Exelon Generation.

10 CFR 50.54(f)

RS-18-098

August 28, 2018

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

Peach Bottom Atomic Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-44 and DPR-56
NRC Docket Nos. 50-277 and 50-278

Subject: Seismic Probabilistic Risk Assessment Report, 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

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." (ML12333A170)
3. Exelon Generation Company, LLC Letter to USNRC, Seismic Hazard and Screening Report (Central and Eastern United States (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 (RS-14-071) (ML14090A247)
4. NRC Letter to Exelon Generation Company, LLC, Peach Bottom Atomic Power Station, Units 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 for Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident, dated April 20, 2015 (ML15051A262)
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)

6. Exelon Generation Company, LLC Letter to USNRC, Request for Extension of Due Date for Seismic Probabilistic Risk Assessment Submittal, dated March 15, 2018 (RS-18-033) (ML18074A303)
7. Exelon Generation Company, LLC Letter to USNRC, Supplement to Request for Extension of Due Date for Seismