved with the hazard curve yields a point estimate SCDF of 1.4E-8. This is less than 2 orders of magnitude below the BFN SCDF so it is judged to be an adequate level for screening of rugged components.
- SSCs that did not fall within inherently rugged or high-capacity were judged to require further evaluation.

The walkdown effort relied heavily on previous IPEEE and A-46 walkdown data. Where original A-46/IPEEE Screening and Evaluation Walkdown Sheets (SEWS) existed, they were credited and a walk-by was performed to evaluate interaction and potential changes to the area since the original walkdown. Where original A-46/IPEEE SEWS did not exist, a full walkdown was performed. The existing A- 46/IPEEE SEWS were reviewed and found to be well documented, although the original A- 46/IPEEE SEWS were performed several years prior to the SPRA evaluations, the BFN design change process implements Tennessee Valley Authority (TVA) seismic qualification standards that achieve equipment capacities at least as high as earthquake experience data. Any changes in the configuration of adjacent commodities to SEL items were specifically addressed during the SPRA walk-bys in the seismic interactions review.

The BFN walkdown team utilized training mock-ups when available to pre-walkdown components and gain insights into cabinet load paths, breaker functions and relay sensitivities. The peer review team identified this as a Best Practice for SPRAs and one that should be utilized when possible to ensure optimal efficiency and quality of inspections during plant walkdowns.

Plant-specific data was collected and applied in the fragility calculations. The Walkdown Team collected information to determine whether capacities from seismic experience data and Generic Equipment Ruggedness Spectra (GERS) and SQUG data were applicable to each SSC. The seismic fragility evaluations relied on the walkdown documentation to establish as-built conditions and identify likely seismic failure modes.

SSCs were evaluated for critical failure modes that could govern fragility of the components, including functional, load path, and anchorage failure modes. Potential failure modes were assessed during the walkdown and documented in the seismic walkdown report and corresponding SEWS. Operator pathways for access to SSCs outside the main control room were also walked down.

Potential seismic-induced flooding and fire sources were considered in the walkdowns. Credible sources were added to the SEL using a combination of input from the SPRA systems analysts and area based walkdowns to identify additional components that may be flood sources from past earthquake experience. Credible seismic-induced flood and fire interactions were identified during the walkdown for further evaluation. Other potential seismic interactions were also identified during the walkdowns and documented on the SEWS. Key issues included interactions between buildings, interactions between cabinets with chatter non-acceptable relays, and limited masonry wall interactions. The peer review team concurs that the walkdowns met the applicable SRs of the ASME/ANS RA-Sb Code Case.

Fragility parameters were calculated for the SSCs listed in the SEL and credited in the plant SPRA Model. Based on a sample review, fragilities for SSCs in the SEL were determined, whether by assigning high capacities or calculated fragilities, and discussions on the definition of failure modes took place throughout the project. Seismically induced soil-related failure modes and embankment failures that affect the SSC included on the SEL were developed. Although the final SEL does not include fragilities for soil-related and embankment failures, these are credited in the final SPRA model quantification.

A capacity-based HCLPF screening level of 2g PGA was established for “high-capacity” SSCs based on a single point estimate risk contribution of 1x10-7. However, no items were screened out of the PRA model and fragility estimates for “high-capacity” and “inherently rugged” SSCs were provided to the PRA analyst.

Fragility calculations for all SSCs not categorized as “high-capacity” and “inherently rugged” were developed in a two-phase approach. First, fragilities were initially developed for large groups of SSCs that were treated as correlated. A lead component was chosen from the group, and the “representative” fragility based on scaling of existing IPEEE calculations was assigned. As quantification iterations identified a need for refinement, the fragility groups were divided into more realistic subsets, and separate fragilities were developed for the potentially important SSCs. For components determined to contribute to CDF/LERF, fragilities were calculated based on the Conservative Deterministic Failure Margin (CDFM) hybrid approach and the Separation of Variables (SOV) method as described in industry-accepted EPRI guidelines. SOV calculations were specifically developed for SSCs with high contribution to CDF/LERF. A review of the final PRA Quantification Notebook showed that some SSCs with fragilities calculated via hybrid or conservative “representative” calculations were ranked as high contributors to CDF/LERF. However, the BFN SPRA provided documentation for sensitivity PRA (BFN-0-19-028, PRA Evaluation Response) runs showing the minimal impact of such conservatism in the overall PRA results.

Although a large number of SSCs remain in the SPRA model with conservative fragilities (more than 90% of fragilities), the BFN SPRA team justified this scenario based on the individual impact of the risk-significant components on the overall SPRA results (BFN- 0-19-028, PRA Evaluation Response). Overall, the peer review team finds the fragilities developed for the risk-significant SSCs to be acceptable for the BFN SPRA.

In summary, the fragility analysis generally meets the applicable requirements of the ASME/ANS RA-Sb Code Case #1.

A.2.3.3 Seismic Plant Response Analysis (SPR)

The SPRA model was developed by modifying the Full Power Internal Events (FPIE) PRA model to incorporate specific aspects of seismic analysis that are different from the FPIE. The logic model appropriately includes seismic-caused initiating events and other failures including seismic-induced SSC failures, non-seismic-induced unreliability and unavailability failure modes (based on the FPIE model), and human errors. Some refinements to the SPRA logic model were identified, including the need for a more comprehensive evaluation for crediting FLEX mitigation strategies.

The BFN SPRA used the EPRI External Hazards Human Reliability Analysis (HRA) approach to account for adjustments to the performance shaping factors to modify the internal events human error probabilities (HEPs), followed by a detailed SHRA of risk-significant human failure events (HFEs). The HRA was generally well-done, but additional refinements were identified to the HRA that could be performed to be more realistic (e.g., assess the impact of seismic effects on HFE time parameters).

The implementation of the EPRI 3002012980 [47] process for seismic-induced fire was comprehensive and complete. Every fire source was considered and clearly dispositioned. Qualitative screening was carefully applied and performed correctly and clearly. The evaluation of seismic-induced fire sources was documented in a clear, comprehensive, and traceable manner. The evaluation of seismic-induced fire impacts was identified as a Best Practice.

No SSCs in the BFN FPIE PRA were screened from the SPRA model. All fragilities associated with FPIE PRA SSCs were incorporated into the SPRA model. The comprehensive modeling of SPRA fragilities helped to ensure that potentially significant risk contributors were not inappropriately screened.

A number of sensitivities were performed to understand the impact of the various modeling and screening assumptions. In addition, a detailed evaluation was performed for the identification and disposition of sources of modeling uncertainty. In these aspects, the quantification of the BFN SPRA is judged to meet the PRA Standard.

The Level 2 SPRA LERF analysis addressed the uncertainty associated with potential extended evacuation times during a seismic event. However, some shortcomings were identified in the modifications incorporated into the Level 2 FPIE LERF analysis to support the development of the Level 2 SPRA LERF analysis.

In conclusion, the SPRA model integrates the seismic hazard, the seismic fragilities, and the systems-analysis aspects appropriately to quantify CDF and LERF, albeit with the above-noted deficiencies.

The seismic-PRA analysis was documented in a manner that facilitates applying and updating the SPRA model.

A.2.3.4 Peer Review Findings

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

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

However, the peer review team identified specific areas for improving the technical adequacy of the SPRA. These areas are documented as F&Os. At the conclusion of the peer review, there were 33 open Finding-Level F&Os as shown in Table A-1. Four SRs (all in the SPR element) were identified as Not Met.

Table A-1 Summary of Facts & Observations for the Browns Ferry Unit 1, 2 and 3 SPRA Peer Review

Element	F&Os			
	Findings	Suggestions	Best Practice	Total by Element
SHA(1)	4	2	1	7
SFR(1)	4	0	1	5
SPR(1)	25	3	1	29
TOTAL(2)	33	5	3	41

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

A.3 Revision of Model and Documentation

Following the peer review, the BFN SPRA model and documentation were updated to address each of the 33 Finding-Level F&Os. In addition, TVA generated closure documentation for each of the F&Os from the peer review against the PRA Standard (ASME/ANS RA-Sb-2013, Code Case 1).

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

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

The BFN Seismic Probabilistic Risk Assessment (SPRA) Finding-Level F&O Independent Assessment & Focused-Scope Peer Review was performed at the TVA Corporate offices in Chattanooga, Tennessee, October 1-4, 2019. The purpose was to perform an independent assessment in accordance with Appendix X of NEI 05-04/12-13 [55, 9] to review TVA's proposed close out of Finding-Level F&Os of record from prior PRA peer reviews against the PRA Standard (ASME/ANS RA-Sb-2013, Code Case 1).

The process used for the independent technical review is outlined in the Appendix X of NEI 12-13, which has been accepted by NRC. The review focused on the closure of the 33 open Finding-Level F&Os.

The review was based on results of a completed Boiling Water Reactor Owners' Group (BWROG) review of the BFN SPRA (final report issued June 2019). The result of this independent assessment is intended to be used to support future License Amendment Request (LAR) submittals. Finding-Level F&O dispositions reviewed and determined to have been adequately addressed through the independent assessment are considered "resolved" and no longer relevant to the current PRA model. Therefore, these F&Os do not need to be carried forward or discussed in future LAR submittals.

The Independent Assessment Team consisted of 6 team members with extensive qualifications and extensive experience in all areas of SPRA. All reviewers met the criteria specified in NEI 05-04 [55], NEI 12-13 [9], ASME/ANS RA-Sb-2013 PRA Standard Section 1-6.2 [8], and in NRC's memoranda outlining expectations for a finding closure independent assessment. Detailed resumes for each of the team members are provided in the closure review report.

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

Review team criteria (NEI 12-13 Section 2.2) and Review Schedule (NEI 12-13 Section 2-3) were addressed in recruiting and approving the closure review team members and defining the schedule for the review. Reviewer independence was established, approved, and documented in the closure review report. Reviewer experience meets the criteria specified in the NEI guidance documents and ASME/ANS RA-Sb-2013 PRA Standard Section 1-6.2. Overall review team experience is such that there were two qualified reviewers for each F&O.

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

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

A.4.2 Independent Technical Review Team Qualifications

Mr. Lawrence Lee is a Principal Engineer with 27 years of experience in the nuclear field specializing in Probabilistic Safety Assessment. Mr. Lee has experience in leading Level 1 and Level 2 PSA updates (internal and external events), shutdown safety assessment, On-line Maintenance, In-Service inspection of piping, MOV prioritization, air-operated valve (AOV) prioritization, and utility response to NRC compliance using PSA techniques. He has been involved in multiple projects ranging from FPIE updates, SPRA model development, and shutdown PRA model development. He has also supported various risk-related industry projects through the EPRI and the BWROG.

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

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

Mr. Apostolos (Paul) Karavoussianis is a Lead Engineer with 31 years of experience in project management and structural engineering. He manages and leads analysis and design projects of nuclear and heavy industrial facilities, semiconductor fabrication plants, petrochemical process plants, and pulp and paper mills. Mr. Karavoussianis managed and was the technical lead for the SPRA projects of Callaway Energy Center, Wolf Creek Generating Station and the new AP1000 units, Virgil C. Summer Nuclear Generating Station Units 2 & 3 and Vogtle Electric Generating Plant Unit 3 & 4. He has also consulted on the Seismic Fragility Assessment project at Palo Verde Nuclear Generating Station.

Mr. Stuart Lewis is currently a Principal Engineer with 40 years of experience in the nuclear industry. In this role, he draws on this experience to support risk assessments and risk-informed applications for nuclear power plants and to identify opportunities for further development of risk methods. Mr. Lewis provides consulting services in the performance of PRAs and in the application of risk results and insights for improving the safety and operational flexibility of nuclear power plants. Projects have related to risk-informed decision-making; various specific risk-informed applications; assessment of the risk benefits of advanced nuclear fuel concepts; advancements in methods for SPRA; peer reviews, including reviews for closure of existing findings; and developing and providing training of utility PRA staff.

Dr. Todd Radford is a Lead Engineer with 13 years of experience including 9 years at Jensen Hughes. In addition to acting as a Structural Engineering Manager in the Wakefield, MA, office of Jensen Hughes, Dr. Radford has been involved in numerous seismic evaluation projects for Jensen Hughes as both project manager and lead analyst. He is an expert in building modeling and SSI and has led response analysis efforts for multiple SPRA projects for nuclear power plants. Dr. Radford has been responsible for post-Fukushima engineering support including R2.3, ESEP, FLEX, SFP evaluations, and R2.1 HF confirmations. Dr. Radford has also led development efforts for Jensen Hughes' internal engineering analysis software including Spectra, SULTAN, and ANCHOR.

A.4.3 Independent Technical Review Team Conclusions

All the F&Os were assessed to be resolved during the closure review. All four SRs that that were previously assessment as "Not Met" were re-assessed as Capability Category II.

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

The BWROG performed a peer review of the SPRA in 2019. The SPRA was peer reviewed relative to Capability II for the full set of requirements in the PRA Standard (ASME/ANS RA-Sb-2013, Code Case 1). After completion of the subsequent independent assessment in 2019, which utilized the process given in Appendix X of NEI 12-13, the full set of SRs were met.

The final F&O dispositions are provided in the following pages in this Appendix. Table A-1 provides the dispositions for the original peer review findings within the scope of the F&O independent assessment.

Table A-1 is sorted by Review Unit in the first column. The columns in the table provide the following information (numbers denote column number):

1. Review Unit (SHA, SFR, or SPR).
2. The SR number against which the peer review Finding was referenced.
3. The original peer review team's assessment of Capability Category for the referenced SR.
4. The Finding Number from the peer review report.
5. The Finding Description from the peer review report.
6. A summary of the Basis and Suggested Resolution for the Finding from the peer review report.
7. TVA's description of the resolution of the Finding.
8. References to appropriate portions of the BFN SPRA Model and documentation to support TVA's resolution.
9. The Independent Assessment Team's assessment of whether TVA's resolution of the Finding represents PRA Maintenance or Upgrade.
10. The Independent Assessment Team's basis for Maintenance or Upgrade determination.
11. The Independent Assessment Team's assessment of adequacy of the Finding resolution.
12. The Independent Assessment Team's assessment of the new Capability Category of the referenced SR given the Finding resolution.

Table A-2 BFN SPRA F&O Closure Review Consensus Table

RU	SR	PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
SHA	C-SHA-I2	Met	20-4	SHA-I2 requires that secondary hazards that do not screen out be further analyzed to evaluate their frequency of occurrence and severity. The method used to estimate the median liquefaction-induced settlement and lateral spreading and associated uncertainty commingles epistemic and aleatory uncertainties and makes it difficult to judge whether the results are realistic. (This F&O originated from SR C-SHA-I2.)	As in the SSC, GMC, and SRA components of the SPRA, there are epistemic and aleatory uncertainties involved in estimating the magnitude of liquefaction-induced settlements and lateral spreading. The approach used in CJC-BFN-C-001 R0 as shown in Figures 5.1 and 5.2 commingles the two types of uncertainty, making it difficult to evaluate whether the results are realistic.	Aleatory and epistemic uncertainties are modeled in separate event and logic trees, respectively with an adaptation to the Latin Hypercube Sampling method used for variance reduction. Incorporation of logic and event trees into the analyses methodology results in changes to much of the text in the report in order to obtain consistently between the process, interim results, final results, and conclusions. Thus, the use of the logic and event trees is described in Section 5 of CJC-BFN-C-001 R1 while the resulting changes are incorporated as needed in the appropriate sections and Appendices. Specific changes associated with the updated methodology are reflected in the Record of Revision of CJC-BFN-C-001 R1.	CJC-BFN-C-001, Rev 1	Maintenance	The resolution was judged to be PRA Maintenance because it is a refinement of an existing calculation that partitions uncertainty into two types. The method used to combine the results from each type to calculate Am, Bu, and Br is documented in Appendix B of EPRI (2013). No new methods were used.	In Report CJC-BFN-C-001, Revision 1 (September 2019), sources of epistemic and aleatory uncertainty in the estimates of liquefaction-induced settlement and lateral spreading have been identified and modeled via logic and event trees, respectively (see Figures 5.1 through 5.4). Sources of epistemic uncertainty include: (i) liquefaction triggering methods for sand-like soils, (ii) shear wave velocity profiles, (iii) residual undrained strength models, and (iv) modulus reduction and damping curves. Sources of aleatory variability include: (i) earthquake magnitude and distance, (ii) various soil properties (see p. 40), (iii), SPT blow count, (iv) the depth to groundwater, (v) soil behavior (clay-like vs. sand-like), (vi) probability of liquefaction or cyclic softening, and (vii) soil amplification. The sources of epistemic and aleatory uncertainty are combined in a manner that yields estimates of Am, Bu, and Br that are more accurate and realistic than previous analyses (in which all the above items were modeled as aleatory uncertainty). This F&O is assessed as CLOSED.	Remains Met
SHA	C-SHA-J1	Met	20-5	In some cases, the specific steps included in the PSHA are based on technical approaches which require additional discussion or documentation to provide sufficient information to ensure that the PSHA results are valid. In these cases, the PSHA documentation should be improved to provide an improved technical basis and documentation. (This F&O originated from SR C-SHA-J1.)	1. Section 3 of the PSHA does not adequately describe the literature search and review of information and data to determine if the CEUS-SSC model can be used without modification. A list of references consulted to evaluate potential changes to the CEUS-SSC to be considered in the BFN PSHA and a brief summary of the disposition of new information contained in each reference should be provided. 2. Section 8.3 of the report does not adequately describe the steps taken to develop the smoothed UHRS (site-specific spectral shape factors that are scales to match the spectral accelerations at the seven frequencies. The procedure used to develop smooth spectra should be explained in more detail. 3. Section 8.3 of the PSHA describes the development of the smoothed horizontal GMRS and FIRS and the corresponding smoothed vertical GMRS and FIRS. However, the Structural Response Analysis described in SC Solution Report BFN-17-001, Rev. 1, uses the Uniform Hazard Response Spectra	1. Section 3.0 now refers to the detailed literature search and evaluation performed in Appendix G of 160029-PR-01 Rev. 3. 2. Section 5.1 of 160029-PR-01 Rev. 3 includes updated documentation on the steps taken to derive the HF and LF smoothed spectrum and the procedure for smoothing the response spectra is included in Section 8.3. 3. The UHRS used at 1E-5 is now included in Section 8.3. 4. Text was added to Section 6.4 of 160029-PR-01 Rev. 3 to acknowledge that there is a minor portion in principle that could be attributed to epistemic uncertainty and that in principle, we are not stating that epistemic uncertainty is zero, but rather that it is very minor and insignificant from a numerical standpoint.	1 through 6; 160029-PR-01, Rev 3 7; TVA-BFN-SPRA-001, Rev 2 8 through 11; CJC-BFN-C-001, Rev 1	Maintenance	The resolution was judged to be PRA Maintenance because it is a correction of documentation deficiencies that provides additional, important information that does not change the basis for the inputs and results.	Three reports were updated to address SHA F&O 20-5; (1) Fugro Consultants, Inc. (Fugro): Probabilistic Seismic Hazard Analysis (PSHA) for TVA Browns Ferry Nuclear Plant, PSHA Results Report PR No. 160029-PR-01, Revision 3 (August 2019), (2) TVA Report TVA-BFN-SPRA-001, Revision 2, Secondary Seismic Hazards for Browns Ferry Nuclear Plan, Seismic Probabilistic Risk Assessment (September 2019), and (3) Carl J. Costantino and Associates (CJC): Updated Soil Failure and Fragility Analysis for Browns Ferry Nuclear Plant (BFN), Report CJC-BFN-C-001, Revision 1 (September 2019). F&O 20-5 includes eleven (11) items. The resolution for each item is listed below. 1. Appendix G, of Fugro PSHA PR No. 160029-PR-01 Rev. 3 provides documentation of the references reviewed including a brief technical assessment for each reference of the possible relevance and need to modify the CEUS seismic source model. Appendix G is referenced in Section 3.0 of Fugro PSHA PR No. 160029-PR-01 Rev. 3. 2. Section 5.1 of Fugro PSHA PR No. 160029-PR-01 Rev. 3 provides an enhanced explanation of the steps taken to derive the HF and LF smoothed UHRS. 3. Section 8.3, Table 8-43, and Figure 8-79 of Fugro PSHA PR No. 160029-PR-01 Rev. 3 provides the smoothed UHRS for FIRS2 at a MAFE of 1x10-5 consistent with the spectra used for the structural analysis. Section 7.4 describes the derivation of V/H ratios at this MAFE, and Section 8.4 describes the development of strain-compatible soil properties at this MAFE. 4. Section 6.4 of Fugro PSHA PR No. 160029-PR-01 Rev. 3 provides an enhanced technical explanation and basis for why the site response analysis is adequately modeled using a single base-case shear wave velocity (VS) model for the rock layers at the BFN site. Fugro acknowledges that a portion of the uncertainty resulting from developing sixty (60) randomized profiles (modeling aleatory variability	

Table A-2 BFN SPRA F&O Closure Review Consensus Table

RU	SR	PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
					at a MAFE of 1x10-5 as input into the structural analysis. The PSHA report should provide sufficient information and explanation to develop the smoothed horizontal and vertical response spectra used for the structural analysis. 4. Section 6.4 of the PSHA indicates that a single base-case profile (i.e., no epistemic uncertainty) was found to be appropriate to model the rock layers at the site for the SRA. The decision to neglect epistemic uncertainty in the shear wave velocity profile below El. 515 ft-MSL must be better justified as a modeling choice rather than the absence of uncertainty in principle. 5.Section 6.4 of the PSHA indicates that for the compacted earth fill and crushed rock fill, industry average shear wave velocity values were used (listed in Tables 6-2 and 6-3). Insufficient information is provided to understand what specific steps were taken to derive the shear wave velocity values for the compacted earth fill and crushed rock fill. 6. Section 6.4.1 of the PSHA indicates that the EPRI (rock) dynamic properties are used to model the site response of the rock layers at the site. Tables 6-3 and 6-4 do not clearly indicate what low-strain damping is used for the SRA for deeper (> 500 ft within the rock) rock layers. The modeling of low-strain damping in the firm rock materials should be explained in more detail. 7.Given the prevalence of limestone in the vicinity of the site that is highly susceptible to the formation of karst features, the seismically induced collapse of karst features should be included as a potential secondary hazard at BFN. 8. Section 2.3 of CJC-BFN-C-001 R0 states that "based on the configuration of the cables within the duct bank and cable tunnels, it is judged that sufficient slack in the cables is available to accommodate the potential deformations of the	5. Section 6.2.1 and 6.4 (160029-PR-01 Rev. 3) documents the steps used to develop the shear wave velocity profiles in the fill. 6. Section 6.4.1 (160029-PR-01 Rev. 3) documents what low-strain damping is used and what kappa values were obtained and confirms its consistency. 7. Report TVA-BFN-SPRA-001, Revision 2 includes a discussion of the potential for seismically induced collapse of karst features based on the information presented in Section 3.3 of the PSHA report. 8, 9, 10, 11 CJC-BFN-C-001 is modified in Revision 1 to address Items 8, 9, 10, and 11. As suggested, Section 2.3 is modified to include the discussion provided in the response to GJR-03, Section 5.5 is deleted, the duplicate paragraph in Section 6.0 is deleted, and Table D1 is updated to reflect the shear wave velocities used in the analyses (consistent with the PSHA).				in VS, layer thickness, and total depth to hard rock) for the rock layers can be attributed to epistemic uncertainty. The impact on the derivation of hazard curves and associated FIRS and GMRS is judged to be insignificant. 5. Sections 6.2.1 and 6.4 of Fugro PSHA PR No. 160029-PR-01 Rev. 3 provide an enhanced explanation and technical basis of the steps taken to develop the VS profiles for the compacted earth fill and crushed rock fill. This includes explicit citation and references for the industry average VS values used, and improved comparison of the available site data with the base-case VS profiles used as input for site response modeling. 6. Section 6.4.1 of Fugro PSHA PR No. 160029-PR-01 Rev. 3 provides clarification of the low-strain damping used for the rock layers below a depth of 500 ft. This section also provides a check of the relative kappa contribution for the rock layers as further support to the low-strain damping for the deeper rock layers. 7. Section 3 of TVA Report TVA-BFN-SPRA-001 Rev. 2 provides discussion of the steps taken to screen out and eliminate the potential impacts resulting from possible karst features based on the discussion found in Section 3 of Fugro PSHA PR No. 160029-PR-01 Rev. 3. 8. Section 2.3 of CJC Report CJC-BFN-C-001, Rev, 1 now indicates that the relative fragility of the piping and the cables/buried conduit is the primary reason for screening out the cables/buried conduit. This section was also revised to reference the TVA drawing that provides insight regarding the configuration of the cables. 9. The test and section that were in Revision 0 of the CJC Report CJC-BFN-C-001 were deleted consistent with recommendation. 10. CJC Report CJC-BFN-C-001, Rev, 1 eliminates the duplicate paragraph that existed in the previous version of the report. 11. Table D1 of CJC Report CJC-BFN-C-001, Rev, 1 was revised to reflect the rock VS used in the site response analyses for secondary hazards, and the revised values are now consistent with those used in the PSHA as found in Fugro PSHA PR No. 160029-PR-01 Rev. 3. This F&O is assessed as CLOSED.	

Table A-2 BFN SPRA F&O Closure Review Consensus Table

RU	SR	PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
					cable tunnel and duct bank caused by potential movement from soil deformation." Similar statements are made for both the power cables for the RHRSW pumps beneath the turbine building Unit 3 and buried conduit exiting Units 1-2 and 3 DGB. As a result, this mode of failure is screened out, with further justification provided in the response to GJR-03. Although the relative fragility of the piping and the cables/buried conduit is noted (p. 18), the relative fragility is a more intuitive, compelling, and persuasive reason for screening out the cables than is the slack in the cables in the opinion of the PRT. 9. Section 5.5 of CJC-BFN-C-001 R0 is potentially misleading because the sensitivity study was performed using soil profiles from Sequoyah Nuclear Plant (SQN), and the application to BFN is questionable. More importantly, Section1 of the BFN PSHA (PR No. 160029-PR-01, Rev. 2) presents a summary of results from BFN that demonstrate that an equivalent linear site response analysis is indeed valid. 10. In CJC-BFN-C-001 R0, the second and third paragraphs in Section 6.0 (p. 51) are duplicates. 11. T able D1 of CJC-BFN-C-001 R0 does not reflect the rock shear wave velocities used in the site response analyses for secondary hazards.						
SHA	C-SHA-B3	Met	20-7	SHA-B3 requires that 'SHA analysts ensure that the' geologic and geotechnical data and information are sufficient to characterize local site effects, including their associated uncertainties.' Data and information related to the various site materials above the bedrock are discussed in the PSHA Report (Pr No. 160029-PR-01, Rev. 2) and the report related to assessing soil failure mechanisms and deformation (CJC-BFN-C-001, Rev. 0).? Review of these documents indicates the lack of an integrated effort to	The geologic and geotechnical information and data described in the PSHA Report (Pr No. 160029-Pr-01) and the report assessing soil failure mechanisms and deformations (CJC-BFN.C-001, Rev. 0) result in the following inconsistencies: (1) whether the geologic materials on both sides of the reactor building (turbine building side versus berm side) are the same or different, including their low-strain stiffness definition, (2) whether the geologic materials under the Yard area are in-situ soils or compacted fill, and (3) the extent and applicability of the SPT blow-count data and how that data is	Section 6.2.1 and 6.4 (160029-PR-01 Rev. 3) documents the clarifications as related to the PSHA report. The text of CJC-BFN-C-001 is modified in Revision 1 of the document to incorporate the responses to each of the Items listed in the F&O. These changes are described in Section 3 and Appendix D.2. It is noted that as a result of these changes the berm properties used in the analyses are affected and therefore liquefaction demands resulting in changes to the text in results and conclusions as	160029-PR-01, Rev 3 CJC-BFN-C-001, Rev 1	Maintenance	The resolution was judged to be PRA Maintenance because it is limited to a reinterpretation of existing geotechnical data. No new data were obtained.	Existing geotechnical data was reviewed and synthesized jointly by the two vendors responsible for performing the PSHA and evaluation of liquefaction-induced settlement and lateral spreading to address inconsistencies in previous, separate interpretations of the data. An assessment was completed and documented clarifying: (1) the nature of geologic materials on both sides of the reactor building (turbine building side versus berm side), including their low-strain stiffness definition, (2) whether the geologic materials under the Yard area are in-situ soils or compacted fill and the similarity of material properties, (3) the extent and applicability of the SPT blow-count data and how that data is used to aid in establishing low-strain stiffness properties, and (4) whether the material above El. 515 ft should be treated differently for site response analysis, soil-structure interaction analysis (see F&O 23-4), and soil deformations. The previous inconsistencies were resolved, and analyses were updated to reflect the revised geotechnical profiles. This F&O is assessed as CLOSED.	Remains Met

Table A-2 BFN SPRA F&O Closure Review Consensus Table

RU	SR	PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
				compile and integrate the available data on in-situ and fill soils at the site, leading to inconsistencies between analyses for site response, soil-structure interaction, and soil failure. (This F&O originated from SR C-SHA-B3.)	used to aid in establishing low-strain stiffness properties. While there is no impact on the GMRS/FIRS1/FIRS3 and the assessment of ground motion at the top of bedrock, the lack of an integrated assessment for the overlying site materials makes it difficult to determine if there are any impacts on the assessment of site response including the resulting strain-compatible properties, and the assessment of soil failure modes and the resulting soil deformations.	well as in supporting sections describing the analyses. Additionally, as indicated in the response to F&O 20-4, the approach used to include epistemic and aleatory uncertainty results in changes to the results. Specific changes to the report are indicated in the Record of Revision of CJC-BFN-C-001 Rev. 1 Refer to F&O 23-4 for the treatment of different soil properties in the SSI.					
SFR	C-SFR-D2	Met	19-13	Isolation valve FCV-069-0012 was not accessible during walkdowns due to high rad lockout. The follow- up resolution states that this valve is assumed similar to FCV-069-001 and 002. This is not a strong basis for similarity. Reliance on plant design documents is preferred. (This F&O originated from SR C-SFR-D2.)	This is an important isolation valve for CDF as it isolates the regenerative heat exchanger in case of breakout. Need to provide stronger argument for walkdown/fragility resolution.	Fragility Report TVAEBFN062-REPT-002 BFN Fragility Report Rev. 1 was updated to more technically address the fragility capability of FCV 69-0012. Data was collected from TVA engineering design basis documents and valve data information. Spatial interaction information around FCV69-0012 was observed based on photographs developed by TVA Radiological Control personnel. The TVA photographs validate that there are no adverse spatial interactions associated with FCV 69-0012. Based on review of TVA engineering data and available photographs it can be concluded that FCV 69-0012 meets all GIP requirements and GERS caveats. Consequently, a HCLPF capacity calculation was developed and inserted into Fragility Report TVAEBFN062-REPT-002 BFN Fragility Report Rev. 1. The result of the HCLPF capacity calculation demonstrated that the controlling isolation valve remained the HCLPF capacity established for the 69-001 valve.	TVAEBFN062-REPT-002, Rev 1	Maintenance	The closure team assessed and agreed with the TVA determination that the Browns Ferry SPRA response to this F&O is a PRA Maintenance activity and not an Upgrade. The activity includes documentation update and fragility calculation using previously used methods.	Photographs developed by TVA Radiological Control personnel were previously obtained and included in TVAEBFN062-REPT-001. These were used to screen the valve for spatial interactions and to judge that the valve was of very high seismic capacity (low eccentricity, light operator, robust yoke) and was conservatively assigned the standard TVA BFN 3g-2g capacity from design criteria BFN-50-C-7106. Based on this, TVAEBFN062-REPT-002 Rev. 1 develops a fragility for valve FCV-069-0012. As there is no longer an assumption of similarity, this F&O is assessed as CLOSED.	Remains Met
SFR	C-SFR-D7	Met	23-3	During the confirmatory walkdowns it was noted for cabinet 0-BDAA-211-0000B along with several other cabinets that fire extinguishers were hung from wall brackets near SEL equipment. The fire extinguishers are not addressed in either the	SFR-D7 specifies the identification of credible seismic interactions during the walkdown effort. While the Seismic Review Team (SRT) may ultimately conclude that the significance of these credible interaction hazards is negligible, the identification of these interaction hazards should be incorporated into	The BFN Seismic Review Team (SRT) agrees with the observation cited by the peer reviewer in the description of F&O 23-3. The CO2 distribution nozzle that is in contact with the top of cabinet skirt of cabinet 0-PNLA-009-0017 does not alter the seismic behavior of the	TVAEBFN062-REPT-001, Rev 1	Maintenance	The closure team assessed and agreed with the TVA determination that the Browns Ferry SPRA response to this F&O is a PRA	An extent of condition review was performed and a list of credible but non-consequential interactions was included in TVAEBFN062-REPT-001 with justification supporting the designation as non-consequential. As documentation has been updated to show that the subject interactions from the F&O were considered, this F&O is assessed as CLOSED.	Remains Met

Table A-2 BFN SPRA F&O Closure Review Consensus Table

RU	SR	PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
				original A-46 SEWS or the SPRA walk by SEWS. Although there are arguments which can be made as to why these fire extinguishers are acceptable as is, it should be documented that they were considered. During the confirmatory walkdowns it was noted that a CO2 distribution nozzle along with a light conduit were in direct contact with the top of the cabinet skirt for cabinet 0-PNLA-009-0017. Although there are arguments which can be made as to why this potential interaction is acceptable as is, it should be documented that they were considered. (This F&O originated from SR C-SFR-D7.)	the walkdown process and documentation with an evaluation of significance.	cabinet. Further, per Drawing BFN-0-47E610-39-1-CC "Mechanical Control Diagram CO2 Storage, Fire Protection & Purging System", the CO2 system is passive and open nozzle. The impact to the CO2 spray nozzle has no consequence on CO2 system. Therefore, although the interaction is credible, it is non-consequential for fragility development purpose. Also, during a seismic event the fire extinguishers that are hung from wall brackets or hooks can potentially fall from the hook and pose an interaction concern. The SRT judged that such interactions even though credible are non-consequential, as the fire extinguishers are unlikely to become missiles or cause harm to the SEL equipment. As an extent of condition review, the SRT revisited the field walkdown notes and photos to document the list of credible but non-consequential seismic interaction concerns. The list is provided in Section 5.28.6 "Non-Consequential Credible Interaction Concerns" of Walkdown Report, TVAEBFN062-REPT-001, Rev 1.			Maintenance activity and not an Upgrade. The activity includes an extent of condition review and documentation update with no new methods or calculations.		
SFR	C-SFR-B5	Met	23-4	A review of the soil properties at the site showed that the soil column south of the RB structure has different properties when compared with the soil on the north side of the RB structure. However, the RB SSI computer model uses the soil properties corresponding to the north side at both sides of the RB structure. This inconsistency in definition of soil properties can lead to differences in in-structure response and therefore changes in fragilities. A justification should be provided on the impact of this inconsistency on the fragilities and PRA results.	This inconsistency in definition of soil properties can lead to differences in in-structure response and therefore changes in fragilities.	Revision 1 of the soil failure and fragility report (document CJC-BFN-C-001 Rev. 1) includes an assessment of potential differences in the soil properties on the north and south sides of the Rector Building. The results provided in that report show that the properties of the soils surrounding the reactor buildings are similar in shear wave velocity (and thus in stiffness), and the differences are not large enough to significantly affect the computed SSI response of the buildings. The applicable text from document CJC-BFN-C-001 Rev. 1. Section D2.2, is excerpted below:	CJC-BFN-C-001, Rev 1 BFN-17-001, Rev 1	Maintenance	The closure team assessed and agreed with the TVA determination that the Browns Ferry SPRA response to this F&O is a PRA Maintenance activity and not an Upgrade. The activity includes documentation update for additional discussion with no new methods or calculations.	Revision 1 of the soil failure and fragility report (document CJC-BFN-C-001 Rev. 1) includes an assessment of potential differences in the soil properties on the north and south sides of the Rector Building (see F&O 20-7). The results provided in that report show that the properties of the soils surrounding the reactor buildings are similar in shear wave velocity (and thus in stiffness), and the differences are not large enough to significantly affect the computed SSI response of the buildings. As the impact of the difference between soil profiles on each side of the RB has been evaluated and the impact on fragilities determined to be non-significant, this F&O is assessed as CLOSED.	Remains Met

Table A-2 BFN SPRA F&O Closure Review Consensus Table

RU	SR	PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
				(This F&O originated from SR C-SFR-B5.)		"……deep foundations are used to support the Turbine Building to the north of the Reactor Building, [and] limited information (qualitative description) of soil overlaying the bedrock is available in this area. From descriptions in the boring log taken to investigate the underlying rock prior to construction (CD-Q0000-884999, "Browns Ferry Geology & Seismology Bore Hole Logs"), the soil materials are consistent with the overall site description of clayey alluvial and residual soils.					
SFR	C-SFR-F1	Met	23-5	The work encompassing the fragility analysis is provided in three separate notebooks - walkdown report, building report and fragility report. These reports provide the input, methodology and results of the overall fragility work in relation to SFR. In general, the notebooks are well organized. However, there were some specific instances where additional documentation and corrections to existing documentation needs to be made. (This F&O originated from SR C-SFR-F1.)	--Documentation and Reference recommendations-- 1 - TVAEBFN062-REPT-002, Section 13 documents risk quantification sensitivity studies to address sources of uncertainty in the SPRA model due to potential fragility improvements and assumptions made during component fragility development. This sensitivity study is documented in BFN-0-19-028. For completeness, this sensitivity study should be referenced in TVAEBFN062-REPT-002, Section 13. 2 - Add references for A-46 walkdowns SEWS to TVAEBFN062-REPT-001. The A-46/IPEEE SEWS are in TVA document W78 070428 003. Walkdown Package No. BFN1 CEB A46 IPEEE. This reference should be added to TVAEBFN062-REPT-001. -- Cut and paste and typo -- The following risk significant SSC's are not listed in Master Fragility table: SEIS_12-1P-1 - RHRSW Pumps based on pipe frag (Pipe Calc) SEIS_12-1P-2 - EECW Pumps based on pipe frag calc -- Errors in DG fragility group -- Relays BFN-C5-RLY-082-DRRA-1, BFN-3-RLY-082-DRRB-1, BFN-3-RLY-082-DRRC-1 & BFN-3-RLY-082-DRRD-1 are identified C5in the wrong Model Fragility Group. They should be in the Model Fragility	1. The reference to the sensitivity study BFN-0-19-028 is added in TVAEBFN062-REPT-002, Rev 1, Section 13 and Section 16. 2. Reference for the A-46/IPEEE SEWS that are documented in TVA document W78 070428 003. Walkdown Package No. BFN1 CEB A46 IPEEE is added to TVAEBFN062-REPT-001. This reference is added as Ref. 3.63 in Section 3, "References" of Walkdown Report. Additionally, the appropriate pointers to reference are added to the text in Section 1.5.1 "USI A-46/IPEEE Walkdowns". 3. The following fragility groups are added to the Master Fragility Table a. SEIS_12-1P-1 - RHRSW Pumps based on pipe frag (Pipe Calc) b. SEIS_12-1P-2 - EECW Pumps based on pipe frag calc 4. The relays identified in the F&O are not chatter sensitive. These relays are modeled as part of the host cabinet and are regrouped with the host cabinet Model Fragility Group. This update has been made in the Master Fragility Table. 5. The excerpt in the finding belongs to Section 11.2 (not Section 11.1) of Fragility Report.	1. TVAEBFN062-REPT-002, Rev 1. 2. TVAEBFN062-REPT-001, Rev 1. 3. Master Fragility Table; TVAEBFN062-REPT-002 Rev. 1, Appendix D. 4. Master Fragility Table; TVAEBFN062-REPT-002, Rev 1, Appendix D. 5. TVAEBFN062-REPT-002, Rev 1.	Maintenance	The closure team assessed and agreed with the TVA determination that the Browns Ferry SPRA response to this F&O is a PRA Maintenance activity and not an Upgrade. The activity includes documentation update only.	1 - TVAEBFN062-REPT-002 is updated appropriately 2 - TVAEBFN062-REPT-001 is updated appropriately 3 - SEIS_12-1P-1 is in the Master Fragility Table associated with component BFN-0-PIPE-023-XXXX and SEIS_12-1P-2 is in the Master Fragility Table associated with component BFN-0-PIPE-067-XXXX. TVAEBFN062-REPT-002 is updated appropriately. 4 - Relays BFN-3-RLY-082-DRRA-1 and BFN-3-RLY-082-DRRC-1 have been assigned fragility group SEIS_1C-4 and Relays BFN-3-RLY-082-DRRB-1 and BFN-3-RLY-082-DRRD-1 have been assigned fragility group SEIS_1C-5. These are consistent with the fragility groups of the host cabinets, therefore TVAEBFN062-REPT-002 is updated appropriately. 5 - TVAEBFN062-REPT-002 is updated appropriately As all documentation has been updated appropriately, this F&O is assessed as CLOSED.	Remains Met

Table A-2 BFN SPRA F&O Closure Review Consensus Table

RU	SR	PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
					Group of their Host cabinet Refer to Peer Review Report for Table of UNID, Component Type, Basis for SRT Review, Fragility Calculation Group, and Model Fragility Group. Cut and paste errors in Section 11.1 of TVAEBFN062-REPT-002 (entire labeled as {} need revision) From the Internal Flooding Analysis (Ref. 16.66), the conditional probability of failure of fire protection piping considering random pipe failure in U1 SE and U3 SW Corner Room is determined as 5.79E-04 and 4.23E-04, respectively. The flooding frequency is obtained by convolving the block wall failure fragility with the RE hazard. A representative fragility for block wall around the elevator shaft is developed based on {review of the 80-11 design evaluation} for masonry block wall and is documented in Appendix A, Attachment 22 to this report. The fragility parameters are summarized below: Am = {1.70}g Beta-R = 0.24 Beta-U = {0.32} HCLPF = {0.67}g	The finding has been resolved as suggested.					
SPR	C-SPR-B2	Met	19-1	Based on document BFN-0-19-029, BFN has 10 open finding F&Os on the internal event and 8 open findings F&Os on the internal flooding PRA. The above open F&Os were reviewed and it was determined that F&O 6-50 and F&O 2-31 may affect the results of the seismic-PRA.	The basis for why these F&Os do not affect the SPRA is not clear. FPIE F&O 6-50 F&O 6-50 identifies that some of the MOVs credited in the BFN ISLOCA fault tree are not tested to close against full DP. Therefore, there is significant uncertainty if the MOVs could actually close when exposed to full reactor pressure and be able to mitigate an ISLOCA event. The F&O identifies an example where gate U1_ISLVSS_2 credits isolation of MOVs FCV-74-52 and FCV-74-66, where FCV-74-66 is not designed to close against full DP. The BFN SPRA review and disposition for F&O 6-50 states that all the MOVs included in the ISLOCA fault tree are in the same fragility	Based on the discussion of the FPIE F&O 6-5 and SPRA F&O 19-1, the credit of 74-52 and 76-66 to mitigate the ISLOCA is questionable because these valves are not designed to close against the full differential pressure. Depressurization or not by the operators, it is likely these valves would not isolate the interface pipelines. Thus, 74-52 and 74-66 are set to failed in the SPRA modeling and there is no credit taken for operator actions to mitigate ISLOCA scenarios. The functional failure modes are the consequential effect of the seismic impact; therefore, the current modeling is acceptable to map the seismic	Methodology, Input and Model; PRA Evaluation BFN-0-19-062	Maintenance	The closure team assessed and agreed with the TVA determination that the BFN SPRA response to SPRA F&O 19-1 is a PRA Maintenance activity and not an Upgrade because the activity includes minor modeling enhancements (i.e., setting basic events to TRUE in a flag file) using no new methods or calculations.	Discussion of F&O 6-50 Based on the discussion of FPIE F&O 6-50 and SPRA F&O 19-1, the crediting of outboard isolation MOVs 74-52 and 74-66 to mitigate the ISLOCA is questionable because these valves are not designed to close against the full RPV differential pressure. Even with successful RPV depressurization, it is likely that these valves would not isolate the interface pipelines. Thus, the SPRA model has been revised to set MOVs 74-52 and 74-66 to failed (i.e., setting basic events for MOVs 74-52 and 74-66 to TRUE in flag file "BFN123_Flag_r8_SEIS.FLG") and there is no credit taken for operator actions to mitigate ISLOCA scenarios. The inboard isolation MOVs 74-53 and 74-67 are calculated to have Am=6.09g (MOV Group 08A-74-3). However, they are conservatively modeled with Am=2.94g (MOV Group 08A-74-4) using SPRA Fragility Group ID SEIS_14-1 that represents a large group of MOVs. The PRA model fragility group SEIS_14-1 contains various sub-groups of MOVs (Fragility Calculation Groups 08A-74-1, 08A-74-2, 08A-74-3, 08A-74-4 etc.). The lowest fragility calculated for any of these subgroups has an Am=2.94. This fragility was used to represent the failure of all MOVs in SEIS_14-1. MOV FCV-74-67 could have been based on MOV Group 08A-74-3 (Am=6.09g) instead of MOV Group 08A-74-4 (Am=2.94g). It is understood that using MOV Group 08A-74-4 with Am=2.94g as the	Remains Met

Table A-2 BFN SPRA F&O Closure Review Consensus Table

RU	SR	PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
					group. If one valve seismically fails, they all fail. Therefore, all the MOVs would fail to operate due to mapping from the same fragility group. However, this disposition is not clear for the BFN ISLOCA discussion. For example, in the ISLOCA evaluation for the RHR B injection path, MOV FCV-74-67 is normally closed (inboard valve) and MOV FCV-74-66 is normally open (outboard valve), where FCV-74-66 is not designed to close against full DP. Given that inboard valve FCV-74-67 is normally closed it is unclear what seismic failure mode is being applied in the BFN SPRA with respect to contributing to a seismic ISLOCA (e.g., boundary rupture failure mode). In addition, under gate !MOCFC1FCV_0740067 for VALVE FCV-74-67 FAILS TO CLOSE ON LAST DEMAND, the failure mode for FCV-74-67 contributing to an ISLOCA event is modeled as a failure to close event, which appears to assume that the normally closed FCV- 74-67 is normally open. Although this issue may be rooted in the modeling of the ISLOCA scenarios in the BFN FPIE PRA model, this issue appears to impact the SPRA model in how the MOV seismic fragilities are mapped into the SPRA model. If the MOV seismic failure is inappropriately mapped to model failure of an MOV to close to mitigate an ISLOCA, when the MOV is normally closed (e.g., inboard MOV FCV-74-67), then the assumption that the same MOV seismic fragility would apply to the normally open FCV-74-66 may not be valid. Typical MOV failure modes in an ISLOCA analysis include, but are not limited to: Failure to Hold on Demand, Internal Rupture, Mispositioned (very small probability). FPIE F&O 2-31 The BFN SPRA review and disposition for F&O 2-31 does not clarify how when swapping from the LPCI mode to the SPC mode, the LPCI injection valves need to cycle closed to prevent potential flow	group fragility to the group of valves. For FPIE F&O 2-31 and SPRA F&O 19-1, the valve status required for LPI and SPC was listed based on the internal events modeling. These valves and their associated basic events representing different failure modes were all mapped to the same fragility group in the Fragility to Component table.				fragility of seismic group SEIS_14-1 is conservative. However, the seismic group SEIS_14-1 did not show up as being risk significant and there were no ISLOCA sequences that appear in the cutsets. This assumption for grouping will be acknowledged in a revision to BFN-0-19-062. The review that shows no ISLOCA sequences appear in the cutsets will also be documented in the revision to BFN-0-19-062. Discussion of F&O 2-31 For FPIE F&O 2-31 and SPRA F&O 19-1, the valve status required for LPI and SPC was listed based on the internal events modeling. These valves and their associated basic events representing different failure modes (i.e., valve fails to open, valve fails to close) were all mapped to the same fragility group associated with a large group of MOVs in the Fragility to Component table using a conservative fragility value (i.e., Am=2.94g using SPRA Fragility Group ID SEIS_14-1). This F&O is assessed as CLOSED.	

Table A-2 BFN SPRA F&O Closure Review Consensus Table

RU	SR	PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
					diversion if the valves had previously successfully opened for LPCI operation. The same is true if needed to swap from SPC mode to LPCI mode during the mission time. Although likely true for the BFN SPRA, it is not clarified if the MOV seismic failure mode applies to both Failure to Open and Failure to Close failure modes. In addition, it is not clarified if the LPCI injection valve MOVs and the SPC return valve MOVs are in the same seismic group.						
SPR	C-SPR-C4	Met	19-3	The fire sources not credited in the SPRA model (assumed failed or inherently rugged) were not added to the SEL, although ultimately they were evaluated for fragility as if they were on the SEL. The SEL is therefore incomplete, although it is noted that there is no effect on the results since they were included in the quantitative analysis.	A significant portion of the credible seismic-fire ignition sources were not included in the SEL, thus the SEL is not complete. This is a systematic deficiency in the SEL.	Fire ignition sources retained in the BFN Fire Scenario Summary list were checked against the SEL. If an ignition source was already in the SEL, no further actions were taken. For ignition sources that were not in the existing SEL, these UNIDs were added individually. For ignition sources that did not have UNIDs, these sources were added to the SEL individually by scenario name. The note field was populated listing all additions as fire ignition sources.	Seismic PRA Equipment List Notebook	Maintenance	The review team concurs with the assessment that the changes to the SPRA constitute Maintenance. The change to the SPRA was to document in the seismic equipment list (SEL) the equipment that had already been considered to constitute potential ignition sources.	Review of Attachment C to the BFN Seismic PRA Seismic Equipment List (Calculation MDN0009992019000269, Rev. 1) confirmed that a large number of entries were added to the SEL to reflect equipment that could constitute a fire ignition source. This F&O is assessed as CLOSED.	Remains Met
SPR	C-SPR-C5	Not Met	19-4	The only credible failures induced by secondary hazards coming from SHA-I2 were liquefaction and lateral spreading. The credible induced events were identified and supplied for fragility analysis, and were treated as if they were in the SEL, but they were not actually incorporated into the SEL.	While there were only two failures induced by secondary hazards that were retained, neither of them was incorporated in the SEL, so a finding under this SR is required.	Two basic events were created in the SEL to represent the buried piping secondary hazards for the RHRSW and EECW piping.	Seismic PRA Equipment List Notebook	Maintenance	The review team concurs with the assessment that the changes to the SPRA constitute Maintenance. Additions to the SEL document the impacts of secondary failures already considered in the SPRA.	Review of Attachment C to the BFN Seismic PRA Seismic Equipment List (Calculation MDN0009992019000269, Rev. 1) confirmed that events BFN-0-PIPE-023-XXXX and BFN-0-PIPE-067-XXXX were added to the SEL to reflect failures of buried piping in the RHRSW and EECW systems, respectively. This F&O is assessed as CLOSED.	Met
SPR	C-SPR-C6	Met	19-5	The identification of the failure modes of interest was only performed for the dominant SSCs. Initial fragilities were provided before the SEL was developed, using lists of equipment such as IPEEE, NTTF 2.3, and ESEP. These were used for the initial runs, and the dominant contributors were discussed, including the	While the process followed did result in consideration of the PRA failure modes for the dominant SSCs, the fact that the information for all SSCs is missing from the SEL and thus is not available for the fragility analysis is a deficiency in the SEL.	The listed requirements were added as columns to the SEL. For the initial state, components like pumps and valves in the PRA were filled in based on the de-energized state shown in flow diagrams or system notebooks. The desire state and failed support system were filled in where applicable if information was readily available. The basic	Seismic PRA Equipment List Database	Maintenance	The review team concurs with the assessment that the changes to the SPRA constitute Maintenance. Additions to the SEL document information that provides a clearer understanding of	Review of the Seismic Equipment List (Access database Composite SEL_draft_20190910 – Copy.accdb) confirmed that information has been added for each SSC to indicate normal status, desired status, and PRA (basic event) failure mode(s). This F&O is assessed as CLOSED.	Remains Met

Table A-2 BFN SPRA F&O Closure Review Consensus Table

RU	SR	PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
				specific PRA failure mode. The PRA failure mode was not provided for the non-dominant contributors and not documented in the SEL. This SR is considered MET because the correct information was exchanged for the dominant contributors and the fragilities were based on that, and for the non-dominant contributors it is not as important. However, the SEL is missing this information and is therefore incomplete.		events failure modes were taken from the internal events analysis and input into the database.			the relevant status and failure modes of the SSCs considered in the SPRA.		
SPR	C-SPR-D5	Met	19-8	Detailed analysis was performed on all HEPs that had an F-V importance greater than 0.005, with one exception. HFA_0IR2_LPI does not have detailed HRA.	HFA_0IR2_LPI has LERF F-V 0.0084 for U1 and 0.009 for U2	This HEP, along with all the other seismic HEPs, was subject to a detailed analysis. As part of the resolution to F&O 19-9, the screening approach was abandoned and a detailed HRA was done for all HEPs. This is documented in the BFN Seismic HRA Notebook.	Seismic PRA Human Reliability Analysis Notebook	Maintenance	The review team concurs with the assessment that the changes to the SPRA constitute Maintenance. Detailed analyses were completed for one event that had previously been assessed using a screening approach. The analyses were performed using the same methods as for other HFEs previously evaluated.	Detailed assessments have been provided for event HFA_0IR2_LPI for use in the internal-events PRA and for seismic HRA bins 1 through 4. These assessments are documented in Appendix A of the Seismic PRA HRA Notebook (MDN0009992019000266, Rev. 1). For bins 1 and 2, the assessments are identical to that for the internal-events PRA, based on the relatively long system time window; the action becomes relevant about 49 min after the earthquake would have occurred, and the total time window is over 2 hr. The assessment for bin 3 accounts for a higher probability of failure due to reduced credit for review that could affect the cognitive contribution. This treatment is consistent with that for other HFEs as assessed for the seismic HRA. This F&O is assessed as CLOSED.	Remains Met
SPR	C-SPR-D3	Met	19-9	The general approach to the quantification is sound. The approach is clearly to adjust the PSFs to reflect the impact of the seismic events. However, there are some deficiencies noted. One such deficiency is as follows: - A table was provided that compared the screening HEPs that were used for the HFEs that were eventually analyzed using detailed analysis. In a number of cases, the screening values were lower than the detailed values.	The screening approach from EPRI 3002008093 was used. These screening values are expected to be conservative relative to the detailed values, otherwise the purpose of using the screening values is not achieved. For the affected HFEs, the screening values were underestimated by as little of 10% up to as much as a factor of 20. This illustrates a systematic problem with the application of the screening approach, and it is clearly possible that HFEs that retained their screening values could also be a lot higher, and thus could be significant contributors.	Since in some cases the screening method used produced HEPs that were lower than the HEPs used in the detailed analysis, screening HEPs were not used in the HRA analysis. Instead each HEP was subject to a detailed analysis where timing factors and stress levels were adjusted according to the seismic bin being analyzed for each HRA. This is discussed in the BFN Seismic HRA Notebook. Additionally, the boundaries of each EPRI seismic bin were redefined such that they were more closely tied to plant-specific factors, rather than the previous criteria which tied the bin boundaries to the percentage of components failed at each bin boundary.	Seismic PRA Human Reliability Analysis Notebook	Maintenance	The review team concurs with the assessment that the changes to the SPRA constitute Maintenance. Additional detailed analyses have been performed for certain HFEs using the same methods as previously employed in the SPRA.	Table 9-2 of the Seismic PRA HRA Notebook (MDN0009992019000266, Rev. 1) summarizes the post-initiator HFEs, including the relevant probabilities from the assessment for internal events and for each of the four seismic bins. All the events that were determined to be feasible were assessed using a detailed approach. Screening probabilities are no longer employed for these events. All HFEs determined to be infeasible or assessed for seismic HRA bin 4 are appropriately assigned failure probabilities of 1.0. This F&O is assessed as CLOSED.	Remains Met

Table A-2 BFN SPRA F&O Closure Review Consensus Table

RU	SR	PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
SPR	C-SPR-D3	Met	19-10	The general approach to the quantification is sound. The approach is clearly to adjust the PSFs to reflect the impact of the seismic events. However, there are some deficiencies noted. One such deficiency is: - Although the timing effects were assessed to some extent, they were not incorporated into the quantification because there was no time-reliability correlation used in the quantification.	The adjusted cognitive HFEs are only evaluated using CBDTM. EPRI TR-100259 states that CBDT was not created as a method of its own, but it was created as a supplement to HCR/ORE to take care of the cases where extrapolation of the HCR/ORE curves may not be appropriate. The concern with using CBDT alone is that it is completely time-insensitive, and so by itself does not account for the effects of time, as required by the SR. This is emphasized in EPRI 3002008093, which states 'Similar to internal events HRA, both the CBDTM and the HCR/ORE are to be considered for external events HRA. Both methods address detection, diagnosis, and decision making—the HCR/ORE implicitly and the CBDTM explicitly. The CBDTM was developed to provide a lower limit on the probability because the HCR/ORE calculates very low probabilities for HFEs for which the time available is long relative to the time required.'	Each seismic HRA was modified such that the CBDTM/HCR-ORE combination (max) method was used in order to more effectively account for HEPs that were sensitive to timing issues. This is documented in the seismic HRA Notebook and reflected in the HEPs calculated in the HRA database, which calculates the cognitive portion of each HEP using both the HCR/ORE method and the CBDTM method and uses the maximum of the two values for the final HEP.	Seismic Probabilistic Risk Assessment Human Reliability Analysis Notebook	Maintenance	The review team concurs with the assessment that the changes to the SPRA constitute Maintenance. Although the use of the HCR/ORE method was new to the SPRA for Browns Ferry, it was based on an earlier set of calculations performed for the internal-events PRA and documented in the BFN PRA – Human Reliability Analysis (NDN-000-999-2007-0032, Rev. 0). That PRA was subjected to a peer review in 2009.	As documented in Section 6 and Appendix A of the Seismic PRA HRA Notebook (MDN0009992019000266, Rev. 1) and in the calculations in the HRA Calculator file (BFN SPRA 8-21-19.hra), both the cause-based decision-tree method (CBDTM) and the HCR/ORE method were used in assessing post-initiator HFEs. The higher of the two results was employed for the cognitive contribution to the probability of each HFE. Timing impacts followed a convention with respect to assessing each HFE for different levels of seismic demand. In general, this entailed increasing the delay time (Tdelay) and, for actions taken outside the main control room, increasing the travel time contributing to the time to execute the action (Texe). A report was generated from the HRA Calculator file to permit comparing the times applied in the HCR/ORE calculations for events across the spectrum of the seismic bins. The conventions for adjusting timing were applied in a generally consistent manner. Exceptions were found for HFEs that accounted for immediate, memorized actions, and for some HFEs that were ultimately not used in the SPRA. This F&O is assessed as CLOSED.	Remains Met
SPR	C-SPR-D3	Met	19-11	The general approach to the quantification is sound. The approach is clearly to adjust the PSFs to reflect the impact of the seismic events. However, there are some deficiencies noted. One such deficiency is as follows. - While PSF adjustments were made in some cases, there were also many cases where no changes were made. This is counter to the experience in most other SHRAs that have been seen for recent PRAs.	Section 6.3.1 of the EPRI 3002008093 methodology provides guidance on adjusting the PSFs for CBDTM. Additional guidance is provided on table 6-15. While this guidance was followed to some extent, the adjustments appear to be minimal considering the guidance. The Pc assessments are based on minimal operator input, usually only at a high level as opposed to specifically addressing the context of the HRA bins. This appears to be in large measure because the plant specific context of each bin (i.e., what specifically has failed in each bin that would affect the HFE context) has not been clearly defined.	The EPRI bin definitions were redefined such that they more closely align with specific plant related phenomena during a seismic event. EPRI bin 1 is defined the same as in the peer review model where the upper bound is the safe-shutdown earthquake. In bin S1 there is no damage to the plant safety-related SSCs or non-safety SSCs required for operation. There is limited damage to non-safety, non-seismic designed SSCs like residences and office buildings. Bin S2 has been defined such that at the upper bound the turbine building has a 25% chance of failure and a loss of offsite power is likely (approximately an 89% chance of a LOOP occurring). In bin S2, there is no expected damage to the plant safety-related SSCs or to rugged industrial type non-safety SSCs required for operation. Damage may be expected to non-safety-related SSCs not important to plant	Seismic PRA Human Reliability Analysis Notebook	Maintenance	The review team concurs with the assessment that the changes to the SPRA constitute Maintenance. The HRA seismic bins were re-aligned to be more consistent with the seismic hazard for BFN, and some of the factors affecting quantification were adjusted, but no new methods or significant change in capability were introduced.	Table 7-1 of the Seismic PRA HRA Notebook (MDN0009992019000266, Rev. 1) summarizes the adjustments made to the assessments of post-initiator HFEs to account for the impacts of earthquakes of increasing magnitude. Included in the table is a description of the level of plant damage associated with each of the four HRA seismic bins. The impacts on timing and other PSFs are noted. These include • Increases in the delay time (1 min for bin 1, up to 5 min for bin 3), with no changes to Tcog or Texe, consistent with the recommendations of EPRI 3002008093, and increased travel time for actions taken outside the main control room; • Increase in assessed stress levels except where the stress was already assessed to be "high"; • Increases the potential for attention to be diverted from important cues, due, for example, to workload; • Expectations regarding availability of crew for review of actions and omissions in the main control room. These are judged to be reasonable steps in adapting the internal-events HRA to apply to seismic scenarios, and they are appropriate in the context of the EPRI guidance. This F&O is assessed as CLOSED.	Remains Met

Table A-2 BFN SPRA F&O Closure Review Consensus Table

RU	SR	PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
						operations and to the switchyard (e.g., LOOP expected). Some falling of suspended ceiling panels. EPRI bin S3 is defined such that there is significant damage to category 2 structures. Widespread damage to non-safety related SSCs and/or some damage expected to safety related SSCs. In bin S3 no credit for alarmed versus non-alarmed cues is taken due to the fact that there are multiple competing alarms and/or vibration trips and alarms occurring. The Turbine building failure probability at the upper bound of bin S3 is 96%. Also in bin S3, the Intake Pumping Station (the most fragile Category I structure) has a 25% chance of failing. In bin S4 (>1.5g) there is substantial damage to both safety-related and non-safety related SSCs.					
SPR	C-SPR-D4	Not Met	19-12	Section 7.1 of MDN0009992019000266 states that HFE timing was adjusted to account for seismic effects. Specifically: - Main Control Room Actions – Timing is adjusted depending on the seismic bin to account for additional time spent on crew briefs, competing actions, or other distractions in the control room. - Local Actions – Timing is adjusted depending on the seismic bin to account for additional time spent on crew briefs and increased travel time to the execution location due to seismic impacts on the operator pathway. However, a review was conducted on a timing output spreadsheet from the HRA calculator and it was determined that there were virtually no changes in time parameters from the FPIE parameters or between the bins.	The impact of seismic effects on HFE time parameters were not assessed in accordance with the statements made in MDN0009992019000266. Those statements are in accordance with the requirements of the SR. Guidance that would comply with the SR is provided in Section 6.3.2 of the EPRI 3002008093 methodology. Additional guidance is provided on table 6-15.	Timing factors were adjusted for each EPRI seismic bin. This generally consisted of adjusting the Tdelay of each HEP according to the given seismic bin in order to account for such factors as increased briefing times and other distractions caused by the seismic event. In addition, stress factors were adjusted to high for all seismic HEPs, and higher seismic bins assumed cues were not alarmed to account for the distraction of multiple alarms going off simultaneously. For operator actions external to the control room the Texe was increased by increasing the travel times to perform the operator action. The sigma values were also adjusted higher for the HCR-ORE method in the higher seismic bins. HRAs occurring in the highest seismic EPRI bin were assumed to fail.	Seismic PRA Human Reliability Analysis Notebook	Maintenance	The review team concurs with the assessment that the changes to the SPRA constitute Maintenance. The timing impacts were incorporated into the assessments of individual HFEs in a consistent manner.	As summarized in Table 7-1 of the Seismic PRA HRA Notebook (MDN0009992019000266, Rev. 1), certain time parameters were adjusted to account for impacts of seismic events. For example, the value from the internal-events PRA for Tdelay was increased by 1, 2 and 5 min for HRA seismic bins 1, 2 and 3, respectively. For ex-control room actions, travel time was increased as well. A new timing report was generated from the HRA Calculator file (BFN SPRA 8-21-19.hra). Comparisons of the timing information for various events confirmed that the intended timing impacts were appropriately accounted for. This F&O is assessed as CLOSED.	Met
SPR	C-SPR-B3	Met	25-1	Seismically induced failure modes and fragility values	Table E-1 of the BFN SPRA Methodology Inputs and Model	Based on the Master Fragility Table Excel file, all fragilities	PRA Evaluation BFN-0-19-062	Maintenance	The review team concurs with the	As summarized in the response to Finding 25-1 in the PRA Evaluation Response (BFN-0-19-029, Rev. 0), a review was made to determine the	Remains Met

Table A-2 BFN SPRA F&O Closure Review Consensus Table

RU	SR	PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
				have been linked to all the components modeled in the IEPRA. Section 5.1 'Assumptions' of the Seismic Methods Notebook identifies that seismic SSC failures are treated as a unique kind of failure mode and are assumed to be complete failures, in that the SSC fails to perform its function. Seismic failure modes are identified by the fragility team for anchorage, functional, and block wall failures. Additional failure modes for other types of SSCs (e.g., structures) are evaluated (e.g., shear and bending) as needed.	Notebook identifies that SPRA model fragility group SEIS_11-3 "HPCI RCIC pumps (Fragility Group 05-01)" is modeled with Am=4.48g. The Master Fragility Table (MFT) (i.e., Column O) identifies that the majority of SSCs included in fragility group SEIS_11-3 are calculated to have Am=4.48g. However, fragility group SEIS_11-3 also includes two (2) SSCs (i.e., CHILLED WATER CIRCULATING CHW PUMP A and B) with a lower Am=2.52g Approximately 10 other examples were identified similar to the above comment. This combines fragility groups that are not actually seismically correlated, which results in collapsing of cutsets. As a result, it can be said that certain specific failures that should be specifically in the model are not. This is an extent of condition issue. The above is only an example.	were reviewed individually and updated or re-mapped in some cases where necessary. All changes are documented in PRA Evaluation BFN-0-19-062. The 'Fragility' table was updated in the FRANX software to reflect updated and newly added fragility groups. Since some components were re-mapped to different fragility groups, the 'Fragility to Comp' table also required updating in the FRANX software. These changes were populated to the CAFTA model database.			assessment that the changes to the SPRA constitute Maintenance. The changes to the mapping to fragility groups more appropriately capture some SSC capacities, but no new methods were employed and no significant changes to the insights from the PRA resulted.	extent of condition for this inconsistency in mapping to fragility groups. As a result, a significant number of SSCs were reassigned to new or different fragility groups. These reassignments are tabulated in the description of the response. A spot check was performed for selected entries in this table relative to the FRANX fragility mapping (BFN_Seismic_Rev1_U1CDF.franx) and the tabulation of fragilities in Appendix D of the Fragility Evaluation Report (TVAEBFN062-REPT-002, Rev. 1). No instances of apparently inappropriate assignment to fragility groups were identified. This F&O is assessed as CLOSED.	
SPR	C-SPR-B6	Met	25-2	The component chatter analysis developed in a detailed and systematic process for the Browns Ferry Units 1-3 Seismic PRA. The identification and initial screening process for the component chatter analysis is documented in the Seismic PRA Chatter Analysis Report (TVAEBFN062-REPT-003). The Chatter analysis report determined which components could not be screened from the SEL for chatter concerns. Fragilities were then developed for the unscreened relays in the Fragility Analysis report (Section 10 and Appendix E) (TVAEBFN062-REPT-002).	Contact chatter fragility calculation SEIS_11-1R1 'Relay group 1 for group SEIS_11-1 (EECW Pp B3&D3 UV device)' includes relays associated with host cabinets on U2 RB Elevation 593 and 621. SPRA Fragility Group SEIS_11-1R1 is shown in the results as risk significant (e.g., U1 Seismic CDF FV = 2.3E-2). Relays in host cabinets that are on different elevations should not he same calculation group.	Table 3 of the Enercon refinement document, attachment 32 of Report TVAEBFN062-REPT-002 was reviewed to determine the relays of interest in the refinement calculation. Based on the review, RHRSW pumps B3 and D3 should be divided further. Following the recommendation, fragility group SEIS_11-1R1 is sub-divided into groups SEIS_11-1R1-1 and SEIS_11-1R1-2. Using the FRANX software, in the 'Fragility' table, group SEIS_11-1R1 was removed and SEIS_11-1R1-1 and SEIS_11-1R1-2 was created. The 'FireInitiatorHRA' table was also updated with basic events tied to these relay groups and populated within the CAFTA database.	PRA Evaluation BFN-0-19-062	Maintenance	The closure team assessed and agreed with the TVA determination that the BFN SPRA response to this F&O is a PRA Maintenance activity and not an Upgrade because the activity includes minor modeling enhancements (i.e., revisions to FRANX input file) using no new methods or calculations.	Based on the recommendation in F&O 25-2, fragility group SEIS_11-1R1 is sub-divided into groups SEIS_11-1R1-1 (EECW Pp B3 UV device) and SEIS_11-1R1-2 (EECW Pp D3 UV device. Using the FRANX software, in the 'Fragility' table, group SEIS_11-1R1 was removed and groups SEIS_11-1R1-1 and SEIS_11-1R1-2 were created. In addition, the 'FireInitiatorHRA' table was also updated with basic events tied to these relay groups and populated within the CAFTA database. The cutsets were reviewed to verify that new fragility groups SEIS_11-1R1-1 and SEIS_11-1R1-2 appeared appropriately. The BFN SPRA team discussed how the FRANX 'fragility' table and the 'FireInitiatorHRA' table were used in conjunction to apply seismic induced fragility events to multiple PRA basic event failure modes (e.g., pump fails to start, pump fails to run). This F&O is assessed as CLOSED.	Remains Met
SPR	C-SPR-E5	Met	25-3	Per Section 9.1 of the BFN SPRA Quantification, Sensitivity and Uncertainty Notebook, 'The uncertainty analysis was performed with	The Monte Carlo evaluation was performed with 20,000 samples, using 1000 cutsets out of approximately 7000 total Seismic CDF cutsets (and processed without	PRA evaluation BFN-0-19-065 completed using the one-top model shows that the uncertainty bands of CDF and LERF are not significantly	PRA Evaluation BFN-0-19-065	Maintenance	The closure team assessed and agreed with the TVA determination	Additional parametric uncertainty calculations were performed by TVA for processing various levels of cutsets (i.e., from 0 up to 500 cutsets) through UNCERT and ACUBE. Figures 1 and 2 in BFN-0-19-065, "BFN SPRA Uncertainty Analysis" indicate that the propagated mean and	Remains Met

Table A-2 BFN SPRA F&O Closure Review Consensus Table

RU	SR	PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
				UNCERT 4.0, using Monte Carlo sampling with 20,000 samples and ACUBE processing of 1,000 cutsets.' TVA identified that, in reality, ACUBE was not used (i.e., 0 cutsets processed with ACUBE) in the UNCERT runs because it caused the computer to crash. (This F&O originated from SR C-SPR-E5.)	ACUBE). Typical industry parametric uncertainty analyses have used a larger number of cutsets. Using more than 1000 of cutsets may provide different risk insights for the parametric uncertainty results. Using only 1000 cutsets with UNCERT (and 0 cutsets processed with ACUBE) may explain why the BFN Unit 1, 2, and 3 propagated uncertainty mean CDFs are approximately a factor of two (2) higher than the respective point estimate CDFs. Processing more than 1000 cutsets with ACUBE may result in calculating significantly lower propagated uncertainty mean CDF values. Similarly, processing more than 1000 cutsets with ACUBE may also result in calculating significantly lower propagated uncertainty mean LERF values.	affected by the number of cutsets processed by ACUBE.			that the BFN SPRA response to this F&O is a PRA Maintenance activity and not an Upgrade because the activity includes running additional parametric uncertainty calculations using UNCERT and ACUBE using no new methods or types of calculations.	uncertainty bands are relatively insensitive to the number of ACUBE cutsets processed. The Unit 1 UNCERT sensitivity case results show that increasing the number of cutsets processed through ACUBE from 200 to 500 decreases the UNCERT mean value for CDF and LERF by approximately 3% and 6%, respectively. Therefore, further increasing the number of cutsets processed through ACUBE for the UNCERT evaluation likely would not result in a significant decrease in the UNCERT mean value. This F&O is assessed as CLOSED.	
SPR	C-SPR-E6	Met	25-4	The seismic impacts (e.g., fragilities, operator action, containment bypass scenarios) were appropriately incorporated into the Level 2 PRA structure taken from the FPIE Level 2 PRA model. The SPRA model assumed that any Large Late Releases from the FPIE PRA modeling insights would be treated as Large Early Releases for seismic ground motion >0.5g. However, the Level 2 CET structures used as input to the Level 2 SPRA model identified selected sequences as “Containment Intact” with no apparent basis. The “Containment Intact” sequences were excluded from the SPRA Level 2 model. Based on this, the base SPRA model may under-predict the LERF because some of the scenarios included in the “Containment Intact” sequences do not have an intact containment (e.g., Level 1 ATWS scenarios with containment modeled as already failed in the Level 2).	To support the BFN ILRT LAR submittal, selected sequences in the BFN FPIE Level 2 CETs were re-defined from the “No LERF” end state to the “Containment Intact” end state for the BFN SPRA. The sequences in question include the following issues: • For CET 1 (for Classes 1, 3A, 3B, and 3C), the “Containment Intact” sequence does not appear to consider the status of containment heat removal in the CET. Without considering containment heat removal, an intact containment cannot be guaranteed. • For CET 2 (for Class 3D), the containment is already assumed failed in the Level 1 because of the LOCA with loss of vapor suppression. By default, the containment is modeled as failed in the Level 2. • For CET 2 (for Class 4), the containment has already failed in the Level 1 because of the unmitigated ATWS event (e.g., high suppression pool temperature and hydrodynamic loads). By default, the containment is assumed failed in the Level 2. In addition, the Level 1 Class II (i.e., Loss of Containment Heat Removal) core damage sequences are not transferred and evaluated in the	The flag basic event FLG_LATE_SEIS_EARLY is used to reverse the base case and the sensitivity case 2 of BFN SPRA Rev0 model. In this Rev1 model, in the flag file “BFN123_Flag_r8_SEIS.FLG” of the base case, the flag FLG_LATE_SEIS_EARLY is set to FALSE such that the large late release (gate U1S_LLR_TOP1 for earthquake >0.5g, %G4 and above) will not be quantified as large early release. For sensitivity case, the flag FLG_LATE_SEIS_EARLY is set to 1.0 by commenting the setting out to quantify the large late release as the large early release. To model the containment intact sequences that potentially could be LERF cases, the fault tree is updated to include all the containment intact groups: Sequences 1and 10 from CET1, Sequence 1 from CET2 Class 3D and Sequence 1 from CET2 Class 4 according to BFN-0-16-040 for Units 1, 2 and 3 integrated leakage rate test risk evaluation. The flag basic event FLG_CONT_INT is set to 1.0 to quantify all the containment intact sequences as LERF cases.	PRA Evaluation BFN-0-19-062.	Maintenance	The closure team assessed and agreed with the TVA determination that the BFN SPRA response to this F&O is a PRA Maintenance activity and not an Upgrade because the activity includes minor modeling enhancements (i.e., setting basic events to FALSE in a flag file and inserting new Level 2 basic events and gates) using no new methods or calculations.	Based on the recommendation in F&O 25-4, the sensitivity case for not converting Large-Late sequences to Large-Early sequences (i.e., Sensitivity Case 2 in Table 10-1 of the BFN SPRA Quantification Notebook) is incorporated into the updated Base Case SPRA model. Typical industry SPRA models have assumed no significant changes to the definition of "early" for the evaluation of LERF. New Sensitivity Case 6 in Table 10-1 of the BFN SPRA Quantification Notebook shows that assuming that Large-Late sequences lead to the Large-Early end state would increase the LERF from 3.29E-6/yr to 4.16E-6/yr (i.e., +27%). For the new sensitivity case, the Level 2 model also incorporates the conservative assumption that the “Containment Intact” sequences lead to the Large-Early end state. This F&O is assessed as CLOSED.	Remains Met

Table A-2 BFN SPRA F&O Closure Review Consensus Table

RU	SR	PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
					BFN Level 2 SPRA. The BFN Full Power Internal Events (FPIE) Level 2 PRA, Level 1 Class II (i.e., Loss of Containment Heat Removal) core damage sequences do not contribute to LERF sequences because the BFN Emergency Action Levels (EALs) support that a General Emergency will be declared sufficiently early such that any offsite releases from Class II events will be non-LERF. Therefore, a Level 2 Class II Containment Event Tree (CET) was not developed for the BFN FPIE Level 2 PRA. A Level 2 Class II CET would have a similar accident sequence structure as the Class IIID and Class IV CETs.	This assumption is very conservative, but there is no basis to determine the split fraction of the LERF sequences out of the intact sequences, in the sensitive case, FLG_CONT_INT can be set to different values like 0.25, 0.5, 0.75 in order to account for the uncertainty of the split fraction. New Basic events and gates are added into the fault tree. It should be noted that the unit 1 case is discussed specifically, but similar logic was also built for Units 2 and 3. The gates U1_CONT_INTACT_G10, U1_CONT_INTACT_G11 and U1_CONT_INTACT_G20 are added to the larger late release to the large early release gate U1S_LLR_TOP for CET1, CDT2_3D and CET2_4 respectively.					
SPR	C-SPR-E3	Met	25-5	QU-D6: Met as significant contributors were identified. However, it appears that two significant contributors were missed.	Fragility group SEIS_12-1a “EECW Pumps (Fragility Group 06-03-01)” with Am of 1.34g has a U1 Seismic CDF FV=0.0 while fragility group SEIS_12-1P-2 “EECW Pumps based on pipe frag calc” is modeled with a higher Am=1.86g and is a top risk contributor (e.g., U1 Seismic CDF FV = 6.0E-2 in Table 8-4 of the BFN SPRA Quantification Notebook (MDN0009992019000268)). Both SEIS_12-1a and SEIS_12-1P-2 should have the same consequences in the SPRA model (i.e., loss of all EECS). Therefore, SEIS_12-1a should have a higher FV. Similarly, fragility group SEIS_12-1b “RHR[SW] Pumps (Fragility Group 06-03)” with Am of 1.45g has a U1 Seismic CDF FV=0.0 while fragility group SEIS_12-1P-1 “RHRSW Pumps based on pipe frag (Pipe calc)” is modeled with a higher Am=1.82g and is a top risk contributor (e.g., U1 Seismic CDF FV = 6.6E-2 in Table 8-4 of the BFN SPRA Quantification Notebook (MDN0009992019000268)). Both SEIS_12-1b and SEIS_12-1P-1 should have the same consequences in the SPRA model (i.e., loss of all RHRSW). Therefore, SEIS_12-1b should have a higher FV.	The mapping in the "FireInitiatorHRA" table was updated with the 'dummy' basic events. The 'dummy' basic event gets created from the "FireInitiatorHRA" table and is injected into the fault tree along with the internal event basic event. A test run was conducted to show the Seismic CDF AND Seismic LERF. Both SEIS_12a and SEIS_12b appeared in the cutsets as well as SEIS_1P-1 and SEIS_1P-2. The purpose of the test run was to show these fragility groups indeed exist in the cutsets and the SYSIMP risk importance results. The results showed that these groups are represented correctly. See PRA Evaluation BFN-0-19-062.	PRA Evaluation BFN-0-19-062	Maintenance	The closure team assessed and agreed with the TVA determination that the BFN SPRA response to this F&O is a PRA Maintenance activity and not an Upgrade because the activity includes minor modeling enhancements (i.e., revisions to FRANX input file) using no new methods or calculations.	The mapping in the "FireInitiatorHRA" table was updated with the 'dummy' basic events (i.e., basic events with the “_dum” on the end of the basic event name). The 'dummy' basic event gets created from the "FireInitiatorHRA" table and is injected into the fault tree along with the internal event basic event. A test run was conducted to show the Seismic CDF AND Seismic LERF. Both SEIS_12a and SEIS_12b appeared in the cutsets as well as SEIS_1P-1 and SEIS_1P-2. The purpose of the test run was to show these fragility groups indeed exist in the cutsets and the SYSIMP risk importance results. The results showed that these groups are represented correctly. This F&O is assessed as CLOSED.	Remains Met

Table A-2 BFN SPRA F&O Closure Review Consensus Table

RU	SR	PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
					In the "Groups to Components" window in FRANX file "BFN_Seismic_Rev0_U1CDF.franx", it appears that fragility groups SEIS_12-1P-1 and SEIS_12-1P-2 (i.e., for the pipe fragilities) are not mapped to any SPRA basic events. However, fragility groups SEIS_12-1a and SEIS_12-1b (i.e., for the pump fragilities) are mapped to SPRA basic events (e.g., RHRSW pump fails to start or run). Based on the above, it would be expected that fragility groups SEIS_12-1P-1 and SEIS_12-1P-2 (i.e., for the pipe fragilities) do not appear in the cutsets, while fragility groups SEIS_12-1a and SEIS_12-1b (i.e., for the pump fragilities) do appear in the cutsets. However, the opposite appears to be true. For example, in file "U1_CDF-consolidated cutset file.cut", fragility groups SEIS_12-1P-1 and SEIS_12-1P-2 (i.e., for the pipe fragilities) do appear in the cutsets, while fragility groups SEIS_12-1a and SEIS_12- 1b (i.e., for the pump fragilities) do not appear in the cutsets.						
SPR	C-SPR-F1	Met	25-6	The seismic plant-response analysis and quantification was documented in 'SPR' Notebooks (Seismic Methods, SQU) in a manner that facilitates PRA applications, upgrades, and peer review.	Discuss the process for accounting for all of the unscreened relays identified in the Seismic PRA Chatter Analysis Report (TVAEBFN062-REPT-003) compared to the contact chatter groups ultimately included in the SPRA model. The BFN SPRA team identified the following: The Seismic PRA Chatter Analysis Report (TVAEBFN062-REPT-003) Appendix B "Components of Chatter Concerns Requiring Functional Fragility Analysis" has 615 line items but does not actually have 615 unscreened relays. The chatter analysis was performed based on component mis-operation and appendix B of the report listed the "affected" component (columns 2 and 3 of the table) and then the relay (column 4 and 5) that causes	An update to section 2.0 of the seismic PRA Chatter Analysis Report (TVABFN062-REPT-003). The changes include the clarification for the response to F&O 25-6, correction for the Errata and additional editorial changes.	TVABFN062-REPT-003, Seismic PRA Chatter Analysis Report, Section 2.0	Maintenance	The closure team assessed and agreed with the TVA determination that the BFN SPRA response to this F&O is a PRA Maintenance activity and not an Upgrade because the activity involves documentation changes only using no new methods or calculations.	Section 2.0 of the seismic PRA Chatter Analysis Report (TVABFN062-REPT-003) was updated to include the clarification for the response to F&O 25-6. This was a documentation update to identify the process for accounting for all the unscreened relays compared to the contact chatter groups ultimately included in the SPRA model. This F&O is assessed as CLOSED.	Remains Met

Table A-2 BFN SPRA F&O Closure Review Consensus Table

RU	SR	PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
					the mis-operation. This results in duplicate relay entries due the same components on each of the units and multiple functions of individual relays. For example, the first line of table identifies the component as BFN-1-FCV-071-0002 and the controlling relay as 13A-K33. This relay appears 6 times in the chatter report. This relay affects an additional valve (BFN-1-FCV-071-0003) and occurs on all three units. This was recognized, and the duplicate items were eliminated during the fragility development. Also, during this review several relays chatter events were noted to be "acceptable" due to operating conditions such as diesel generator lockouts when the diesel is in a test configuration. These were left in the table but annotated as "Chatter Acceptable".						
SPR	C-SPR-B4	Met	26-2	Fragility group definitions were provided by the fragility team. The correlations between different components were established based on the failure mode, similarity, orientation, location, fragility, and other seismic concerns. Section 6.7 of TVAEBFN062-REPT-002 (Browns Ferry Nuclear Power Plant Components and Structures Fragility Evaluation) summarizes technical basis for establishing correlations and grouping. The Fragility group definitions provided by the fragility team are summarized in Column "Fragility Calculation Group" of Appendix D "Master Fragility File" of the TVAEBFN062-REPT-002 BFN Fragility Report. The fragility groups actually included in the SPRA model are shown in Column "Model Fragility Group" of Appendix D.	SPRA model fragility group SEIS_12-2 "(RHR CS pump (Fragility Group 06-01 AND 06-02)" is inappropriately mapped to fail U2 and U3 RCIC pumps in addition to U1, U2, and U3 RHR and Core Spray pumps. The U2 and U3 RCIC SSCs mapped to Model fragility group SEIS_12-2 should be changed to Model fragility group SEIS_11-3. Model Fragility group SEIS_11-3 contains the corresponding U1 RCIC pump.	The Unit 2 and Unit 3 pumps are mapped differently from the Unit 1 RCIC pumps in the BFN SPRA Rev 0 model. This mapping was corrected by mapping the unit 2 and unit 3 RCIC pumps to SEIS_11-3 within the FRANX software in the 'Fragility_To_Comp' table. The new mapping was then populated to the CAFTA database. A test run was conducted to show Seismic CDF and Seismic LERF for one of the updated units. See PRA Evaluation BFN-0-19-062.	PRA Evaluation BFN-0-19-062	Maintenance	The review team concurs with the assessment that the changes to the SPRA constitute Maintenance. The changes to the assignment to fragility groups has a small impact on the PRA results, generally leading to a very small decrease in CDF and LERF. No new methods were introduced, and no significant new insights resulted from this change.	The response to Finding 26-2 in the PRA Evaluation Response (BFN-0-19-029, Rev. 0), provides a thorough discussion of the steps taken to change the mapping of the RCIC pumps to the proper fragility group. Review of the FRANX fragility mapping (BFN_Seismic_Rev1_U1CDF.franx) confirmed that the re-assignments have been properly implemented. This F&O is assessed as CLOSED.	Remains Met
SPR	C-SPR-B8	Not Met	26-3	The only new logic related to system modeling added to the BFN SPRA was regarding crediting the FLEX nitrogen bottles for hardened wetwell	1. Clarify if the FLEX Nitrogen bottles (basic event TKURP0TNK_N2FLEX) and associated operator action HFA_OPS_FLEXN2ALIGN (Seismic -	In the 'Components' table, a pseudo UNID was created to represent the 4 nitrogen carts functioning as an alternate supply to the SRVs with the	PRA Evaluation BFN-0-29-062	Maintenance	The closure team assessed and agreed with the TVA determination	In the 'Components' table, a UNID was created to represent the four (4) nitrogen carts functioning as an alternate supply to the SRVs with the assumption of complete correlation. A new UNID name was created for the HCVS N2 bottles and a new component failure basic event TKURP0TNK_N2HCVS and a newly created operator action	Met

Table A-2 BFN SPRA F&O Closure Review Consensus Table

RU	SR	PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
				venting. An operator action to connect and operate the equipment was added to the model. No other new equipment was added to the model. This is discussed in Section 6.5.2 of the Methodology, Inputs, and Model Notebook. The FLEX Nitrogen bottles (basic event TKURP0TNK_N2FLEX) and associated operator action HFA_OPS_FLEXN2ALIGN (Seismic - Operator fails to align FLEX N2 backup to DCA) discussed in Section 6.5.2 support backup pneumatic supply for both the Hardened Wetwell Vent valves and the SRVs for Reactor Depressurization. The fault tree logic (e.g., gate U1_VENT_N2FLEX) models that the same FLEX Nitrogen bottles and operator action supports both the Hardened Wetwell Vent valves and the SRVs for Reactor Depressurization. However, based on information provided by BFN, the Nitrogen bottles providing backup supply to the Hardened Wetwell Vent valves are different than the Nitrogen bottles providing backup supply to the SRVs for Reactor Depressurization. In addition, the operator actions should be different. Crediting the FLEX Nitrogen bottles was the only change incorporated for the SPRA logic and the fault tree model was not updated appropriately (i.e., assumed the same fragility and the same operator action).	Operator fails to align FLEX N2 backup to DCA) discussed in Section 6.5.2 support backup pneumatic supply for both the Hardened Wetwell Vent valves and the SRVs for Reactor Depressurization. The fault tree logic (e.g., gate U1_VENT_N2FLEX) models that the same FLEX Nitrogen bottles and operator action supports both the Hardened Wetwell Vent valves and the SRVs for Reactor Depressurization. 2.Based on information provided by BFN, the (permanent) Nitrogen bottles providing backup supply to the Hardened Wetwell Vent valves appear to be different than the (portable, i.e., on carts) Nitrogen bottles providing backup supply to the SRVs for Reactor Depressurization. 3.The fragility analysis should distinguish between the Nitrogen bottles providing backup supply to the Hardened Wetwell Vent valves and the Nitrogen bottles providing backup supply to the SRVs for Reactor Depressurization. The same fragility group was applied to both N2 bottle sets - (1) for portable N2 bottles (carts) used for backup supply to ADS SRVs, and (2) for permanently installed N2 bottles for backup supply to HCVS AOVs. 4.The human reliability analysis (HRA) should distinguish between the operator action to align the (permanent) Nitrogen bottles providing backup supply to the Hardened Wetwell Vent valves and the operator action to align (portable) Nitrogen bottles providing backup supply to the SRVs for Reactor Depressurization. The current HRA for operator action HFA_OPS_FLEXN2ALIGN references procedure 1-EOI Appendix-20H (Alternate N2 Supply to SRVs) 5.Provide the MAAP run (identified as from 'EVB') that is used as the basis for the time available (i.e., Tsw) of 6.27 hours for operator action HFA_OPS_FLEXN2ALIGN.	assumption of complete correlation. A new pseudo UNID name was created for the HCVS N2 bottles and a new component failure basic event TKURP0TNK_N2HCVS and a newly created operator action HFA_OPS_HWWVN2ALIGN are associated with primary containment vent. Fragility grouping was evaluated separately for both sets on nitrogen bottles, although both were determined to be rugged and assigned to SEIS_0-20. Since the HCVS N2 bottles are stored in the diesel building, the structure fragility corresponds to SEIS_BLD-DGB for basic event TKURP0TNK_N2HCVS. SEIS_BLD-RB was the structure fragility of the SRV N2 bottles since they reside in the reactor building. N2 bottle fault tree logic was updated to ensure that the basic events for flex n2 bottle alignment for RPV depressurization correctly feed in inputs for basic events TKURP0TNK_N2FLEX, and logic to backing up hardened wet well vent correctly feed in inputs from basic event TKURP0TNK_N2HCVS. The associated mapping was updated in the 'Fragility_To_Comp' table and the human actions in the 'FireInitiatorHRA' table. All updates were populated to the CAFTA database.			that the BFN SPRA response to this F&O is a PRA Maintenance activity and not an Upgrade because the activity includes minor modeling enhancements (i.e., add basic events and gates to fault tree logic and revisions to FRANX input file) using no new methods or calculations.	HFA_OPS_HWWVN2ALIGN are associated with primary containment vent. HEPs for both of the operator actions were developed appropriately. Fragility grouping was evaluated separately for both sets of nitrogen bottles, although both were determined to be rugged and assigned to fragility group ID SEIS_0-20 "Plant Ruggedness Fragility". Since the HCVS N2 bottles are stored in the diesel building, the structure fragility corresponding to SEIS_BLD-DGB was modeled as a limiting fragility for basic event TKURP0TNK_N2HCVS. Similarly, SEIS_BLD-RB was the structure fragility modeled as a limiting fragility for the SRV N2 bottles since they reside in the reactor building. N2 bottle fault tree logic was updated to ensure that the basic events for FLEX N2 bottle alignment for RPV depressurization correctly feeds into inputs for basic events TKURP0TNK_N2FLEX, and logic for backing up hardened wet well vent correctly feeds into inputs from basic event TKURP0TNK_N2HCVS. The associated mapping was updated in the 'Fragility_To_Comp' table and the human actions in the 'FireInitiatorHRA' table. All updates were populated to the CAFTA database. This F&O is assessed as CLOSED.	
SPR	C-SPR-A3	Not Met	26-4	There is no discussion of the available seismic risk	SPRA notebooks provide no evidence that this assessment of	A table was assembled listing a review of external events for applicability to seismic events. This list is derived from	Methodology, Inputs and Model Notebook; PRA Evaluation BFN-0-19-062	Maintenance	The review team concurs with the assessment that the changes to the	Table 6-8 of the (MDN-000-999-2019-000267, Rev. 1) provides a summary of responses to earthquakes for several plants and identifies the relevance of those responses to the BFN SPRA.	Met

Table A-2 BFN SPRA F&O Closure Review Consensus Table

RU	SR	PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
				evaluations for other nuclear plants or industry experience.	other nuclear plants or industry experience was performed	reference 3002000709, Seismic Probabilistic Risk Assessment Implementation Guide, EPRI, Palo, Alto, CA.			SPRA constitute Maintenance. This comparison was made to document the check made of the adequacy of the set of initiating events considered in the SPRA for BFN. No changes were made to the SPRA based on this comparison.	A comparison of the seismically induced initiating events from three other BWRs to those relevant to BFN was made. This comparison is summarized in a table in the PRA Evaluation Response (BFN-0-19-029, Rev. 0), which describes in detail the manner in which each of the findings from the peer review was addressed. This F&O is assessed as CLOSED.	
SPR	C-SPR-A4	Met	26-5	Table 4-3 in the Seismic Methodology, Inputs, and Model Notebook includes a listing of the seismic initiating events that were included in the model. It includes a discussion of where the initiating events were inserted into the model.	For Break Outside of Containment (BOC), the BFN SPRA identifies Feedwater and Main Steam piping failures in fragility group SEIS_23-2_BOC. However, the following additional BOC initiators have not been identified or assessed. No fragility value was provided for BOC for the following system in table 6-8.(ISLOCA) or in the SPRA model, whereas this info was provided for FW /MS (SEIS_23-2_BOC 1. RCIC - for BOC, not modeled and no basis provided for screening 2. HPCI - for BOC, not modeled and no basis provided for screening 3. SDV - for BOC, not modeled and no basis provided for screening 4. MSL Drains - for BOC, not modeled and no basis provided for screening 5. RWCU - for BOC, not modeled and no basis provided for screening. (Note: ISLOCA is different than BOC (in BWRs). These issues made it difficult for PRT SPR reviewers to assess BOC.)	The seismic impact of these pipe lines on the system functions has been modeled in the SPRA model; Given the rugged piping and relative high fragilities of the valves, given an earthquake, the risk contribution from these lines as an initiating event is negligible and can be capped by the reactor building failure. The failure of the reactor building is modeled as leading directly to core damage. Due to the rugged piping line and high fragilities of the associated valves, the risk contribution from these piping lines is very small and negligible. No further modeling change is necessary.	PRA Evaluation BFN-0-19-062	Maintenance	The review team concurs with the assessment that the changes to the SPRA constitute Maintenance. Additional documentation of the potential for breaks outside containment was provided, but not changes were made to the PRA models to address this Finding.	The rationale for screening of specific lines whose failure might constitute a break outside containment is provided in Table 6-8 of the Seismic Methodology, Inputs and Model Notebook (MDN-000-999-2019-000267, Rev. 1). In the summary of actions taken to address Finding 26-5 in the PRA Evaluation Response (BFN-0-19-029, Rev. 0), further details supporting these rationales are provided. These summaries include reference to specific penetrations, facilitating review of each of the lines cited above based on the information in Table 6-8. No issues were identified with respect to the treatment of these lines. The amplified rationales provided in BFN-0-19-029 provide additional perspective on the ruggedness of the respective lines. This F&O is assessed as CLOSED.	Remains Met
SPR	C-SPR-F1	Met	26-6	The seismic plant-response analysis and quantification was documented in 'SPR' Notebooks (Seismic Methods, SQU) in a manner that facilitates PRA applications, upgrades, and peer review.	1. Ref. BFN SPRA Methods Notebook do not explicitly state that 'no other secondary hazards (from SHA-I2) were explicitly retained in the SPRA.' 2. Ref. BFN SPRA Methods Notebook, table E-1, page 364, fragility group SEIS_23-2 _BOC (Feedwater piping (BOC)) also apply to main steam piping.	Methodology, model and inputs notebook was updated in section 6.2.5 to include the statement "No other secondary hazards were explicitly retained in the SPRA." The description for SEIS_23-2 was updated to include main steam in Table E-1 of the methodology, inputs and model notebook.	Methodology, Inputs and Model Notebook	Maintenance	The review team concurs with the assessment that the changes to the SPRA constitute Maintenance. The changes affect only documentation of aspects of the SPRA.	Review of available documentation confirmed that appropriate changes have been made. At the end of Section 6.2.5 of the Seismic Methodology, Inputs and Model Notebook (MDN-000-999-2019-000267, Rev. 1), a summary is provided of the secondary hazards considered and screened for BFN. This summary includes the statement that no other secondary hazards were explicitly addressed in the SPRA. The entry for fragility group SEIS_23-2_BOC in Table E-1 of the same document has been revised to note that it included feedwater and main steam piping. This F&O is assessed as CLOSED.	Remains Met

Table A-2 BFN SPRA F&O Closure Review Consensus Table

RU	SR	PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
SPR	C-SPR-F2	Met	26-8	The process to perform the BFN SPRA seismic plant-response analysis and quantification was documented in the SEL, Seismic Methods, Model Development _ Q1,Q2,Q3 (027), SHRA, SQU, Truncation (028) and Convergence (022) Notebooks, and associated PRA Model documentation.	1.In section 4.4 of MDN0009992019000266, HFA_OP-LOCKOUT_4kVSDBD_S is incorrect. It should be HFA_OPS_4KVSDBDRESET. 2. In the SEISMIC METHODOLOGY, INPUTS, AND MODEL 6.3.1.1 SIET Top Event S-DCD (SEIS-08) The SIET sequence SEIS-08 covers seismic failures that lead directly to core damage. Figure 6- shows. 3. SEISMIC METHODOLOGY, INPUTS, AND MODEL The SIET top events in page 74 and 75 doesn't match table 6.9 4. SEISMIC METHODOLOGY, INPUTS, AND MODEL shutdown boards have been unitized.it should be utilized. 5. Quantification, Sensitivity and Uncertainty Notebook Table 8-2 last column is wrong. 6. SEISMIC METHODOLOGY, INPUTS, AND MODEL occurrence of the failures from the switch gear to. It should be switchgear	Operator action HFA_OP-LOCKOUT_4kVSDBD_S in section 4.4 of the HRA Notebook was corrected to read HFA_OPS_4kVSDBDRESET. In the Seismic Methodology, Inputs and Model Notebook, in section 6.3.1.1, the line that reads, "The SIET sequence SEIS-08 covers seismic failures that lead directly to core damage. Figure 6- shows.." is now updated to read Figure 6-5. The mismatch in Table 6-9 and Figure 6-3 was corrected in the Methodology, Inputs and Model Notebook. Section 6.5.16 was corrected to "should be utilized." in the Methodology, Inputs and Model Notebook. Table 8.2 of the Quantification, Sensitivity and Uncertainty Notebook was corrected in the last column. Section 5.3.2 of the methodology, inputs and model Notebook was updated from "occurrence of the failures from the switch gear" to read "switchgear."	Methodology, Inputs and Model Notebook; Quantification, Sensitivity and Uncertainty Notebook; Seismic PRA Human Reliability Analysis Notebook	Maintenance	The review team concurs with the assessment that the changes to the SPRA constitute Maintenance. The changes correct minor documentation errors in various parts of the SPRA.	Review of available documentation confirmed that appropriate changes have been made: • HFE HFA_OP-LOCKOUT_4kVSDBD_S is now HFE HFA_OPS_4KVSDBDRESET in the Seismic PRA HRA Notebook (MDN-000-999-2019-000266, Rev. 1) and in the HRA Calculator file (BFN SPRA 8-21-19.hra). It has also been incorporated into the PRA model via the use of FRANX. • In Section 6.3.1.1 of the Seismic Methodology, Inputs and Model Notebook (MDN-000-999-2019-000267, Rev. 1), the figure number has been corrected to Figure 6-5. • The events in Figure 6-3 and the entries in Table 6-9 of the Seismic Methodology, Inputs and Model Notebook have been made consistent. • In Section 6.5.16 of the Seismic Methodology, Inputs and Model Notebook, "unitized" has been corrected to "utilized". • In Table 8-2 of the Seismic PRA Quantification, Sensitivity and Uncertainty Notebook (MDN-000-999-2019-000268, Rev. 1), the contributions by percent (the last column in the table) has been corrected. • In Section 5.3.2 of the Seismic Methodology, Inputs and Model Notebook, "switch gear" has been corrected to "switchgear". This F&O is assessed as CLOSED.	Remains Met
SPR	C-SPR-B2	Met	26-9	Under gate 'U1S_REMOTE_SHUTDN', the failure of the remote shutdown is model as a failure of human action 'U1_CREVACSTDNFAILS_S'. There is no failure of the remote shutdown associated to equipment failure.	Failure of equipment due to seismic event especially panel inside the remote shutdown room should be modeled. Without adding this to the model it is not possible to determine the importance of the equipment failure or its contribution to risk:	Three pseudo components representing the backup control panel were created in the components table within the FRANX software. The components were then tied to the human action event Ux_CREVACSTDNFAILS_S in the "Fragility_to_Components" table so that the failure of the panel fails the operator action. The new mapping was populated in the CAFTA database.	Methodology, Inputs and Model Notebook; PRA Evaluation BFN-0-19-062	Maintenance	The closure team assessed and agreed with the TVA determination that the BFN SPRA response to this F&O is a PRA Maintenance activity and not an Upgrade because the activity includes minor modeling enhancements (i.e., revisions to FRANX input file) using no new methods or calculations.	Three components (i.e., one for each unit) representing the backup control panel were created in the components table within the FRANX software. The three components were modeled with SPRA fragility ID SEIS_5-3, "RCIC Local Control, BFN-1/2/3-LPNL-925- 0031 and 0032". The components were then tied to the human action event Ux_CREVACSTDNFAILS_S in the "Fragility_to_Components" table so that the failure of the panel fails the operator action. The new mapping was populated in the CAFTA database. This F&O is assessed as CLOSED.	Remains Met
SPR	C-SPR-F2	Met	26-10	The process to perform the BFN SPRA seismic plant-response analysis and quantification was documented in the SEL,	1.CST Am fragility value is incorrectly reported as 0.1 g in FRANX Fragility Editor (Mapped to SEIS_HLF, High Likelihood of Failure Fragility) and does not agree with	Section 6.5.13 of the Methodology, Inputs and Model Notebook was updated to accurately reflect the new CST fragility value of 0.42g. The	Methodology, Inputs and Model Notebook	Maintenance	The review team concurs with the assessment that the changes to the SPRA constitute	The median capacity for the CST has been corrected in Table E-1 of the Seismic Methodology, Inputs and Model Notebook (MDN-000-999-2019-000267, Rev. 1) to be the updated value of 0.42. The updated value (for	Remains Met

Table A-2 BFN SPRA F&O Closure Review Consensus Table

RU	SR	PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
				Seismic Methods, Model Development _ Q1,Q2,Q3 (027), SHRA, SQU, Truncation (028) and Convergence (022) notebooks, and associated PRA Model documentation.	value of 0.3 g reported in Methods Notebook, Section 6.5.13.	FRANX table was changed to reflect this value as well.			Maintenance. The changes entail updating a fragility value and ensuring that it is correctly reflected in the documentation. The update did not entail the use of new methods or require significant changes to the PRA model or results.	group SEIS_19-3) is also entered into the fragility table in the FRANX file (e.g., in BFN_Seismic_Rev1_U1CDF.franx). This F&O is assessed as CLOSED.	
SPR	C-SPR-C2	Met	27-1	Additional SSCs, e.g., structures, passive components, panels, and cabinets / relays that house PRA components not included in IEPRA were added to SPRA. These are listed in SEL Notebook Appendix D-2, Table of New Seismic Basic Events, and include but are not limited to the following. 1. Structures (Rx, DG and IPS Buildings) 2. Block wall failures 3. Rule of the Box failures 4. Relay chatter 5. Various LOCA types, including ISLOCA 6. IPEEE / A-46 SSEL SSCs. Distribution systems (e.g., cable tray, HVAC ducts) were assessed via fragility walkdown and screened out from being added to SEL. Distribution piping failures (ex. Fire piping) and associated impacts were identified in other SR SPR-C5.	Permanent N2 bottles in U1/U2 and U3 DG corridor that were added as a FLEX strategy for HCVS B/U N2 supply are not included in SEL but are credited in the SPRA via fragility basic events (TKURP0TNK_N2FLEX, SSC ID C0_TNK_N2FLEX-U- Gxxx) and HEP basic events (HFA_OPS_FLEXN2ALIGN) in BFN SPRA for each unit. And, the N2 bottles have no UN ID cited in the SPRA notebooks.	The UNIDs representing the HCVS nitrogen bottles, storage racks and associated tubing, valves and instrumentation were added to the SEL. The permanent equipment installed by the DCN related to the N2 storage only included the N2 bottle storage racks. The bottles themselves do not have UNIDs since they are not permanent and can be recharged and replaced as needed.	Seismic PRA Seismic Equipment List Notebook	Maintenance	The review team concurs with the assessment that the changes to the SPRA constitute Maintenance. The changes entailed updates to documentation only, and no changes to the SPRA model itself.	A review of the Seismic Equipment List (MDN-000-999-2019-000269, Rev. 1) and a search of the Access database containing the SEL (Composite SEL_draft_20190910 – Copy.accdb) confirmed that the following two entries had been added: • BFN-0-LPNL-925-6100 – U2/U3 Nitrogen gas bottle rack • BFN-1-LPNL-925-6100 – Nitrogen Gas Bottle Rack This F&O is assessed as CLOSED.	Remains Met
SPR	C-SPR-E7	Met	27-2	The sources of uncertainty are discussed in the Quantification Notebook (Section 9.0) and assessed sources of model uncertainty and their probable effects on the model are discussed in Appendix G). In addition, sensitivity studies were performed in the Quantification Notebook (Section 10.0) to address the potential effects of changing various variables, parameters,	FLEX PPs / GENs were not adequately considered for inclusion in the SPRA for ELAP or non-ELAP scenarios, as a Rx injection source, for Seismic CDF / Seismic LERF reduction, based on feasibility to deploy within time needed for success given expected battery life for HPCI/RCIC (4 hrs), MAAP run, time to TAF (Rx core damage) and FLEX deployment time. TVA BFN PRA engineers responded	A sensitivity study has been performed to evaluate the risk significance of FLEX. A recovery file was created only allowing in sequences where FLEX would be implemented. Although FLEX may reduce the risk of certain accident sequences, BFN risk evaluation BFN-0-19-074 concluded that the inclusion of FLEX recovery does not have a significant effect on CDF and	PRA Evaluation BFN-0-19-074	Maintenance	The closure team assessed and agreed with the TVA determination that the BFN SPRA response to this F&O is a PRA Maintenance activity and not an Upgrade because the activity includes various	A sensitivity study has been performed to evaluate the risk significance of crediting additional FLEX strategies. A recovery file was created to credit FLEX in a simplified manner in sequences where FLEX would be implemented (e.g., ELAP sequences with initial success of HPCI or RCIC to allow sufficient time to align FLEX equipment). FLEX was not credited for scenarios with judged inadequate time to align FLEX (e.g., ELAP sequences without HPCI and RCIC, ATWS). The assumptions for the FLEX sensitivity case were reasonable and appropriate. BFN risk evaluation BFN-0-19-074 concluded that the inclusion of FLEX recovery does not have a significant overall effect on	Remains Met

Table A-2 BFN SPRA F&O Closure Review Consensus Table

RU	SR	PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
				or assumption in the SPRA model.	to Peer Review Team Q's PJT-06, PJT-07 and PJT-21, and reported that (1) FLEX Strategy Validation Report for BFN Units 1, 2, and 3 (03/23/2018) allows up to 8 hrs to deploy FLEX PPs and GENs. The report states that during FLEX validation, it was done in 6 hours and 10 minutes (6.17 hrs). A MAAP case associated with a different HEP, HFA_OPS_FLEXN2ALIGN (N2 bottles, showed a Tsw of 6.27 hours are available prior to core damage based on initial RPV injection from HPCI or RCIC for 4 hours. BFN PRA engineers reported since only 6 minutes of time margin exists between 6.27 hrs available and 6.17 hrs to deploy, the time margin was too small to warrant crediting FLEX PPs / GENs in SPRA. However, the follow 2 factors warrant more consideration of FLEX for the SPRA, especially since FLEX has been included in the SPRAs developed and peer reviewed IAW the PRA Standard (Add. B or Code Case) in the past 2 years thus making it a state of practice. 1. The SPRA dominant accident sequences are GTRAN_005 & 006 (total of 43%) where HPCI & RCIC are available for 4 hrs prior to battery depletion at 4 hours for S-R 250 VDC, whereas less dominant seq. include GTRAN_011 & 012 (total of 13%) where HPCI & RCIC are initially unavailable and use of FLEX PPs / GENs does not avert early core damage (< 4 hours). 2. Per BFN 0-FSI-1, page 2, safety-related battery life for HPCI/RCIC can be extended from 4 hrs to 11 or 12 hours if S-R DC load shed is accomplished. BFN 0-FSI-1, page 2 states "shed loads from BB 1, 2, 3 in accordance with 0-FSI-3F within 1 hour to extend battery coping time to 12 hours, or within 2 hours to extend coping time to 11 hours." Thus, this can extend a MAAP run Tsw from 6.27 hours to > 12 hours when accounting for Rx level	LERF and does not need to be added to the model.			modeling enhancements (i.e., revisions to flag and recovery files) using no new methods or calculations.	CDF, LERF, or the risk insights and does not need to be explicitly added to the Base Case SPRA model. This F&O is assessed as CLOSED.	

Table A-2 BFN SPRA F&O Closure Review Consensus Table

RU	SR	PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
					decrease to TAF and resulting in core damage.						

A.6 Summary of Technical Adequacy of the Seismic PRA

The set of SRs from the PRA Standard [8] that are identified in Tables 6-4 through 6-6 of the SPID [2] define the technical attributes of a PRA model required for a SPRA used to respond to implement the 50.54(f) letter. The conclusions of the peer review discussed above and summarized in this submittal demonstrates that the BFN SPRA model meets the expectations for PRA scope and technical adequacy as presented in NRC RG 1.200, Rev. 2 [45] as clarified in the SPID.

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

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

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

The BFN SPRA reflects the as-built and as-operated plant as of the cutoff date for the SPRA, January 2016.

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

The BFN SPRA was built on the BFN Internal Events and Internal Flooding PRA model. The following sections describe the peer reviews performed on these models.

The BFN Internal Events (excluding Internal Flooding) PRA was subjected to a full scope peer review in May 2009 [54], in accordance with the requirements of NEI 05-04 [55]. The review covered all technical elements in Part 2 plus the configuration control element from the PRA Standard RA-Sa-2009 that was endorsed by RG 1.200, Rev. 2. Table A-3 presents the results of this peer review.

Table A-3 Internal Events PRA Model 2009 Peer Review SR Capability Category Distribution

Capability Category	Number	Percent
Not Met	53	20
I	10	4
I/II	10	4
II	26	10
II/III	20	8

Capability Category	Number	Percent
III	3	1
Met (All)	140	53
Not Applicable	2	1
TOTAL	**264**	**100%**

Internal Flooding was not included in the scope of this review. Of these 264 PRA Standard SRs reviewed, approximately 76% are supportive of Capability Category II or greater. A total of 189 unique F&Os were generated by the peer review team, from which 95 were Findings, 92 were Suggestions, and 2 were Best Practices.

A separate Internal Flooding Focused-Scope Peer Review (FSPR) was performed on the BFN PRA in September 2009 [56]. The review covered all technical elements from the PRA Standard Part RA-Sa-2009 [57]. Table A-4 presents the results of this FSPR.

Table A-4 IF PRA Model 2009 Focused-Scope Peer Review SR Capability Category Distribution

Capability Category	Number	Percent
Not Met	26	42
I	3	5
II or better	30	48
Not Applicable	2	3
Not Reviewed	1	2
TOTAL	**62**	**100%**

The BFN internal flood PRA met Capability Category II or higher for about 48% of the applicable SRs. The BFN internal flood PRA met Capability Category I level for an additional 5% of the applicable SRs. A total of 50 F&Os were generated during this focused-scope peer review, including 29 Findings and 21 Suggestions.

The key problem areas for the internal flood PRA were documentation and flood scenario development. All 15 documentation SRs were rated as not meeting the standard requirements. The primary problem associated with documentation was lack of details, numerous inconsistencies, and incomplete information in the input data, process, and results. The internal flood PRA was not prepared in a manner that can facilitate PRA applications, upgrades and peer review. To be consistent with the applicable SRs, more effort was needed to enhance the documentation. The major problem associated with the flood scenario development was that the development of flood scenarios was not rigorously performed. Many flood areas, flood sources, and flood scenarios were dismissed without adequate considerations of all the possible flooding effects that may cause damage to SSCs credited in the PRA. As a result, the total number of flood scenarios that were quantitatively evaluated was far less than expected, and the results from some top internal flood-induced risk contributors were not completely realistic.

The internal flooding model was updated, and an additional Internal Flooding model FSPR was completed in September 2018 [58], again covering all technical elements from the PRA Standard Part RA-Sa-2009 Part 3 [57]. This internal flood FSPR was conducted concurrently with an Internal Events F&O closure review activity [59]. The internal flood FSPR and associated conclusions supersedes the internal flood PRA Peer Review and associated findings from 2009. As a result of this FSPR, all existing F&Os were considered to be no longer applicable, and 11 new F&Os were generated. A total of 7 Findings, 3 Suggestions and 1 Best Practice were reported by the peer review team. The results of this assessment are reported in Table A-5.

Table A-5 Internal Flood PRA Model 2018 Focused-Scope Peer Review SR Capability Category Distribution

Capability Category	Number	Percent
Not Met or CC I	7	11
II or better	55	89
Not Applicable	0	0
Not Reviewed	0	0
TOTAL	**62**	**100%**

The peer review team concluded that, from a technical perspective, the internal flood analyses appeared to address the appropriate inputs and outputs, and the modeling approaches appeared sound. In addition, the team concluded that the changes made appeared to meet most of the requirements in the PRA Standard at or above Capability Category II, with the caveats provided with the F&Os.

<u>Internal Events PRA F&O Closure Review</u>

The 95 Internal Events Finding F&Os identified in the peer review in May 2009 were subjected to a F&O Resolution FSPR in 2015 [60], which followed the guidance from NEI 05-04 [55]. The review was conducted over a three-day period by a team of four independent PRA experts, and included a consensus process to determine the adequacy of the resolution to each reviewed Finding. Following that review, there were 48 Findings that remained open, including 9 that were not assessed due to time constraints.

A subsequent F&O Closeout Assessment was completed in September 2018 at the TVA Chattanooga offices [59] for the 48 Internal Events Findings that remained open. This assessment was completed in accordance with the process documented in Appendix X to NEI 05-04/07-12/12-13 [9], as well as the requirements published in the PRA Standard (RA-Sa-2009) and RG 1.200 Rev. 2 [45], including NRC expectations. A team of three independent PRA experts performed the F&O reviews along with the consensus sessions. The review met the Appendix X requirement that each F&O review include two qualified reviewers. Furthermore, the team examined the changes made to the BFN PRA model, data, and documentation to address the findings to determine if the Capability Category II (or better) requirements of the PRA Standard, including clarifications imposed by RG 1.200, Rev. 2 were met.

The closure peer review team had significant PRA experience, and each team member confirmed they were not TVA employees, had no involvement in development of the BFN PRA or performance of risk applications for BFN, and no conflicts of interests, incentives, or disincentives.

The closure review team concluded that all but 10 of the 48 F&Os reviewed met the criteria for closure. In addition, an assessment was performed to determine if the F&O resolution resulted in an upgrade to the PRA or used new PRA methods. The peer review team concluded that those F&Os that were closed did not fall in the upgrade category and did not use new PRA methods (those F&Os remaining open were not assessed). Table A-5 presents the BFN Internal Events PRA F&O Closure Review Consensus, which includes the 10 Internal Events Finding F&Os that remain open.

Finally, the 7 Finding F&Os identified in the 2018 internal flood FSPR remain open, as no formal closure review has been performed to date. These Internal Flooding Finding F&Os that remain open are listed in Table A-6.

The open Internal Events and Internal Flooding F&Os were reviewed by the BFN SPRA peer review team. The SPRA peer review team created one F&O (19-1) that stated two of the open Internal Events may impact the SPRA. That F&O has been closed by the closure team as discussed in Table A-1. No Internal Flooding F&Os were identified as having an impact on the SPRA.

Table A-6 BFN Internal Events PRA F&O Closure Review Consensus Table

RU	SR	Prior PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
AS	SY-B14 AS-B3	AS-B3 is Not Met SY-B14 is Met.	1-6	The sequence descriptions generally include a description of the sequences, but the phenomenological conditions created are not specifically identified. Some references to phenomenology are provided but not consistently (e.g., ATWS sequence descriptions conclude with the statement "There are no phenomenological conditions identified.")	The SR calls for identification of the phenomenological conditions for each sequence.	Revision 3 of the Accident Sequence Notebook Section 6.3.4.5.7 has been updated to include the following: "The suppression pool suction strainers are modeled under gate Ux_LPC_G11 for MLOCA where x is the Unit number. Suppression pool suction strainer plugging would fail the suppression pool cooling pathway." Revision 3 of the Accident Sequence Notebook Section 6.3.5.5.7 has been updated to include the following: "The suppression pool suction strainers are modeled under gate Ux_LPC_G10 for LLOCA where x is the Unit number. Suppression pool suction strainer plugging would fail the suppression pool cooling pathway. During an excessive LOCA SPC is not credited."	NDN-000-999-2007-0036, Rev.5; BFN-0-16-031	Maintenance	This resolution is limited to enhancing documentation to provide a more detailed discussion of the quantification results. (1) No new methods were used. (2) There was no change in PRA scope. (3) There was no change in PRA capability.	The documentation provides explanation of the failures defined and the basis for assessment related to the postulated failure mode associated with LERF. This F&O is assessed as Closed.	AS-B3 was previously Not Met and is now Met. SY-B14 was previously Met and remains Met.
AS	AS-A7	AS-A7 is met at Cat I/II	5-5	Section 6.3.2.4.1 of the Accident Sequence Analysis states that if Alternate Rod Insertion succeeds and either the recirculation pumps fail to trip of the SRVs fail to open, then a non-ATWS LOCA occurs which is not modeled in the PRA. While this new LOCA might be quantitatively insignificant, no qualitative argument is made to justify its omission.	The omission of this sequence could result in an incorrectly low CDF or cause the analyst to miss important insight about the event.	The AS Notebook was updated to explain that not only would the ATWS Induced LOCA probability be below the ASME initiator frequency cutoff recommended by IE-C4 of 1e-7 but would also be bounded by other LOCA IEs.	NDN-000-999-2007-0036, Rev.5; NDN-000-999-2007-0041, Rev.8	Maintenance	This resolution is limited to enhancing documentation to provide a more detailed discussion of the quantification results. (1) No new methods were used. (2) There was no change in PRA scope. (3) There was no change in PRA capability.	The assessment utilizes a screening approach based on low frequency of occurrence as is allowed by the Standard. The discussion provides the supporting information to conclude that the sequence can be excluded and is bounded by other similar scenarios. This F&O is assessed as Closed.	AS-A7 was previously met at Cat I/II and remains met at Cat I/II.
DA	DA-C6	DA-C6 is Not Met.	1-17	Reviewed DA.01. The source of demands is not discussed. Based upon discussions with the PRA staff, exposure is collected directly from plant data systems and is therefore actual component exposure. However, post-maintenance testing demands are also included in these numbers and are not removed.	Post-maintenance testing must be excluded from the exposure data per the SR.	As mentioned in the DA Notebook, the only demands that are included in the data analysis update of failure rates are those that come directly from PEDs, from the IST database or from the system engineer directly. The IST database gives just those successful demands that occur for each test (i.e., no post	NDN-000-999-2007-0033, Rev. 9 (DA.01); BFN-0-15-079; BFN-0-18-006; SY.21 (Safety Relief Valve System Notebook)	Open	none	BFN uses an automatic demand counter to populate the data. As such this would include all related surveillance, maintenance and operational demands. Because the system may count additional demands for PMTs BFN has estimated these additional demands and performed sensitivities to support the impact on the failure rates. Although the sensitivities may justify a minimal impact, it does not meet the SR (DA-C6). This F&O is assessed as Open.	DA-C6 remains Not Met.

Table A-6 BFN Internal Events PRA F&O Closure Review Consensus Table

RU	SR	Prior PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
						maintenance demands included). PEDs/ the system engineer gives the actual number of demands the component observes which could potentially include post maintenance demands, however a sensitivity was performed (BFN-0-15-079) which shows that the model is not sensitive PMTs.					
DA	DA-C10	DA-C10 is not Met.	1-22	There is no discussion of the process to be applied in the use of surveillance test data. The use of this data is required for situations in which there is no MR data available (for example), so a process for its use should be in place.	All levels of capability in this SR indicate that the process for use of surveillance data needs to possess specific attributes. There is no process defined.	DataWare takes the actual component demands and hours as documented in PEDS. When PEDS does not track the component being looked at, either the IST database was used, or the system engineer was contacted to figure out the number of demands/hours that occurred for that particular component. This information is described in the DA Notebook and Table 13 shows specifically where the success information comes from. The process described in the DA notebook was updated to clarify how the data collection is performed.	NDN-000-999-2007-0033, Rev. 10 (DA.01), SY.07	Maintenance	The change provides documentation describing how the data is collected and reviewed. (1) No new methods were used. (2) There was no change in scope of the PRA. (3) There was no change in the capability of the PRA.	BFN uses an automatic demand counter to populate the data. As such this would include all related surveillance tests. Because the system may count additional demands from incomplete tests or unplanned operation as a success, the data should be reviewed adjusted as necessary to account for these demands. BFN reviewed all data for inappropriate inclusion of data and made adjustments, if necessary. This additional step was added to the Data Notebook (DA.01) by Revision 10 and is described in Section 7.1. In addition, Section 7.3 and Assumption 7 of DA.01 describe the process to be followed when the electronic data system does not include the PRA component failure mode of interest, which includes the requirement to review test procedures. A sample of system notebooks (SY.07 and SY.21) were reviewed confirming that a listing of applicable test procedures and a discussion for how demands are determined was used for these cases. This F&O is assessed as Closed.	DA-C10 is now Met.
DA	DA-B2	DA-B2 is not Met.	5-3	The data analysis does not appear to consider outlier components.	The inclusion of outlier components can incorrectly impact the failure rate assigned to a component group. Such outlier components should be placed into a separate suitable component group.	During the data collection the current grouping of the component is used, however the data analyst looks at the data and any components that are never tested would have little or no data to update the failure rate of the typecode in the model with. These are looked at and determined whether it is more appropriate to keep them within the same grouping as they are the same type of component, experience the same type of environmental conditions, and have about the same type of failure rates or whether they should be put into a separate grouping. This was the intent of bullet 3 of Section 5.0. As shown in the DA Notebook Appendix E, the prior and posterior distributions were reviewed, and it was determined whether generic data was a suitable representation for BFN.	NDN-000-999-2007-0033, Rev.9 (DA.01)	Maintenance	The change is documentation only. (1) No new methods were used. (2) There was no change in scope of the PRA. (3) There was no change in the capability of the PRA.	The process documented in Attachment E of DA.01 ensures that the data groupings are compared to generic data to identify outlier behaviors. Where plant data is outside the generic 5% or 95% bounds, plant-specific data is used. The bases that outliers components are not inappropriately grouped is documented in DA-01. No outliers were identified, and these results are unchanged. This F&O is assessed as Closed.	DA-B2 is now Met.

Table A-6 BFN Internal Events PRA F&O Closure Review Consensus Table

RU	SR	Prior PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
HR	HR-C3 HR-D5	HR-C3 is not Met. HR-D5 is not Met.	2-14	HFL_1003_CCFT0056 is Common cause miscalibration of all 4 level transmitters, inspection of the fault tree shows that specific pairs of failures (AC, BD) would also cause a failure to initiate the logic. These CCF pairs should be added to the model. This will apply to other miscalibration CCFs also.	The pair CCFs will have a higher value than the 4 of 4 event thus impact the results.	HFL_1003_CCFT0056 is Common cause miscalibration of all 4 level transmitters; Events for the critical 2 of 4 failure groups have not been added to the model and the original F&O issue has not been addressed. • ADDED pre-initiator HEs for miscalibration of 2 of 4 instruments to the PRA model. See gates U1_LM_G01, U2_LM_G01, U3_LM_G01	BFN IE PRA Model, Rev 8	Maintenance	The change involved adding the basic events for 2 of 4 CCF events. (1) No new methods were used. The model change was made using processes and tools. (2) The scope included several new basic events. There was no significant change to the risk insights. (3) There was no change in PRA capability.	The Calibration CCF of 2 of 4 instruments has been added to the model. This F&O is assessed as Closed.	HR-C3 is now Met. HR-D5 is now Met.
HR	HR-G2	HR-G2 is Met.	4-18	Some operator actions assume that the execution failure probability (Pe) is including: HFA_0_ADSINHIBIT, HFA_0_ATWSLEVEL, HFA_0024RCWINTAKE, HFA_0027INTAKE, HFA_0IR2_LPI, HFA_1063SLCINJECT, HFA_0024IFISOL Example 1: Several operator actions for ATWS scenarios (e.g., HFA_1063SLCINJECT: Failure to SLC in response to an ATWS event) assume the execution failure probability (Pe) is 0.0. Example 2: Operator action HFA_0024RCWINTAKE (Failure to clear debris at intake before reactor scram) assumes an execution error of 0.0 based on the following: 'Cleaning traveling screens does not relate to a series of manual actions, but to an effort among several operators. It is assumed that, if the action is initiated within 1 hr, it will be successful.' The same rationale is provided for no execution error in HFA_0027INTAKE.	Execution failure is a required part of the HEP calculation, and the argument for ignoring execution failure is not necessarily compelling, especially for maintaining level (HFA_0_ATWSLEVEL). Some of the actions for which Pe is not considered are important to the overall results. Note 1: The explanation given for no execution failure for HFA_0_ATWSLEVEL describes the actions required for starting SLC (HFA_1063SLCINJECT). Note 2: Cleaning debris from traveling screens is not a simple action, an assumption, that if the actions are started, they are guaranteed to be completed in 1 hour, is not justified.	• Execution error has not been included for ADS inhibit (HFA_0_ADSINHIBIT). This is modeled only for ATWS in the PRA. There is a single step to implement this action, errors of omission are integral to the cognitive error to omit the action. Errors of commission are neglected because the action to inhibit ADS is unique (no transition to any EOI Appendix is required, and there are several places in the EOI that call for inhibiting ADS), and because it is routinely performed for every reactor scram, graphically distinct and performed after SLC. • Execution error was added for SLC. This is a time critical operator action, and the EOI specifies the appropriate steps required in EOI-Appendix 3A. While the actions are simple, these require transition between procedures for the execution, so it is appropriate to include execution errors. • HFA_0_ATWSLEVEL -Execution errors are included for this event. NO CHANGE. • HFA_0024RCWINTAKE - Execution error set to zero and it deemed not necessary to add detail for this activity. Clearing traveling screens does not	NDN-000-999-2007-0032, Rev 6	Maintenance	The update involves updating HEP execution failure probabilities. (1) No new methods were used. The HEP update uses the HRA Calculator consistent with current practice. (2) The scope was limited to a limited set of HFEs. (3) There was no change in PRA capability.	Execution failure probability has been added to some HFEs but not others. HFA_0024RCWINTAKE involves physically cleaning the intake screens within time to prevent a plant trip or equipment overheating. Assuming the execution failure probability is zero is inappropriate. This F&O is assessed as Partially Closed.	HR-G2 remains Met.

Table A-6 BFN Internal Events PRA F&O Closure Review Consensus Table

RU	SR	Prior PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
						relate to a series of manual actions, but to an effort among several operators, so errors of execution are in parallel and considered unlikely. It is assumed that, if the action is initiated within 1 hour, it will be successful (i.e., only the cognitive error is included). The RCW system is supplied river water from the CCW conduits of each unit through fine mesh strainers that include a dP alarm. Pumps are run periodically to avoid fouling. • HFA_0027INTAKE - Basic event is not in the model. NO CHANGE • HFA_0IR2_LPI -Execution errors are included for this event. NO CHANGE. • HFA_0024IFISOL - This event is not used in the PRA model. NO CHANGE.					
HR	HR-F2 HR-G4 HR-G5	HR-F2 is Not Met HR-G4 is Not Met HR-G5 is Not Met	4-25	There are many operator actions that use screening values; see Table 8 of the HRA. None of these actions appear to use any information to base the time available and the times to operator cues and perform the actions are not documented.	Without any real timing information, it is not possible to estimate, even at a screening level, the probability of operator failure or success.	Clarification on the basis for the timing has been added to the HRA Notebook.	NDN-000-999-2007-0032, Rev.6 BFN-0-16-031	Open	none	BFN-0-16-031 list several HFEs with clarification of the timing information. These are not the HFEs listed in Table 8 as referenced in the F&O, nor is there any discussion why these events were selected. NDN-000-999-2007-0032 Assumption 10 assumes that screened HFEs all have a delay time of 24h. This is not consistent with several of the event descriptions, which imply the timing would need to be less than 24h for success (some screened events list times of 15m or less in the description). This F&O is assessed as Open.	HR-F2 remains Not Met (F&O 4-25) HR-G4 remains Not Met (F&O 4-25) HR-G5 remains Not Met (F&O 4-25)
HR	HR-C1	HR-C1 is Met.	4-28	Non-screened miscalibration events are not provided with designators in Appendix A of the HRA. Therefore, HFEs associated with these miscalibration events cannot be readily determined.	The requirements of HR-C1 cannot be verified due to lack of traceability from HRA Appendix A table to the rest of the preinitiator analysis.	A table has been added to Appendix A of the HRA Notebook that list all the pre-initiator CCF HFEs	NDN-000-999-2007-0032, Rev.6 BFN IE PRA Model, Rev 8.	Maintenance	The resolution enhanced the documentation for screened HFEs. (1) No new methods were used. (2) There was no change in PRA scope. (3) There was no change in PRA capability.	HRA Notebook Section 6.2.3 discusses screening values for pre-initiators and references section 6.2.2.3 for detailed analysis. A table has been added to NDN-000-999-2007-0032 Appendix A that specifically lists the pre-initiator CCF events. The detailed results from the HRA Calculator are included as Appendix B. This F&O is assessed as Closed.	HR-C1 remains Met.
HR	HR-A1 HR-A2	HR-A1 is Met. HR-A2 is Met.	4-29	The list of activities reviewed in the HRA Appendix A table is primarily focused on Unit 2 or Unit 0 SRs and SIs. There are a few Unit 1 procedures listed, but it is not clear why certain procedures from Unit 1 are reviewed but not others. More importantly, there do not appear to be any Unit 3	The review of procedures should not be limited to one unit. Differences between units may present additional pre-initiator actions. Although the one example found would not likely result in a pre-initiator, the point is that there	As mentioned on Section 4.3 of the HRA Notebook, the operating practices, staffing and training for all three units are identical. Differences that could relate to the HRA are reflected in the system fault trees. The procedures were reviewed and only one unit was referenced	NDN-000-999-2007-0032, Rev.7	Maintenance	The resolution involves a documentation change only to revise procedure references. (1) No new methods were used.	The HRA Notebook discusses the required procedure reviews in section 4.3. Procedures are reviewed for all the units. This F&O is assessed as Closed.	HR-A1 remains Met. HR-A2 remains Met.

Table A-6 BFN Internal Events PRA F&O Closure Review Consensus Table

RU	SR	Prior PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
				procedures reviewed. A sample review of one procedure between all three units (3.5.1.5(CS I)) found that the Units 1/2 tests affected two relays that are not tested in the Unit 3 procedure.	are differences between the units' procedures.	for each different procedure in the HRA Notebook. This is because each unit has the same steps within the procedure and the only thing that would be different would be specific UNIDs, but the overall result would be the same.			(2) There was no change in PRA scope. (3) There was no change in PRA capability.		
HR	HR-H1	HR-H1 is Met.	6-25	Event HFA_3003P_START_A does not appear to be applied correctly in the model. A question was asked of the analysts on the logic, and the response referred to gate U3_FWH_INIT for events were FW recovery is not credited. However, the logic under gate U3_FWH_G50 limits the operator failure event to only excessive FW events; resulting in no failures coming through for other events were FW is credited.	Significance is unknown, since model modification is required in order to determine the impact.	The modeling for HFA_0003P_START_A was reviewed by plant personnel and it was determined that there are still some issues related to this event that need to be resolved. • It was determined that this event should be set to TRUE due to inadequate time to restart a pump during at ATWS prior to MSIV isolation (After MSIV isolation steam is not available to drive the RFW pump). In addition, the ATWS procedure calls for terminating all injection except for RCIC, CRD and SLC.	BFN IE PRA Model, Rev 8	Maintenance	This item was a modeling fix involving removal of credit for an HFE. (1) No new methods were used. The model was updated using existing processes and tools. (2) The PRA scope was limited to a single event. (3) There was no change in PRA capability.	HFA_0003P_START_A is no longer credited but is still in the model with probability of 1.0. This F&O is assessed as Closed.	HR-H1 remains Met.
HR	HR-G5	HR-G5 is Not Met	6-28	Basis for operator action time (30 min) for HFA_0085ALIGNCST appears to be roughly estimated, as is the time available (7 hours).	Event provides over 5% of CDF.	F&O 6-28 states that for HFA_0085ALIGNCST, rough estimates are used for Tm (30 minutes) and that Tsw=10 hours is not consistent with info from plant personnel. The basis for both of these are shown in the HRA Notebook Revision 5. The basis for the Tm value is given in the Operator Interview Insights section. According to 1-ARP-9-6B an alarm actuates in MCR when the CST reaches the 12-foot level and there is a 10-minute decision time (Tcog) associated with whether to crosstie or not. OSC takes 20 minutes to staff and stage in field. Then the valves that need to be manipulated would take 10 minutes to get to and manipulate (estimate from operator that has recently performed actions). (Tm=Texe=20+10=30 min) The Tsw issue was that we used a different inventory level of 180,000 gallons instead of 135,000 gallons (standpipe) as done for all other MAAP estimates. An assumption is specified in the HRA Notebook	NDN-000-999-2007-0032, Rev.6 0-OI-2B	Maintenance	The change enhanced the justification for HFE timing assumptions and revised a HEP. (1) No new methods were used. The HEP analysis uses the HRA Calculator consistent with accepted guidance. (2) The scope was limited to a specific HFE. (3) There was no change in PRA capability.	Per NDN-000-999-2007-0032, Rev.6, HFA_0085ALIGNCST has been re-evaluated. A basis has been included for Texe of 30 minutes based on operator insights. This F&O is assessed as Closed.	HR-G5 remains Not Met (due to F&O 4-25)

Table A-6 BFN Internal Events PRA F&O Closure Review Consensus Table

RU	SR	Prior PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
						Revision 5, that explains the 180,000 gallons corresponds to the available volume in the standpipe plus a CST height of 15 feet. This is used based on 0-OI-2B which cautions that a CST level below 15 feet may cause a loss of condenser vacuum. See BFN-0-16-031.					
HR	QU-C2 HR-G7	QU-C2 is Not Met HR-G7 is Not Met	6-30	Dependencies between operator actions appear to be non-conservatively applied. Mainly, the Zero Dependence (ZD) between actions is commonly applied, simply when one of the actions takes longer than 60 minutes. What appears to be the mistake is applying the last event tree node in the Dependency Event Tree. In this tree, if the stress of either HFE is moderate or high, the upper leg of the event tree is used. So for combo 2, the HRA assumes ZD, while the event tree would designate Low Dependency.	Systematic error affecting around 1/2 of the combo events, including combo 18.	• The basis for ZD between early depressurization HFA_0001HPRVD1, and failure to align suppression pool cooling is significant differences, cues and timing. Early depressurization is associated with failure to maintain RPV level, while failure to align SPC (non-ATWS/IORV) is associated with SP temperature. MAAP analysis demonstrates that operators have 3 hours to start suppression pool cooling to avoid exceeding 190F and thus eventually impacting HPI systems taking suction from the SP. Since HPCI and RCIC take suction from the CST initially, it would take several hours to deplete the CST prior to any swapping suction to the SP. Early SPC failure was included in the model under late failure for HPI since early failure would result in high SP temperature that may preclude late swap over of suctions for HPI. • The basis for the User Defined dependency levels has been added to the HRA calculation in Appendix E.	NDN-000-999-2007-0032, Rev.6, NDN-000-999-2007-0041, Rev 8	Open	none	The stated resolution addresses only some specific HFEs, however during discussion it was identified that the dependency analyses were completely redone. The actual process used to identify and process dependencies in general is not described, only that the "EPRI recommended" method is used. More detail is needed. HRA NB Section 6.3.3 points to the Quantification and Quantification NB points back to HRA NB. The use of automated tools is mentioned but the actual tools and how they are used is not discussed. There is and assumption (in HRA and Quant) that HFEs with screening HEPs of 0.1 or greater are treated as independent. Discussions with the analyst indicated this is not how they are treated. In the Quantification NB it states that the base quantification use a seed value of 0.15 for all HEPs. In section 6.3.1.9 its states that a sensitivity is performed using 1.0 as the seed value and references the HRA calc. It is not clear how the dependent HFEs are identified. This F&O is assessed as Open.	QU-C2 remains Not Met HR-G7 remains Not Met
IE	IE-A7	IE-A7 is Met.	3-7	Scheduled manual shutdowns (especially for refueling outages) should not be included in the statistical basis for the scram initiator. This can lead to an overly conservative scram initiator frequency. Note that CNRM interpretation for FAQ 06-1060 (should non-forced manual trips which are part of the normal shutdown procedure be counted) states that 'a normal controlled shutdown would not present the same challenges as a trip from full power if the manual	CRNM ASME Standard Interpretation #5 (for FAQ 06-1060) states that normal controlled shutdowns should not be included when counting initiating events. The current practice at Browns Ferry regarding this item, therefore, does not meet the requirements of the standard.	Manual Scrams have been broken up into its own initiator in the MOR R7. Further refinement of the manual scram initiator is currently being investigated and will be included once finished. Since the results are conservative without the refinement, the model will continue to use the unrefined manual scram initiator.	NDN-000-999-2007-0030, Rev 2 (IE.01)	Maintenance	The change involves removal of planned shutdowns (e.g., a refueling outage shutdown with a planned manual SCRAM) from the initiating event frequency. (1) No new methods used. This removal is consistent with current practice	The Initiating Events Notebook, IE.01 was updated to include tabular information for each of the SCRAMs in the defined period. These tables assign an initiator bin to each SCRAM. While a statement that planned shutdowns are excluded is not included, a review of the calculation indicates that the number of events has excluded the planned manual SCRAMs. This F&O is assessed as Closed.	IE-A7 remains Met.

Table A-6 BFN Internal Events PRA F&O Closure Review Consensus Table

RU	SR	Prior PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
				trip was prompted by conditions other than the normal shutdown procedure which could occur at full power, it should be counted.					and uses the existing tools. (2) The scope includes a change in the initiating event frequency with expected impact on the results. Risk insights are not significantly impacted. (3) There is no change to the PRA capability.		
IE	IE-C8	IE-C8 is Not Met	6-10	CCF for Battery Chargers is not included in the Initiating Event Fault Tree for loss of 2 DC buses, other than for the standby chargers (not in the yearly failure rate logic).	Can affect the loss of DC initiating events by a factor of 10, depending on how CCF is calculated.	The IE Notebook lists an Assumption about why inclusion of common cause is not included for support system initiators. Inclusion of common cause into the support system initiator development would produce overly conservative initiator frequencies as mentioned in the previous response. In order to obtain a more realistic model TVA decided to leave out the common cause events for initiator development. Inclusion of the common cause for support system initiator development will be reevaluated and incorporated as required following completion of the evaluation.	IE.01 = NDN-000-999-2007-0030, Rev 2 EPRI TR1016741, Support System Initiating Events	Open	none	An assumption in IE.01 states that inclusion of common cause failures in the initiating event tree would yield inappropriate/conservatively high frequencies. This is counter to current guidance in EPRI TR1016741. An update to IE.01 should be prepared following the EPRI process which allows for appropriate screening of events and other adjustments. This F&O is assessed as Open.	IE-C8 remains Not Met (see F&O 6-10).
IE	IE-C14	IE-C14 is Not Met	6-13	The impact of Surveillance Procedures is not included in the ISLOCA Calculation. For example, for Core Spray, Surveillances in the CS Notebook indicate an MOV opening every 92 days. The likelihood of an ISLOCA during this MOV test is not calculated in the ISLOCA IE Fault Tree, including the sequence where the check valve would have previously failed prior to the surveillance.	Unknown impact on the ISLOCA Frequency, without analyzing the specifics of the site procedure. If the procedure has the operator check downstream pressure (etc.) prior to opening the MOV, likely there is minimal impact. However, given the ISLOCA has a large impact on LERF, the impact could be significant.	There is an open permissive interlock between the inboard and outboard injection valves that allows both valves to be open only when reactor pressure is below the low reactor pressure setpoint. The CS inboard and outboard injection valves have in-line valve interlocks to prevent both valves from being opened with RPV pressure at or above 450 psig. Both receive auto open signals when there is a CS initiation signal and RPV pressure is below 450 psig. The inboard valve may be throttled immediately after initiation. Therefore, failure of the operator to check downstream pressure prior to opening the	IE.01 - NDN-000-999-2007-0030, Rev 2 SY.04 Core Spray System Notebook IE.02 - ISLOCA Initiating Events	Maintenance	The change is documentation only. The additional documentation justifies the continued exclusion of a potential ISLOCA initiator. (1) No new methods were used. (2) There was no change in scope of the PRA. (3) There was no change in the capability of the PRA.	SY.04, Core Spray System Notebook and IE.02, ISLOCA Initiating Events document the interlock between the inboard and outboard isolation valves will not allow the condition of an undetected rupture of the testable check valve and the opening of the MOV because the pressure interlock is set at 450 psig. On this basis, the exclusion of the MOV test is justified. This F&O is assessed as Closed.	IE-C14 is now Met

Table A-6 BFN Internal Events PRA F&O Closure Review Consensus Table

RU	SR	Prior PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
						MOV for testing would not occur due to the low-pressure permissive interlocks.					
IE	IE-A5 IE-C6	IE-A5 meets Cat II. IE-C6 is Not Met.	6-2	Loss of HVAC as an initiating event is screened, based on the 1995 PRA of the event. It appears the model and the assumptions for loss of HVAC have changed, and loss of HVAC as an initiating event should not be screened.	Modeling changes have resulted in HVAC becoming one of the top 5 systems in the present PRA. Based on this, a loss of HVAC initiating event is likely to be significant as a contributor to core damage and should not be screened.	Revision 2 of the Initiating Event Notebook Section 6.2.3.3 now states: "The initial plant faults that result in reactor pressurization are given in Table 10. The reactor pressurization initial plant faults include RCS high pressure, loss of condenser heat sink and turbine trip. The loss of condenser heat sink is made up of Inadvertent Closure of all MSIVs (including from loss of HVAC), Loss of Condenser Vacuum, Turbine Bypass Unavailable, and Loss of Plant Air. None of these SCRAMs are grouped. For the loss of condenser heat sink, the MSIV closure, turbine bypass unavailable, and loss of condenser vacuum have different effects on the pressure spike. The loss of plant control air has a different timing and affects a number of components." Revision 2 of the Initiating Event Notebook Section 6.2.3.8 now states: "The loss of important HVAC systems is well annunciated, and heatup calculations show that there is ample time for the operators to restore HVAC or take procedurally guided steps to prevent unnecessary isolation or SCRAM. Additionally, many of the systems cool areas that do not have high heat loads during normal power operations or do not have equipment necessary for normal operation. For additional discussion see the BFN PRA HVAC system notebook. An exception to this is the CRD system which is only available for 4 hours after a loss of HVAC. The difference in the CRD system is that it does not cause a SCRAM or preclude a SCRAM to occur from another accident signal. The Technical Specifications do not require	IE.01 = NDN-000-999-2007-0030, Rev 2 Initiating Events SY.08, HVAC System Notebook SY.09, Main Steam System Notebook	Maintenance	Additional documentation was provided for the exclusion of the CRD cooling as an initiator, no change in methods was introduced. The documentation was also updated to provide the basis for the inclusion of another initiator, loss of ventilation to the steam tunnel causing inadvertent MSIV closure. The additional changes described in the disposition are not expected to have a significant impact on results.	The HVAC System Notebook SY.08 was updated to identify components requiring HVAC for success, including areas with temperature detectors that if not cooled would cause MSIV closure. The Main Steam System Notebook, SY.09 provides that for temperatures greater than 189F in unspecified areas the MSIVs are failed closed. Based on SY.08, this area is assumed to be the Main Steam Tunnel. The fault tree model includes BEs (e.g., FANFR1FAN_0640135) that cause closure of the MSIVs in the power conversion system tree; however, this BE is not an input to the inadvertent MSIV closure initiator fault tree, rather a point estimate is provided for IMSIV. There is no discussion about timing for area heatup or operator indication for areas with temperature detectors causing MSIV closure. Adding this information could provide a basis that this initiator could potentially be screened. Other initiators, such as loss of power to the shutdown board have a similar affect as the IMSIV imitator and would be caused by some of the same issues causing failure of the fan. Therefore, the addition of the ventilation fan failure to the IMSIV initiator is not expected to cause a significant change in results. The Initiating Events Notebook IE.01 includes a systematic Failure Modes and Effects Analysis (FMEA) of key support systems failures that could cause a reactor SCRAM, including this item. Separately, other information was incorporated into IE.01 to exclude loss of cooling to CRD pumps as an initiator. The basis for this exclusion is provided. This F&O is assessed as Closed.	IE-A5 remains met at Cat II. IE-C6 is now Met.

Table A-6 BFN Internal Events PRA F&O Closure Review Consensus Table

RU	SR	Prior PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
						the CRDs to be functional, just that the control rods be able to insert if the need arises, but it is stated that if both CRD pumps are not available to start, a manual SCRAM would occur. Therefore, HVAC induced loss of CRD is not a direct initiator in the PRA but is modeled implicitly in the manual SCRAM initiator." See BFN-0-16-031.					
IE	IE-C8	IE-C8 remains Not Met	6-36	The ISLOCA Conditional Pipe Break Frequencies calculated for the analysis appear to be too low, in comparison with other pants. From NUREG/CR-5102, Appendix F, Table 2, the RHR and CS piping would generally get a failure probability of 2.65E-02 and 2.54E-03 respectively. Other reference documents used should get similar results. The BFN analysis is supported by and Excel Spreadsheet for the overpressure estimate, and this analysis is not included in the system notebook. In the excel spreadsheet it appears the temperature assumed for the CS and RHR analysis assumes room temperature, whereas full RCS temperature is more appropriate.	ISLOCA is a significant contributor to LERF.	The indicated parameters were updated in MOR 7 to indicate the documented values. The reference links were broken when the document was converted into a PDF for record submittal and this was missed prior to issuance into the vault. The references were fixed for all applicable instances and the specific references for the "tables for section 6" are either labeled directly or taken from references 26 and 27 of the ISLOCA Notebook. Section 6.3.3 specifies that these are the references for the overpressure analysis.	NDN-000-999-2007-0039, Rev 2 (ID.02) BFN IE PRA Model, Rev 8	Maintenance	The change involves both correction of a BE value in the model and a documentation correction. (1) No new methods were used. The updated used existing methods and tools. (2) There was no change in scope of the PRA. No significant impact in results or in risk insights resulted from this level of change. (3) There was no change in the capability of the PRA. The corrected values resulted in the expected change in results with the % contribution for these ISLOCA initiators remaining very low after the change, approximately 0.5% of the total contribution from all initiators.	The model file was confirmed to be consistent with the intended IE.02 documented values for ISLV21 and ISLV23. This corrected the noted deficiency. The pdf "broken link" issues were also corrected in the updated document. This F&O is assessed as Closed.	IE-C8 remains Not Met (see F&O 6-10).
IE	IE-B4 IE-A5 IE-C6	IE-A5 is Met IE-B4 is Met. IE-C6 is Not Met.	6-5	The calculation of HPCI Steam Lines breaks (IE Section 6.2.3.8) does not appear to be reasonable, using older EPRI data and Wash-1400 data. The resulting steam line	Pipe break in the HPCI line can affect RCIC and many other components, due to the HPCI pump being open to other areas. The modeling as documented does	The pipe rupture frequency numbers in Section 6.2.3.8 were updated to the reference the current pipe failure rates for HPCI, RCIC, and RWCU and the	NDN-000-999-2030 Rev 2 (IE.01); EPRI Report 3002000079.	Maintenance	This response applies the most recent data for calculating the subject event	The IE.01 Notebook has been updated by using the latest pipe break frequencies to calculate the HPCI unisolable break frequency. However, there is no discussion of why the specific values were selected (For example, the HPCI line break uses the table for NPS >10 and EBS of 10 in). The calculation is also using updated failure rates	IE-A5 remains met at Cat II. IE-B4 remains Met. IE-C6 is now Met.

Table A-6 BFN Internal Events PRA F&O Closure Review Consensus Table

RU	SR	Prior PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
				break calculated is 4.55E-10/year, which does not compare with results from other plants. Using newer data, the pipe break frequencies would likely be 2-orders of magnitude higher. Additionally, although the isolation valves may be available to eventually isolate the break, the impact of the break may have already occurred prior to isolation. Also, the generic MOV FTC value (from NUREG/CR-6928) in Data Table 4 is 1.07E-03/demand. Finally, the CCF probability used should be changed to the HPCI MOV FTC, with Alpha = 1.41E-02.	not provide basis for screening, and if re-performed, the analysis will likely result in orders of magnitude increases here.	rationale shown below (in bold) was added to the last paragraph to explain why each break outside containment was screened. The 2009 Standard describes a significant cutset as the summed percentage is 95% and the individual percentage is 1% of the applicable hazard group . The calculated probabilities of each of the lines shown above for an individual cutset is less than 1% of hazard group contribution. Summing the contribution for RWCU, HPCI and RCIC (~1.5E-8) the sum is less than 95% of the hazard group. Therefore, each of these is considered to be an insignificant contributor to CDF and LERF, so the HPCI, RCIC, and RWCU BOC initiators are not included in the BFN model. In addition, if the HPCI line fails then RCIC would provide a backup source of inventory into the reactor along with LPCI once the operators depressurize. If the RCIC line fails, then HPCI would provide a backup source of inventory into the reactor along with LPCI once the operators depressurize. If RWCU fails, then HPCI and RCIC can provide inventory into the reactor. See BFN-0-16-031.			frequencies. (1) No new methods were used. Only data was updated. (2) There was no change in scope of the PRA. (3) There was no change in the capability of the PRA.	and CCF factors. This updated frequencies for the subject initiators are below the IE-C6 screening criteria and additional discussion is provided to demonstrate that at least two trains of mitigating systems remain available for the events being screened. As an example, for unisolable HPCI line breaks, RCIC and LPCI are stated as being available. This discussion does not provide supporting detail that the nearby RCIC pump would not be affected by the steam and water released from the HPCI line break. This F&O is assessed as Closed.	
IE	IE-C8	IE-C8 is Not Met	6-8	RCW initiating event appears to be incorrectly reduced by factor RCWMTCF for combinations where the reduction factor does not appear to be valid. In particular, the event is applied to cutsets containing common transformer events. Also, reduction factor appears to be calculated incorrectly (1/365)**2.	Loss of RCW initiating event appears to be reduced by a factor of 1E-02 from the actual.	This was a legacy item that was never removed but should have been. This has been confirmed to have been removed from the latest MOR.	IE.01 = NDN-000-999-2007-0030, Rev 2, Initiating Events Notebook Recovery Rule file = BFN123_MREC_r8.RECV	Maintenance	This issue was addressed by removing incorrect factor and updating documentation to reflect this. No new methodology was involved in making the change. (1) No new methods were used. The updated used existing methods and tools. (2) There was no change in scope of the PRA. The	Review of the BFN recovery rule file confirms that the invalid correction factor has been removed. This F&O is assessed as Closed	IE-C8 remains Not Met (see F&O 6-10).

Table A-6 BFN Internal Events PRA F&O Closure Review Consensus Table

RU	SR	Prior PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
									percent contribution of the loss of Raw Water initiator to all initiators is very low (0.2% for MOR Rev0 and currently below truncation). (3) There was no change in the capability of the PRA. This accident sequence and risk insights are not significantly changed.		
LE	LE-F2	SR LE-F2 is Not Met	1-33	There is no discussion of the review of the LERF contributors (ASME/ANS RA-Sa2009 Table 2-2.8-9) for reasonableness per the review of the QU Notebook and LE.01.	A review of the reasonableness of the results of the analysis of the contributors to LERF is required per the SR.	The review of the CDF and LERF cutsets was performed and documented in Attachment D and E of the Quantification Notebook. Section 6.3.2.3 of the Quantification Notebook specifies the types of things that were looked at when reviewing the cutsets. The Top 100 cutsets, a sample of 100 cutsets from the middle and the last 100 cutsets were all reviewed and showed no signs of inconsistencies in logic.	LE.01 - LERF Main Report, Rev 5; NDN-000-999-2007-0041; NDN-000-000-2010-0001 Rev, 008	Open	none	The current documentation provides a listing of addressed phenomena and failures postulated to lead to LERF in Table A.1-2. How the BFN model maps to these postulated events is provided in Table 11. The model mapping is again provided in the QU notebook in Table 6.3–11. The frequency results are tabular in the QU notebook and there is a comparison of absolute frequency to similar designs. However, there is no documented review of the results to determine if the LERF results are reasonable and that the identified contributors (categories) are consistent with expectations. A pointer to the summary document was provided but the requested information was not found at that location. This F&O is assessed as Open.	SR LE-F2 was previously Not Met and remains Not Met.
LE	LE-D1	LE-D1 is met	2-35	The containment structural analysis does not address the Unit 3 primary containment ultimate capacity in section 6.3.	All three unit containments must be addressed	The Containment Ultimate Capacity that is currently addressed in the LERF Notebook is applicable for all 3 units. There were no identified differences between the three units with respect to containment parameters, so none were specified in the LERF Notebook.	LE.01 - LERF Main Report, Rev 5; NDN-000-999-2007-0038	Maintenance	No change required since the documentation can be obtained in the current references. The change is documentation only. (1) No new methods were used. (2) There was no change in scope of the PRA. (3) There was no change in the capability of the PRA.	Supporting documentation for the LERF report identifies in Section 6.1.2 Primary and Secondary Containment Building Walk-Through that "Based on a review of these photographs it is apparent that the BFN Unit 1 Primary Containment is sufficiently similar to BFN Units 2 & 3 that a physical walkdown is not warranted." The report also states in section 6.3 that "The Browns Ferry Unit 1 containment structural is identical in construction to the Unit 2 containment." Therefore, the existing information seems appropriate and documented. This F&O is assessed as Closed.	LE-D1 was previously met at Cat II and remains Met at Cat II.
LE	LE-C11 LE-C12	LE-C11 meets Cat I LE-C12 meets Cat I	4-48	No credit is taken for equipment survivability or human actions following containment failure.	LE-C11 implies credit be taken for equipment survivability following containment failure, for Cat II/III.	The LERF Notebook Attachment A discusses the phenomenological conditions that go into CZ1. Figure A.6-2 and Table A.6-2 show the fault	LE.01 - LERF Main Report, Rev 5; NDN-000-075-2007-0010, Rev, 005	Maintenance	The resolution involved correction of existing information and was documentation	The LERF model discusses SA/EQ in section 3.1.3 Equipment Survivability in a Severe Accident Environment. The existing model does incorporate in the system modeling response options designed to improve performance. These are documented in the system modeled development. For example, the CS system can be aligned to the CST to allow continued injection (NDN-000-075-2007-0010 Rev: 005, 3.2.3 Alternate System Alignments).	LE-C11 was previously met at Cat I and is now met at Cat I/II, LE-C12 was previously met at Cat

Table A-6 BFN Internal Events PRA F&O Closure Review Consensus Table

RU	SR	Prior PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
						tree modeling and split fraction designations of CZ1.			only. (1) No new methods were used. (2) There was no change in scope of the PRA. (3) There was no change in the capability of the PRA.	This F&O is assessed as Closed.	I and is now met at Cat II
LE	LE-C10	LE-C10 meets Cat I	4-50	Although equipment survivability beyond equipment qualification limits is credited, there is no indication that significant accident progression sequences were reviewed to determine if continued equipment operation could be credited to REDUCE LERF.	LE-C10 Cat II/III requirements are to REVIEW significant sequences to determine if engineering analyses can be used to take credit for additional equipment operation beyond normal qualification limits to reduce LERF.	The significant accident sequences were reviewed as described in the QU Notebook Section 6.3.2.3. Equipment survivability was looked at and used where achievable as the discussion in the ISLOCA Notebook Section 6.3.4.5 attests. The two cutsets that were brought up were not refined for equipment reliability as they would both involve addition of HRA events while we are already at the floor value with respect to HRA. Addition of the HRA events would increase the number of combination events which increases the number of HRA recovery rules, and that would in turn increase the time to quantify. There would not be much gain (if any) of LERF either, as the dependency analysis would limit the amount of credit the human action would give.	LE.01 – LERF Main Report, Rev 5; NDN-000-075-2007-0010, Rev: 005	Maintenance	The resolution involved correction of existing information and is documentation only. (1) No new methods were used. (2) There was no change in scope of the PRA. (3) There was no change in the capability of the PRA.	The LERF model discusses SA/EQ in section 3.1.3 Equipment Survivability in a Severe Accident Environment. The existing model does incorporate in the system modeling response options designed to improve performance. These are documented in the system modeled development. For example, the CS system can be aligned to the CST to allow continued injection (NDN-000-075-2007-0010 Rev: 005, 3.2.3 Alternate System Alignments). This F&O is assessed as Closed.	LE-C10 was previously met at Cat I and is now met at Cat II.
LE	LE-C1 LE-C8	LE-C1 meets Cat-I LE-C8 is Met.	4-51	Class3A (B,C)-006 LERF sequences are non-sensical. In these sequences, TD2 succeeds (i.e., DW Spray hardware is available and operator initiates injection per Table A.5.7-1) but DWS fails later in the CET (DWS_ALL_SUP branch is questioned).	Class3A(B,C)-006 LERF sequences are non-sensical. In these sequences, TD2 succeeds (i.e., DW Spray hardware is available and operator initiates injection per Table A.5.7-1) but DWS fails later in the CET (DWS_ALL_SUP branch is questioned).	The LERF Notebook Attachment A Section A.7.6.1 under the heading "Nodes Affected by Success/Failure of (FD/FC)" states: The upward path will be used to represent flooding of containment resulting in a release through the drywell vent. This will not be a contributor to LERF if coolant injection is available to the debris, i.e., TD/TR = S. This question is treated in the RME/RBE node. From the RME/RBE node section Table A.10-3, RME6 results in LERF due to the DW vent and the operator action failure to initiate DWS, while RME4 does not contribute to LERF because	LE.01 - LERF Main Report, Rev 5; CET1_r8.eta; BFN IE PRA Model. Rev 8	Maintenance	The resolution was to explain the modeling using the existing information. The change is documentation only. (1) No new methods were used. (2) There was a minor change in scope of the PRA by the addition of the events. (3) There was no change in the capability of the PRA.	The model was reviewed to the original F&O related to Top TD and the resolution put forth by TVA was found to be accurate and the F&O is closed. A second F&O from the 2015 review as also examined related to containment flooding related to the ability to have a flooded drywell and still have an impaired containment (Top RME). CET top RME addresses the effectiveness of the reactor building. For the case in question the flooding in the drywell is assumed to lift dampers and result in bypass of the drywell leading to an early release path. The assessment is conservative, and the model is consistent. No modeling changes are necessary. This F&O is assessed as Closed.	LE-C1 remains Met at Cat-II. LE-C8 remains Met.

Table A-6 BFN Internal Events PRA F&O Closure Review Consensus Table

RU	SR	Prior PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
						of the delayed containment failure.					
LE	LE-G2	LE-G2 is Met.	4-54	The method used to quantify split fractions was very difficult to review and appears to be based on an old LERF model that is not consistent with the current Level 1 model. The split fraction fault trees were not provided. Further, many of the split fraction descriptions provided in Appendix A of LE.01 do not appear to be current or are no longer used in the LERF model.	Split fraction values could not be determined by the reviewer, and descriptions for many split fractions do not appear to be valid anymore.	As mentioned in the LERF Notebook Attachment A Section A.6.4, the detailed phenomenology fault tree was developed and quantified using RISKMAN and has not changed for the current PRA. The nodes are calculated using the old RISKMAN Program and are shown in Table A.6-4. The phenomenological basic events do not need to have any importances assessed as they involve no equipment failures and the split fractions are generated using current industry practices.	LE.01 - LERF Main Report, Rev 5; Browns Ferry Nuclear -8/25/2009- RG 1.200 PRA Peer Review; P0132150002-5175	Maintenance	The update added fault tree models that are now a part of the base CAFTA model and represented a model change. Since the same data is utilized and the model is only used to represent the same logic there is no change in method. Use of the same model data and same model structure should result in the same results such that there is no chance in capacity. Overall the change should not significantly alter results. (1) Putting the logic in a small fault tree is a minor change in methods. (2) No new events were added but the small CAFTA models were added and represent a minor scope change for the PRA. (3) There was no change in the capability of the PRA.	A review of the documentation identified simplified fault trees that represent the logic for the CET tops. Tables in Appendix A provide the status for success and failure, failure probability and basis. The current CAFTA model also has the fault trees for the tops. The CET endstates are quantified through the combination of sequence events (example, U1_CET1_003P). This allows for CD insights to be propagated appropriately to the CET. This F&O is assessed as Closed.	LE-G2 remains Met.
LE	LE-B1	LE-B1 is Not Met	7-6	Section 7.1 of LE.01 directly addresses those contributors from the table, but plant specific issues do not appear to be addressed.	The SR requires the consideration of unique plant issues.	The LERF Notebook Section 7.1 was revised to indicate that MSBOC and FWBOC are both LERF contributors. The common cause failure of the battery was also included as a plant specific contributor to LERF.	LE.01 - LERF Main Report, Rev 5	Maintenance	The resolution is a change to documentation. (1) No new methods were used. (2) There was no change in scope of the PRA. (3) There was no change in the capability of the PRA.	The LERF contributors are identified in the text defined in Section 7.1. This F&O is assessed as Closed.	LE-B1 was previously Not Met and is now Met at Cat II.

Table A-6 BFN Internal Events PRA F&O Closure Review Consensus Table

RU	SR	Prior PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
LE	LE-B1	LE-B1 is Not Met	7-7	The definition of Early appears to be inconsistent and may eliminate some scenarios from consideration for LERF.	Definition of the timing of accident sequences determines whether a sequence can contribute to LERF. Timing based from accident initiation will be different than timing from declaration of General Emergency.	The referenced EALs do specify a GE at containment pressure of 55 psig, however the referenced EALs also specify a GE at a reactor water level not being maintained (i.e., low). During a loss of decay heat removal, the water level would decrease and HCTL limits would be exceeded which would signify impending loss of fission product barriers which would lead to the declaration of the General Emergency. It is TVA's belief that the LERF Notebook adequately states the definition of LERF based on the above discussion.	LE.01 - LERF Main Report, Rev 6; NDN-000-999-2008-0006, Rev 006	Maintenance	This resolution is limited to enhancing documentation to provide a more detailed discussion of the quantification results. (1) No new methods were used. (2) There was no change in PRA scope. (3) There was no change in PRA capability.	The documentation was reviewed to determine the basis for LERF timing. The development in the SC notebook is well documented and identifies a period of 6 hours for the breakpoint between LERF and non-LERF cases. Representative values have been between 4 and 8 hours so this is considered reasonable for the BFN site and population density. The time is based on the evacuation of the population such that any releases would not involve exposed population. Therefore the 6 hours is a fixed time that is tied to the initiation of a general emergency (GE) such that evacuation is initiated. The timing for the GE is based on plant-specific MAAP assessments and is again, an appropriate metric. This F&O is assessed as Closed.	LE-B1 was previously Not Met and is now met at Cat II
QU	QU-D2	QU-D2 is not Met.	1-34	Additional attention should be applied to significant cutsets to determine that the bases for the cutsets are consistent with modeling and operating philosophies.	The top accident sequence cutset for both CDF and LERF deals with clogging of the intake and includes events that are very uncertain. The attention given this cutset to minimize the uncertainty associated with the contributing basic events has not been sufficient. The approach to dealing with such important cutsets should assure that the contributors are understood and are supported by appropriate rigorous analyses and/or assessment.	The review of the CDF and LERF cutsets was performed and documented in Attachment D and E of the Quantification Notebook. Section 6.3.2.3 of the Quantification Notebook specifies the types of things that were looked at when reviewing the cutsets. The Top 100 cutsets, a sample of 100 cutsets from the middle and the last 100 cutsets were all reviewed and showed no signs of inconsistencies in logic. In addition, the top accident sequences were also reviewed as documented in Section 6.3.2.2 of the QU Notebook. Each of these were reviewed to determine whether they were appropriate. In regards to the questions asked, the mechanisms by which 480V AC bus failures become initiating events would be documented in the system notebooks or initiating event notebooks, the basis for the 2 CRD pump requirement is in the Success Criteria Notebook based on the MAAP runs in the Thermal Hydraulics calculation, the conservatisms related to modeling transients with stuck open MSIVs directly as LERF events would be in the Accident Sequence Notebook.	NDN-000-999-2007-0041, Rev 8	Maintenance	This resolution is limited to enhancing documentation to provide a more detailed discussion of the quantification results. (1) No new methods were used. (2) There was no change in PRA scope. (3) There was no change in PRA capability.	The quantification results are reviewed. The review identifies the most significant contributors. Section 6.3.2.3 includes an assessment of modeling consistency. QU-D2 is now Met. This F&O is assessed as Closed.	QU-D2 is now Met.

Table A-6 BFN Internal Events PRA F&O Closure Review Consensus Table

RU	SR	Prior PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
QU	HR-I3 IE-D3 LE-F3 SY-C3 SC-C3 QU-E2 QU-E4 QU-F4 QU-E1 DA-E3	HR-I3 is not Met. IE-D3 is not Met. LE-F3 is not Met. SY-C3 is not Met. SC-C3 is not Met. QU-E1 is not Met. QU-E2 is not Met. QU-E4 is not Met. QU-F4 is not Met. DA-E3 is not Met.	3-10	Modeling uncertainty comes from two general types of issues, plant specific and generic. Plant specific uncertainties and assumptions should be identified and documented during the model development. The generic sources of uncertainty are listed in EPRI Report 1016737 Table A-1. Both types of uncertainties must be addressed for the base model. Examples of plant specific uncertainties include: (1) ISLOCA valve failing to close after testing is not listed in the sources of uncertainty, nor is the conditional probability that the break is greater than 93 or 600 gpm. (2) For Initiating Events, the factors affecting INTAKE initiating event is not included in the assumptions section, nor are any of the other assumptions in the analysis. (3) Specific assumptions for the detailed HFEs is not discussed, including assumptions made for timing of operator responses (versus analyzed or those observed on a simulator)	Sources of uncertainty must be identified and documented.	EPRI Report 1016737 gives guidance on how to perform an uncertainty analysis. The report goes into parameter uncertainty, model uncertainty and completion uncertainty. BFN takes into account parameter uncertainty by having an uncertainty value tied to each basic event and initiating event. For those basic events that were updated using plant specific data, WinBUGs was run to see how well the data fit the Poisson Distribution and this is documented in the Data Notebook. A probability density function was also created for all of the basic events and initiating events that have been updated with plant specific data and this was used to see how well the posterior data compares to the prior data. BFN describes the model uncertainties in Attachment H of the Quantification Notebook. All the model uncertainties that were identified are within this attachment with a reference as to what notebook the uncertainty is taken from. Any uncertainty related to completeness is specified in the specific notebook in which the completeness is not taken into account (i.e., the system notebooks has the excluded components section that list why components are not taken into account and any assumptions made from that are listed in the assumptions section). See BFN-0-16-031.	NDN-000-999-2007-0041, Rev 8, BFN-0-16-031	Maintenance	The is primarily a documentation update to provide enhanced discussion of uncertainties. (1) No new methods were used. (2) There was no change in scope of the PRA. (3) There was no change in the capability of the PRA.	Attachment H to the Quantification Notebook includes a discussion of assumptions and modeling uncertainties and includes a table of the various items identified in the BFN PRA. The table includes a description and basis to characterization the items and potential impacts to the results. This F&O is assessed as Closed.	HR-I3 is now Met. IE-D3 is now Met. LE-F3 is now Met. SY-C3 is now Met. SC-C3 is now Met. QU-E1 is now Met. QU-E2 is now Met. QU-E4 is now Met. QU-F4 is now Met. DA-E3 is now Met.
QU	QU-F2 QU-F1	QU-F1 is not Met. QU-F2 is not Met.	3-28	A detailed discussion of the quantification asymmetries (with respect to different units, system alignments, etc.) is not presented.	This is an important part of the quantification documentation process.	The QU Notebook Section 4.3 was expanded to include specific differences that impact the results between units. The numbering was also revised in the QU Notebook.	NDN-000-999-2007-0041, Rev 9, SY.08 - HVAC Rev 5	Maintenance	This change is limited to documentation updates. (1) No new methods were used. (2) There was no change in scope of the PRA. (3) The was no	Discussion of modeling asymmetries and the impact on results between the units is provided in Section 4.3 of the Quantification Notebook. This F&O is assessed as Closed.	QU-F1 is now Met. QU-F2 is now Met.

Table A-6 BFN Internal Events PRA F&O Closure Review Consensus Table

RU	SR	Prior PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
									change in the PRA capability.		
QU	QU-F3 LE-G6 QU-D6 QU-F6	LE-G6 is not Met. QU-F6 is not Met. QU-D6 is Met QU-F3 is Met	3-31	The definitions for significant when presenting lists of important equipment, operator actions, etc. do not always conform to the strict ASME standard definition of significant. Justifications for the alternatives used are not presented.	This issue causes the supporting requirement QU-F6 not to be met.	The QU Notebook Section 6.3.2.2 was revised to indicate significant accident sequences were those that contributed at least 1% to CDF or at least 1% to LERF. The definition of significant accident sequences now appropriately reflects that of the ASME standard.	NDN-000-999-2007-0041, Rev 8	Maintenance	The change is documentation only. The definition of significant accident sequences used is documented and the results are presented consistently. (1) No new methods were used. (2) There was no change in scope of the PRA. (3) There was no change in the capability of the PRA.	The Quantification Notebook documents the definition of significant that is used consistent with the ASME Standard. This F&O is assessed as Closed.	LE-G6 is now Met. QU-F6 is now Met. QU-D6 was previously Met at CC-II/III and continues to be Met at CC-II/III. QU-F3 was previously Met at CC-II/III and continues to be Met at CC-II/III.
QU	QU-C2 HR-I3 HR-G7	QU-C2 is Not Met. (HR-I3 is Not Met. HR-G7 is Not Met.	4-21	The joint HEP for several combined operator actions are too low and cannot be justified. Specifically, three combined actions have joint HEPs of less than 1E-7, and eight are less than 1E-6. Note that the HRA acknowledges these low combined HEPs but does not enforce any lower bound. Further, it states that a sensitivity will be performed in the Quantification Notebook, but none is performed.	If the joint HEP for combined events is too low, sequence and overall results may be artificially lowered, and the importance of the operator actions may be understated.	Basis for JHEP floor value (The floor value applied in the dependency analysis lacks a justification for divergence from industry standards and it has a significant impact on BFN results. In addition, the automated HRAC dependency process does not account for intervening successes in the accident sequences, which is an element of this SR). • The HRA floor value recommended by is 1E-5. However, this arbitrary value tends to skew the PRA results. The HRA industry group has been working on developing guidance for the minimum Joint Human Error Probability to be used in PRA, but this guidance is not available at this time. However, as recommended by EPRI HRAUG, sensitivity #3 is included in the Quantification Notebook to determine the impact of the selected floor value.	NDN-000-999-2007-0041, Rev 8, NDN-000-999-2007-0032, Rev 5, U1_CDF-1E-12.CUT	Maintenance	The change involves adding a floor value for HEP credit. (1) No new methods used. The method uses fault tree modeling and recovery rules. It is consistent with current practice and uses the existing tools. (2) The scope includes all HFE combinations with HEPs less that 1e-5 or 1e-6. The PRA results are impacted as demonstrated with a sensitivity study showing an expected increase in CDF. Risk insights are not significantly impacted. (3) There is no change to the PRA capability.	A floor value has been implemented to limit HEP credit in the BFN PRA as described in Section 4.1 of the HRA Notebook and Section 6.3.3 of the Quantification Notebook Sensitivity Case 3. The basis of the floor values is provided and the values are consistent with other plants. This F&O is assessed as Closed.	QU-C2 remains Not Met. (see F&O 6-30) HR-I3 is now Met. HR-G7 remains Not Met. (see F&O 6-30)

Table A-6 BFN Internal Events PRA F&O Closure Review Consensus Table

RU	SR	Prior PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
QU	QU-D2 QU-F2 QU-D7	QU-D2 is Not Met. QU-D7 is Not Met. QU-F2 is Met.	4-36	The assumption that A HVAC is normally running and B HVAC is in standby leads to skewed basic event importance's and non-sensical cutsets. For example, with A HVAC always running: (1) The Loss of RMOV Board A importance is much higher than RMOV Board B (10% vs. 2.5%) (2) Non-sensical cutsets exist, such as where RMOV Board A is in maintenance and B HVAC fails to start (due to operator or hardware failure).	The assumption that one train is always normally running (the HVAC is only an example) does not reflect the plant operation and can result in skewed importance results or missing cutsets/sequences (i.e., how would the results be different if the other train were assumed to be running?).	The run and standby flags have been reviewed and the model is reflective of the normal operating configuration. A sensitivity was run on the PCS pump configuration which is documented in the QU Notebook. This sensitivity showed no change to CDF/LERF for any unit.	NDN-000-999-2007-0041, Rev 8, SY.08 - HVAC, Rev 5, BFN IE PRA Model Rev 8	Maintenance	The resolution involves a probability value update to the PRA model. (1) No new methods used. The change involves fault tree and data updates consistent with current practice and uses the existing tools. (2) The scope includes multiple equipment alignment flags/logic. The PRA results now distribute the risk appropriately between trains. Risk insights are not significantly impacted. (3) There is no change to the PRA capability.	Based on inspection of the PRA fault tree and flag file, the normal/stby flags for HVAC have been replaced with appropriate split fractions. (Note, this is contrary to the stated resolution, which indicates the change was not made. Also, the related discussion in the SY Notebook is incorrect-see F&O 3-28) This F&O is assessed as Closed.	QU-D2 is now Met. QU-D7 is now Met. QU-F2 was previously Met and continues to be Met.
QU	HR-H3 QU-D5	QU-D5 is not Met. HR-H3 is not Met.	4-40	A review of non-significant cutsets found many LOOP cutsets that have combinations of two independent HFEs which should have some level of dependency: HFA_02114KVCRSTIE (Failure to cross-tie 4kV SD Board) AND HFA_0231480SDBTIE (Failure to provide alternate power to 480V SD Board).	This is an example of non-significant cutsets that, had they been reviewed, would have uncovered the need to perform additional operator dependency analyses.	Revision 7 of the Quantification Notebook Section 6.3.2.10 now states: "All accident sequences in each event tree model were quantified and reviewed early in the quantification process to check, debug, and finalize the model. This was found to be helpful in identifying modeling enhancements and to ensure that the event tree modeling logic is correct. The bottom 100 CDF and LERF cutsets for each unit are provided in Attachment D and Attachment E, respectively. These cutsets were reviewed in the same manner as the significant cutsets were reviewed, and there were no identified inconsistencies in the logic." See BFN-0-16-031.	NDN-000-999-2007-0041, Rev 8, NDN-000-999-2007-0032, Rev 5, U1_CDF-1E-12.CUT	Maintenance	The changes for this F&O are primarily documentation. The cutset reviews are performed and documented. The actual HRA dependency analysis is addressed by F&O 6-30. (1) No new methods were used. (2) There was no change in scope of the PRA. (3) There was no change in the capability of the PRA.	The HFE dependency analysis was re-performed using the HRA Calculator. The non-significant cutsets are reviewed as discussed in Section 6.3.2.10 of the Quantification Notebook. The cutsets were reviewed and no independent HFEs were identified as described in the F&O. This F&O is assessed as Closed.	QU-D5 is now Met. HR-H3 was previously Not Met and is now Met.
QU	QU-D3	QU-D3 is Met	4-41	Offsite power recovery is applied in cutsets where it might not be possible. See U1 CDF cutset at 1.151E-08: LOOP with common	Recoveries should only be applied to scenarios or cutsets where the	The offsite power recoveries are applied to cutsets that involve a loss of offsite power event or a loss of a diesel generator.	BFN IE PRA Model Rev 8, U1_CDF-1E-12.CUT,	Maintenance	Resolution of this F&O was based on justification of the existing PRA	Discussions with BFN staff demonstrated that offsite power could still be applied for the cutsets described in the F&O. The breaker failures block use of the DGs but still allow off-site power to be restored.	QU-D3 was previously Met and Continues to be Met.

Table A-6 BFN Internal Events PRA F&O Closure Review Consensus Table

RU	SR	Prior PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
				cause failure of shutdown board normal feeder breakers to open.	recovery can be expected to be successful	Therefore, recovering power to the shutdown board would still be a viable pathway as at least one shutdown board would still be available. In the LOOP recovery rules, there are some instances where battery depletion or HVAC might be lost along with a LOOP, but this is still a recoverable event as the operator would still have at least one shutdown board available to recover power to. In addition, a review of the CDF and LERF cutsets was performed and documented in Attachment D and E of the Quantification Notebook. The Top 100 cutsets, a sample of 100 cutsets from the middle and the last 100 cutsets were all reviewed and there were no identified instances where recoveries were applied to non-recoverable failures.	BFN-0-15E500-1-CC_004337876, Rev 44		model. (1) No new methods were used. (2) There was no change in scope of the PRA. (3) There was no change in the capability of the PRA.	This F&O is assessed as Closed.	
SC	SC-B1 SY-B6 SY-B7	SC-B1 is Met SY-B6 is not Met. SY-B7 is Met	1-12	Several examples found for lack of engineering analyses regarding HVAC that could be justified by calcs. Condensate System Notebook (SY.01) assumes active ventilation is not required due to plant experience Core Spray System Notebook (SY.04) assumes keepfill system is not required. HPCI System Notebook (SY.07) assumes dependence on quad cooling for the remaining 20 hours of post-accident operation.	The SR expects that engineering analyses will be performed to determine whether these statements are correct.	The keep-fill system is not modeled as a CS support system. Failures of this system would be detected and corrected during normal operation. Daily instrument checks are performed through procedure x-SR-2, attachment 2. (where x is the unit)	NDN-000-999-2007 (SC.01); SY.01 Condensate System Notebook; SY-04 - Core Spray; SY-07 HPCI	Maintenance	The change is a documentation update only. (1) No new methods were used. (2) There was no change in scope of the PRA. (3) There was no change in the capability of the PRA.	Most of the success criteria are based on realistic (MAAP) analysis. Some of the supporting analysis for systems are based on generic or conservative assumptions for room heatup; however, these assumptions do not affect the determination of which systems are required to respond to imitating events. Additional documentation was included in the system notebooks providing the basis for the room heatup analysis. Where available, calculations were used for the room heatup. This F&O is assessed as Closed.	SC-B1 remains met at Cat II. SY-B6 is now Met. SY-B7 remains met at Cat II.
SC	SC-A5	SC-A5 is not met	3-12	There is no evidence of an analysis for sequences that go beyond the 24-hour period to evaluate the appropriate treatment relative to the CC II/III requirements for SC-A5.	A CC II/III for SC-A5 requires that options other than assuming sequences in which a stable state has not been reached in 24 hours goes to core damage.	Basis for "Safe and Stable" for HFA_0085ALIGNCST - During a single unit accident, refill of the CST inventory is credited in the model (HFA_0085ALIGNCST) by refilling from the non-accident unit's CST. During a multi-unit accident, it is assumed that the TSC would direct the operators to provide additional inventory to the CSTs from an outside source given the CST depletion would not occur for 10 hours. This assumption is not documented in the current model. • It is already considered in the	SC.1 (NDN-000-999-2007-0035 R3) Success Criteria	Open	none	Additional discussion of the bases for "safe and stable" has been added. However, there is no discussion whether any sequences were identified that require a mission time beyond 24 hours to reach safe and stable. Note that Table 6-1 of SC.1 contains several statements implies that sequences may not safe be and stable at 24 hours and a bounding PDS may be assigned. This instruction in Table 6-1 is consistent with SC-A5 Cat I. This F&O is assessed as Open.	SC-A5 remains met at Cat I.

Table A-6 BFN Internal Events PRA F&O Closure Review Consensus Table

RU	SR	Prior PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
						cognitive analysis for HFA_0085ALIGNCST and the assumption that the TSC would direct operators to provide additional inventory to the CSTs is documented in the HRA Notebook. The alarm response procedures 1(2,3)ARP-9-6B provides a list of alternative sources including: 1) Hotwell or Radwaste transfer to CST, 2) Demin or another CST transfer to the affected CST, and 3) CST Crosstie. The TSC and OSC would determine and perform the appropriate actions based on conditions at the plant and the choices identified in ARP.					
SY	SY-A3	SY-A3 is Met.	2-23	In section 3.2.6.1 of the HVAC system Notebook, it states that the running ACU for unit 3 electric boards must be tripped before the standby unit can be started. Failure of this trip to occur is not reflected in the fault tree.	A breaker failing to provide tripped indication for a start permissive can happen and this failure mode should be included.	Failure of the operating unit to trip has been added to the model as a failure mode of the standby unit.	BFN IE PRA Model. Rev 8	Maintenance	No change is required to close this F&O. (1) No new methods were used. (2) There was no change in scope of the PRA. (3) There was no change in the capability of the PRA.	The standby ACU will not be required until the normal operating unit fails. This will, in itself, meet the trip condition based on discussions with knowledgeable TVA staff. Therefore, no change is required to address this F&O. The model was examined (gate U1_U1_HVAC_EBR and U1_U1_HVEBR_G005) and found be appropriate. This F&O is assessed as Closed.	SY-A3 remains Met.
SY	SY-A5 SY-A13	SY-A5 is Met. SY-A13 is Met.	2-31	For SPC and LPCI, the LPCI injection valves and SPC return valves are required to reposition when swapping RHR modes, but this is not included in the model. The RHR system Notebook indicates that these valves need to close for the opposite function. However, in one location in the notebook it is indicated that flow can be split between LPCI and SPC.	All active components should be included in the failure modes of a system.	The injection valves do need to change position for split LPCI/SPC flow; two valves would have to fail to modulate or close in either path to fail either system. An operator interview was conducted to address this issue. The common cause failure probability of two MOV's to close is less than 1E-5. The RHR pump start failure probability is approximately 1.4E-3. The failure of two MOV's to close is less than 2 orders of magnitude lower than another failure that would fail the system in a similar manner. Therefore, failure to close (or modulate) either the LPCI or SPC injection path can be neglected. The RHR System Notebook was modified to reflect this and the operator interview was added.	NDN-000-074-2007-0025, Rev 005; BFN IE PRA Model, Rev 8	Open	none	The model includes other valve realignments and common cause. It is unclear why this specific change would warrant a unique modeling approach. The absence of this failure mode could alter the importance calculations for the identified components and impact the ability to determine MSPI characteristics. It would be expected that these valves would need to be included since it does involve a physical change in state. This F&O is assessed as Open.	SY-A5 remains Met. SY-A13 remains Met.

Table A-6 BFN Internal Events PRA F&O Closure Review Consensus Table

RU	SR	Prior PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
SY	SY-A8 SY-B9	SY-A8 is Met. SY-B9 is Met.	4-32	Several electrical system boards are modeled to receive power from multiple sources (e.g., normal and alternate buses, and/or EDGs) without considering the need for undervoltage detection and operation circuitry for breakers and EDGs.	Component boundaries for breakers do not include such circuitry, based on NUREG/CR-6928. Note that local circuitry and protection devices are included.	The EDG logic to start and load (close output breaker) are currently modeled. The component description for the circuit breaker component in Appendix A of NUREG/CR-6928 states: The circuit breaker (CBK) is defined as the breaker itself and local instrumentation and control circuitry. External equipment used to monitor under voltage, ground faults, differential faults, and other protection schemes for individual breakers are considered part of the breaker. External equipment used to monitor under voltage is considered part of the breaker. The modeling of automatic bus transfer in the BFN model contains both the normal supply breaker failure to open (FTO), and the alternate supply breaker failure to close (FTC). Since both failure modes are included, and the data from NUREG/CR-6928 includes under voltage detection in the breaker boundary, the current modeling methodology is appropriate.	NDN-000-082-2007-0012, Rev 4; NDN-000-999-2007-0007, Rev 5; BFN IE PRA Model, Rev 8	Maintenance	The resolution requires no changes to the model or documentation. (1) No new methods were used. (2) There was a minor change in scope of the PRA by the addition of the events. (3) There was no change in the capability of the PRA."	The legacy resolution is not consistent with the current model. The current model does contain sequencer logic events (e.g., SEQFD0SEQ_082__DGA). This closes the F&O. This F&O is assessed as Closed.	SY-A8 remains Met. SY-B9 remains Met.
SY	SY-A19	SY-A19 is Met.	4-33	The unavailability or failure of a bus is not considered in the logic used to provide alternate electrical power supplies to other buses and boards. Example: U1_SDREC_A is used to re-energize 4kV SD Board A from 4kV SD Board 3A. However, the unavailability or failure of 4kV SD Board 3A does not fail the function (it should).	Unavailability or failure of the alternate power supply would prevent being able to credit it as an alternate source. Although the failure probability of a bus is much less than the failure probability of other equipment that could affect the power transfer (e.g., breaker demand failure), the unavailability could be substantial, especially during an outage of the other unit.	The failure of the bus has been included in the BFN PRA model. The applicable 4-kV shutdown board failure has been added to gates U1_SDREC_A, U2_SDREC_A, U3_SDREC_A, U1_SDREC_B, U2_SDREC_B, U3_SDREC_B, U1_SDREC_C, U2_SDREC_C, U3_SDREC_C, 1_SDREC_D, U2_SDREC_D, and U3_SDREC_D.	BFN IE PRA Model, Rev 8; NDN-000-999-2007-0007, Rev 005	Maintenance	The model revisions are not a change in the modeling approach. Twelve new events were added to the model with low probability of failure and are anticipated to have little impact on results. (1) No new methods were used. (2) There was no change in scope of the PRA. (3) There was no change in the capability of the PRA.	Model reviewed and events found to be properly located. For example, event TM_3BDAA211003EA was found to be appropriately included in the model and documentation. This F&O is assessed as Closed.	SY-A19 remains Met.

Table A-6 BFN Internal Events PRA F&O Closure Review Consensus Table

RU	SR	Prior PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
SY	SY-A8	SY-A8 is Met.	4-42	Table 3 of the data Notebook says that EDG boundaries included the output breakers, but the EDG system Notebook and the model have them as separate events. NUREG/CR-6928 lists breakers as WITHIN the boundary of the EDG.	Apparent inconsistency in data and component boundary definitions.	The EDG output breakers 1818, 1822, 1812, 1816, 1838, 1842, 1832, and 1836 have been included within the boundary of the EDG. The output breakers are no longer explicitly modeled. The EDG system Notebook and table 4 have been updated to reflect this change.	NDN-000-082-2007-0012, Rev, 004; BFN IE PRA Model Rev 8	Open	none	The system notebook did indicate that the failure of output circuit breakers was included within the EDG boundary. However, the CAFTA model still had separate events for breaker failure with probability included (CBKFC0BKR_211A_022). This F&O is assessed as Open.	SY-A8 remains Met.
SY	SY-A11 SY-B6 SY-B9	SY-A11 is Met. SY-B6 is Not Met SY-B9 is Met.	5-7	Control power for the RHRSW and RCW pumps is currently modeled such that failure of control power will result in failure of the pumps to continue running. Typically, control power is only needed for starting the pump	Apparent inconsistency in data and component boundary definitions.	Control power was placed under pump start gates for all pumps and air compressors where it was determined that control power was not necessary to maintain a running pump.	NDN-000-023-2007-0026, Rev, 005; MOR R8, NDN-000-032-2007-0016	Maintenance	The update represented a documentation change. (1) No new methods were used. (2) There was no change in scope of the PRA. (3) There was no change in the capability of the PRA.	A review of the system model identified control power associated to standby pumps including HPCI (U3_SUPP_HPI116), RCIC (U3_RCI_G01) as is appropriate. Air compressors A-D did not have any separate power supply, so it appears to be appropriately modeled. The modeling for RHRSW was also correctly developed since the pumps are normally in standby. The modeling appears to be correct. The modeling for RCW includes flags to truncate or include start faults and dc control power if the pumps are assumed to be in standby. The F&O is considered closed. This F&O is assessed as Closed.	SY-A11 remains Met. SY-B6 remains Not Met (see F&O 1-12). SY-B9 remains Met.
SY	SY-A2	SY-A2 is Met.	6-17	System models do not appear to incorporate operating experience in developing the fault tree logic. RHR Service Water operating experience does not appear to be complete or reviewed. HVAC Notebook says LERs and OER was reviewed, but none are listed (no evidence of the review). Similarly, for 120 VAC and others. CRD Notebook includes only a discussion of the BFN Fire, but no review of OE is presented.	Review of experience from BFN and other plants does not appear to be used in developing the fault tree system logic or data. In some cases, review of BFN OE is not included in the notebooks.	The write-up in the system notebooks discussing the level of SER, OER and LER reviews has been enhanced. There is no requirement in the ASME standard that requires a detailed listing or discussion of the generic or plant specific experience reviewed.	NDN-000-023-2007-0026, Rev, 005; NDN-000-075-2007-0010, Rev, 005	Maintenance	Resolution required a documentation change. (1) No new methods were used. (2) There was no change in scope of the PRA. (3) There was no change in the capability of the PRA.	A review of several system notebooks was performed. In the Core Spray Notebook section 3.2.6 included operating experience. Similar information was observed for Main Steam and Emergency Equipment Cooling Water System. The intent of the SR is considered to be met and the F&O closed. This F&O is assessed as Closed.	SY-A2 is now Met.
SY	SY-A14	SY-A14 is Met.	6-20	Event STRPL1STN_0750664, CS Suction Strainer Plugging, is only assumed for Large LOCA in the Model. The phenomenon causing plugging is not limited to large LOCA only, and is possible on Medium LOCA, SRV opening, etc. A question was asked to the analyst on this, and the reference to the absence of permanently installed air filters or other sources in the drywell. However, the debris, if present, would be swept into the suction strainer by any LOCA.	Affects multiple Initiating Events. Pre-existing material in the Torus can also affect the strainer plugging likelihood.	The strainer plugging event was added for MLOCA. All SRVs discharge directly to the suppression pool, so a stuck open SRV could not dislodge material from the drywell.	BFN IE PRA Model, Rev 8	Maintenance	Plugging was defined previously and only expanded to address medium LOCA. (1) No new methods were utilized. (2) The addition of the sump plugging for the medium LOCA represents a small change in the scope of the PRA. (3) There was no	The model was examined and the strainer modeling for one unit was examined. The gate U3_SP_STN_01 was found to contain strainer plugging and it was appropriately linked to RHR and CS suctions for large and medium LOCA. The omission of SRV LOCA and the basis presented by TVA is appropriate. The F&O is considered to be closed. This F&O is assessed as Closed.	SY-A14 remains Met.

Table A-6 BFN Internal Events PRA F&O Closure Review Consensus Table

RU	SR	Prior PRT CC II Assessment	Finding No.	Description	Prior Peer Review Assessment	Self-Assessment Closure Basis	Self-Assessment Reference Document(s)	Maint (M) or Upgrade?	Basis for Maint (M) or Upgrade	Independent Review Team Disposition	Independent Review Team SR Assessment
									change in the capability of the PRA.		
SY	SY-B11	SY-B11 is Met.	6-41	Fuel oil transfer pumps to refill the day tank are not part of the EDG boundary in NUREG/CR-6928.	Issue with EDG Component Boundary.	NUREG/CR-6928 states that the EDG boundary is the following: "The EDG boundary includes the diesel engine with all components in the exhaust path, electrical generator, generator exciter, output breaker, combustion air, lube oil systems, fuel oil system, and starting compressed air system, and local instrumentation and control circuitry. However, the sequencer is not included. For the service water system providing cooling to the EDGs, only the devices providing control of cooling flow to the EDG heat exchangers are included. Room heating and ventilating is not included." The "fuel oil system" is interpreted as up to the fuel oil day tank including the fuel oil transfer pumps. Each EDG at BFN has a 550-gallon day tank that provides enough fuel to operate for 2-1/2 hours at full load. Fuel is then transferred from the 40,000-gallon 7-day diesel storage tank with the diesel fuel oil transfer pump to continue operation. There is one 40,000-gallon 7-day diesel storage tank for each diesel generator, and it is included in the diesel generator boundary. The pumps that transfer fuel from the yard storage tank are outside the boundary and are not considered in the model.	NDN-000-082-2007-0012, Rev 004; BFN IE PRA Model, Rev 8	Maintenance	The current model was correct as presented. Closure documentation is the only change. (1) No new methods were used. (2) There was no change in scope of the PRA. (3) There was no change in the capability of the PRA.	The stated text from the NUREG/CR-6928 is correct. The fuel oil transfer pump should be considered outside the component boundary of the EDG. The current modeling and documentation (Figure 10) are correct and the F&O is resolved. This F&O is assessed as Closed.	SY-B11 remains Met.
SY	IE-C11 SY-A22	SY-A22 is Met. IE-C11 is Met.	6-50	Some of the MOVs credited in the ISLOCA Fault Tree are not tested to close against full DP. These MOVs are not originally included in the design as RCS isolation valves. Examples include 74-55 and 74-66 (note: this is not a complete list, but 2 of 4 valves reviewed were not in the MOVATs 89-10 program).	MOVs closing for ISLOCA are risk significant, with a RAW of greater than 2.	Assumption was added to the ISLOCA Notebook. Depressurization is not modeled in the ISLOCA initiator before valve closure. The probability of this failing to occur is only 5.077E-02. The fact that all ISLOCA events go directly to core damage without any mitigation actions is more than adequate to make up for not modeling the low probability of SRV failure.	vsloca_r8; BFN IE PRA Model, Rev 8	Open	none	A review of the ET representation identifies operator mitigation actions are included in the ET. This was also found to be the case when the ISLOCA modeling in the CAFTA model was reviewed (for example, gate U1_VRLOCA_002 includes gate U1_ISLV55_2 dealing with isolation). This F&O is assessed as Open.	SY-A22 remains met at Cat II. IE-C11 remains Met.

Table A-7 Disposition of Open BFN Internal Flooding F&Os

F&O IFSN-A8-01	It was stated that no credit was taken for the removal of water via the drain system, with the exception of spray events (≤100 gpm). No scenarios were modeled that included backflow through drain lines. Although this is reasonable based on the layout of the large open areas in the Reactor Building and Turbine Building, no discussion of the elimination of backflow was provided in the documentation.
Possible Resolution	Expand discussion in the Internal Flooding Notebook that explains how drain backflow was treated in the internal flood model. Include enough detail to justify screening. 1) What screening criteria was used? 2) How is the drain system configured? a. Are there separate drain systems in each building? (i.e., RB, TB, CB, etc.) b. Can a drain line become blocked downstream? c. Where does the water end up? (Sump on lower level?, Holding tank?, Outside?) 3) Include general references that can be validated by the reviewer, such as the system description and/or drawings used to support the assumptions for screening. 4) Is screening conservative? Why? This does not need to be a large effort but a statement that "any of the rooms within a building already show water propagating to the bottom elevation of that building" does not provide enough detail to demonstrate that the drain impacts were sufficiently assessed for screening.
Associated SRs	IFSN-A8
Plant Response	In the BFN Internal Flooding Analysis, it was determined that the only place that drain backflow could occur and potentially cause any issues would be in the lowest elevations of each building. The affect from this occurrence is already accounted for in each of the flooding scenarios as they all propagate to the lowest elevation. The drain lines are not connected for each building so water could not propagate from one building to another. The upper elevation drainage systems were not analyzed as a potential backflow situation as the drains are relatively small compared to the open hatches and stairwells that would cause the water to propagate to the lowest elevations. In addition, the areas in which the water would be susceptible to drainage are

Table A-7 Disposition of Open BFN Internal Flooding F&Os

	large rooms where the water would have to significantly fill in order to even reach a drain. Section 6.1.3 of the Internal Flooding Notebook explains that we screened drainage backflow from the analysis and provides the justification.
Impact on SPRA	This is a documentation issue. Therefore, there is no impact on the Seismic PRA.
F&O IFSN-A9-01	No specific flow rate calculations were performed. Flow rates were modeled to be the maximum flow rate for a given break category. For example, all flood events were assumed to result in a break flow of 2,000 gpm. This results in very conservative times to component failure. It could result in incorrect ranking of the risk importance of the flooding scenarios.
Possible Resolution	As a minimum, perform calculations to estimate the actual flow rates of modeled breaks for the most risk-significant scenarios.
Associated SRs	IFSN-A9
Plant Response	The BFN Internal Flooding Analysis conservatively assumed that the flows out of the pipe breaks were at the top end of each of the generic flow rate values. This was done to assure that we properly addressed the importance of each scenario. The pipe-break frequencies are given for the range of flows and the frequency does not change whether the top end flow rate or a lower flow rate is used unless it changes which range of flows you are using. The only time you would be concerned with the flow rate would be when you are performing an operator action to prevent water accumulation within a room. The BFN Internal Flooding analysis did not credit any of these types of operator actions except for in the RB at EL 519. The flow rates that could cause this elevation to flood could be from any water source in the RB, so the highest flow rate possible for both the flood scenario and the major flood scenario was used in calculating timing for the HRA action. This gives the smallest possible timeframe with which to perform the action and ensures that the results are conservative and risk insights are reasonable.
Impact on SPRA	Even with conservative flowrates assumed for pipe breaks, seismically induced flooding is not a significant contributor to seismic risk.

Table A-7 Disposition of Open BFN Internal Flooding F&Os

F&O IFSN-A10-01	Spray events in the RB general areas (multiple elevations) are assumed to result in a manual trip and are analyzed. Larger flooding events are not considered an initiating event unless operators fail to isolate the flood prior to reaching the level of equipment damage (5') at EL 519'. This appears to be an inconsistency between the spray and flood events. Although less frequent than spray events, flood events in these areas could in total be a significant contributor to CDF.
Possible Resolution	Develop some initiating event that models floods in the general areas of the RB, along with successful isolation of the flood prior to equipment damage on EL 519' of the RB. Based on the results, determine whether the entire group of these scenarios should be included in the internal flood model.
Associated SRs	IFSN-A10 IFEV-A1
Plant Response	When analyzing the spray events, it was assumed that for every spray scenario the operators would manually scram the reactor. This is a conservative assumption as the operators may not need to shut down the plant. By analyzing every spray scenario with a manual scram, we were able to see what the impact from a spray scenario would be to the plant. The flooding scenarios, on the other hand, were not analyzed as during an RB flood scenario all the water would propagate down to EL 519' of the RB. If the operators are successful in isolating the pipe rupture prior to reaching 5' in EL 519', the plant would not necessarily be tripped. While it is true that some equipment might be lost, which is similar to that seen for the spray events, the flooding analysis viewed the equipment impact separately from the flooding scenario as the flood has been terminated. Therefore, the impact from the equipment being lost would be characterized by the internal events PRA model. Each of the RB flooding scenarios that are successfully mitigated by the HRA action for EL 519' submergence will be reviewed to determine whether a potential scenario would exist or not. In addition, the spray scenarios will be reviewed to determine whether those are potential scenarios, and the results will be documented within the Internal Flooding Notebook.

Table A-7 Disposition of Open BFN Internal Flooding F&Os

Impact on SPRA	This F&O is not applicable to the SPRA since the seismic event is the initiating event. Human actions for high-magnitude seismic events that could potential damage piping are assumed to be failed.
F&O IFSN-A10-02	Only a 2000 gpm flood initiating event was modeled in the Unit 1 Shutdown Board Room A. Spray events were not modeled. Given that there are no drains nor indication in that room (and an informal analysis), there is a possibility that a spray event of 100 gpm could also result in similar consequences.
Possible Resolution	Perform a calculation at 100 gpm to determine whether a spray scenario is, in fact, a valid initiating event in this area. If so, include spray events in that area in the model.
Associated SRs	IFSN-A10
Plant Response	Each room was looked at for potential spray effects, including the 4KV Shutdown Board Room A. This spray scenario is in the model as U1-621-R02_025_S with a contribution of 1.54E-10 to CDF, which constitutes 0.002% of the Internal Flooding CDF for Unit 1. This spray scenario will be reviewed to ensure that it is treated appropriately within the model, and any changes will be documented in the next revision of the Internal Flooding Notebook.
Impact on SPRA	For a given area, the equipment damaged by seismically induced flooding is always assumed to be from the largest possible flood in that area. Even with conservative flowrates assumed for pipe breaks, seismically induced flooding is not a significant contributor to seismic risk.
F&O IFEV-A1-01	For spray events in the general areas of the RB, all the possible spray frequencies were added to obtain on combined frequency for one event. The impact of this spray event was the combined impact of all the possible spray events on that elevation.
Possible Resolution	Separate out spray events in these areas to provide a better picture of which spray sources and which impacted equipment are the more significant contributor.
Associated SRs	IFEV-A1 IFEV-A2

Table A-7 Disposition of Open BFN Internal Flooding F&Os

Plant Response	For the general areas of the RB, all spray scenarios were determined to occur at the same time, and all equipment affected by a certain system piping were all failed. Because this is such a big room, this modeling approach was too conservative. Each of the spray scenarios within the general area of the RB will be reviewed to determine which components can be failed by what portions of piping, and new scenarios will be developed to ensure that only the pipe ruptures that affect a component are used to fail a particular component.
Impact on SPRA	This F&O is not applicable to the SPRA since the seismic event is the initiating event. For a given area, the equipment damaged by seismically induced flooding is always assumed to be from the largest possible flood in that area. Even with conservative flowrates assumed for pipe breaks, seismically induced flooding is not a significant contributor to seismic risk.
F&O IFQU-A6-01	The HRA assessment needs to incorporate several items: a) Cues and indicators need to be documented in the first mitigation HRA (HFA_0_519FLOOD) b) With a), indicators should be assessed for flood damage c) PSFs need to be altered for general worst case in environment (e.g., radiation, etc.). This is because the flood mitigation actions are general and are not specific in place or time. d) Why is the belief in the adequacy of instruction set to "No"? For non-mitigation post-initiator HRAs: a) Needs to discuss blocked path for each scenario
Possible Resolution	Incorporate the missing pieces to the mitigation HRAs. a) Cues and indicators for the first mitigation HRA (HFA_0_519FLOOD) b) With a), indicators should be assessed for flood damage c) Alter PSFs for general worst case in environment (e.g., radiation, etc.) d) Alter or add some discussion on why the Belief in Adequacy is set to "No" For non-mitigation post-initiator HRAs: a) Discuss or incorporate blocked path for each scenario
Associated SRs	IFQU-A6

Table A-7 Disposition of Open BFN Internal Flooding F&Os

Plant Response	The HRA Assessment on HFA_0_519FLOOD was done generically as the only indication would be the Alarm coming in saying that there is water building up in EL 519'. The operator would be sent out to see if the alarm was valid and then try and isolate the pipe rupture. The Cues and Indicators will be updated to reflect the Alarm Indication. The flooding detectors are designed to get wet and would not be damaged by a flood. In addition, there are multiple flooding detectors within EL 519', so if any of the detectors work, the operators would still be able to mitigate the flood. The belief in adequacy of instruction was set to "No" as the operators would most likely question whether there is an actual flood within the RB. The operators would still comply with the procedure and perform the action as stated. There is a timing aspect included that is to assess whether the flood actually occurred. The PSFs were reviewed to assess whether an operator would experience any adverse situation outside of what would be experienced through everyday work. Because the flood and associated mitigation accident would occur prior to reactor trip, the shaping factors were consistent with a normal workload within the RB. Lighting would not be affected by the flood; heat/humidity would be normal for the areas that would be traversed. All of the areas within BFN are radiation areas, so there is no increased stress from radiation; isolating the pipe would be a simple action, and the stress was expected to be low as there is plenty of time to perform the action. It is expected that the action to close a couple of valves would not increase the stress on the operator. Each of the HRAs will be reviewed to determine what the impact would be from a blocked path, and this will be documented within the Internal Flooding Notebook.
Impact on SPRA	This F&O is not applicable to the SPRA since the seismic event is the initiating event. Human actions for high-magnitude seismic events that could potential damage piping are assumed to be failed.

Table A-7 Disposition of Open BFN Internal Flooding F&Os

F&O IFQU-A9-01	No modeling of direct effects due to a flooding event were identified. The rational was that for large flooding events in the RB, only those floods that resulted in flood levels reaching 5' in EL 519' were modeled. For those events, the required SSCs have failed due the indirect effects of the flooding. Therefore, the direct effects of the flooding need not be considered. It is our contention that floods in the RB that are successfully isolated before damage occurs to components on EL 519' should be included as initiators. These events will still result in damage to SSCs and direct failure to part of the breached system.
Possible Resolution	Include floods on the RB at EL 565' and above, even with successful isolation prior to equipment damage on EL 519'. For those events, model the direct failure of the breached system.
Associated SRs	IFQU-A9
Plant Response	This F&O is similar to F&O IFSN-A10-01. As mentioned in the response for that F&O, an operator may not need to scram the reactor for a loss of a component affected by a flooding event. Each of the RB flooding scenarios that are successfully mitigated by the HRA action for EL 519' submergence will be reviewed to determine whether a potential scenario would exist or not.
Impact on SPRA	This F&O is not applicable to the SPRA since the seismic event is the initiating event. Human actions for high-magnitude seismic events that could potential damage piping are assumed to be failed.

A.8 Identification of Key Assumptions and Uncertainties

The PRA Standard [8] includes several requirements related to identification and evaluation of the impact of assumptions and sources of uncertainty on the PRA results. NUREG-1855 [61] and EPRI 1016737 [51] 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 BFN Seismic PRA model quantification (see Section 5 of this submittal).
- Modeling uncertainties are considered in both the base internal events PRA and the Seismic PRA. Assumptions are made during the PRA development to

address a particular modeling uncertainty because there is not a single definitive approach. Plant-specific assumptions made for each of the BFN Seismic PRA technical elements are noted in the Seismic PRA 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 Seismic PRA peer review.

A summary of potentially important sources of uncertainty in the BFN Seismic PRA is listed in Table A-8.

Table A-8 Summary of Potentially Important Sources of Uncertainty

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

A.9 Identification of Plant Changes Not Reflected in the Seismic PRA

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

Appendix B

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

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

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

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

   <u>BFN Response</u>

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

   Section A.4 of Appendix A describes how each of the Finding-Level BFN SPRA F&Os were closed using an NRC-accepted process. Section A.6 of Appendix A describes how most of the Finding-Level BFN Internal Events F&Os were closed using an NRC-accepted process. Table A-5 shows the basis for closing these F&Os.

   Table A-6 gives the disposition of each open Internal Events and Internal Flooding F&Os. The BFN Seismic PRA Peer Review and Closure Review teams have reviewed these dispositions and have agreed that there is no impact on the BFN Seismic PRA. There are no open BFN Seismic PRA F&Os associated with the open Internal Events or Internal Flooding F&Os.

2. *Consideration of Staff Comments on Industry Documents*

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

for FLEX equipment and actions included in the SPRA, how the staff's comments on NEI 16-06 were appropriately considered, and the basis of as well as results from relevant sensitivity studies.

BFN Response

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

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

BFN Response to NRC Conclusion 1

No credit was taken for any portable offsite FLEX equipment in the BFN SPRA. The FLEX operator actions to align backup nitrogen to drywell control air for the SRVs were included in the model. No other FLEX systems were credited in the model.

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

BFN Response to NRC Conclusion 2

This is not applicable to this submittal; however, a peer review has been performed on the BFN SPRA model that includes the FLEX operator actions to align backup nitrogen to drywell control air for the SRVs that were included in the model. No other FLEX systems were credited in the model. This is documented in Section A.2 of Appendix A.

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

BFN Response to NRC Conclusion 3

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

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

BFN Response to NRC Conclusion 4

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

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

BFN Response to NRC Conclusion 6

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

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

BFN Response to NRC Conclusion 6

The FLEX operator actions to align backup nitrogen to drywell control air for the SRVs were included in the model. No other portable FLEX systems were credited in the model. The valves and piping used to align the backup nitrogen tanks are permanently installed. The nitrogen tanks have been evaluated by the fragility team and were determined to be seismically rugged. The random failure of the tanks was determined to be insignificant when compared to the failure rates of the modeled operator actions required for aligning the nitrogen supply.

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

BFN Response to NRC Conclusion 7

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

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

BFN Response to NRC Conclusion 8

The FLEX operator actions to align backup nitrogen to drywell control air for the SRVs were included in the model. No other portable FLEX systems were credited in the model. The valves and piping used to align the backup nitrogen tanks are permanently installed. The nitrogen tanks have been evaluated by the fragility team and were determined to be seismically rugged. The random failure of the tanks was determined to be insignificant when compared to the failure rates of the modeled operator actions required for aligning the nitrogen supply. Therefore, the uncertainty associated with failure rates of portable equipment is insignificant.

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

BFN Response to NRC Conclusion 9

PWROG-14003 was not used in the development of the BFN Seismic PRA.

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

BFN Response to NRC Conclusion 10

The FLEX operator actions to align backup nitrogen to drywell control air for the SRVs were included in the model. No other portable FLEX systems were credited in the model. The valves and piping used to align the backup nitrogen tanks are permanently installed. The nitrogen tanks have been evaluated by the fragility team and were determined to be seismically rugged. The random failure of the tanks was determined to be insignificant when compared to the failure rates of the modeled operator actions required for aligning the nitrogen supply.

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

BFN Response to NRC Conclusion 11

No credit is taken for debris removal, transportation of portable equipment, installation of equipment at a staging location, or routing of cables and hoses. The FLEX operator actions to align backup nitrogen to drywell control air for the SRVs were included in the model. No other portable FLEX systems were credited in the model. The valves and piping used to align the backup nitrogen tanks are permanently installed.

The operator actions associated with aligning the backup nitrogen are not significantly different than other ex-control room operator actions associated with other permanently installed equipment. These actions are properly evaluated using the existing HRA tools (HRA Calculator).

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

BFN Response to NRC Conclusion 12

The procedures associated with aligning the backup nitrogen are explicit, and the associated failure probabilities are directly analyzed by accepted approaches.

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

BFN Response to NRC Conclusion 13

There are no unique aspects of pre-initiator human failure events associated with aligning the backup nitrogen when compared to other ex-control room operator actions credited in the model.

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

<u>BFN Response</u>

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

3. *Combining potential improvements during detailed screening:*

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

   <u>BFN Response</u>

   Prior to the completion of the SPRA, select plant modifications were completed to reduce seismic risk based on IPEEE results and early SPRA quantifications. The modifications include 1) replacing the diesel auxiliary transformers and 2) installing fasteners on removable grating above the diesel generator intake and exhaust dampers.

   Additionally, studies have been completed to evaluate a potential modification of SEIS_5-2B Initiation Panels and Relays to improve SLERF. Based on risk sensitivity studies, SLERF appeared to be driven by SEIS_5-2B. An increase in the fragility value up to functional failure made a difference in SLERF but did not seem a realistic target since it would be over double the current fragility level. Therefore, an increase of 50% was evaluated and found to not significantly improve SLERF. The failure mode of the panels is anchorage. Completing a detailed SoV fragility calculation on these panels is not expected to provide more than approximately 20% increase in fragility. Based on these risk sensitivity studies, possible improvements to the anchorage were assessed to determine if it was feasible to perform a modification to obtain a substantial increase in the fragility value. The controlling failure is the concrete anchors. The welds between the panel and the sill channel is the next controlling failure mode. A slight improvement (~ 5%) to the concrete anchors would shift the controlling failure mode to the weld. Thus, two modifications would be required; first, a replacement of the concrete anchors and second, additional weld of panel to the sill channel. Interferences within the panel make replacement of the concrete anchors impractical. Additional possible modifications were considered such as top bracing or bracing from the floor. These panels are located in the auxiliary instrument room which contains many panels. Bracing to the floor would cause accessibility issues. Top bracing was also considered but deemed impractical due to the number of panels in the room and block walls as the only potential support point. Furthermore, these panels house relays that are operational during all modes of operation; while it may seem feasible to work modifications in during outages when some relay functions are not necessary, this would not be recommended due to those relays in the panel that are required for operability during the outage. Detailed studies on the amount of improvement to the fragility through modification was not evaluated further due to the impracticality of the modification.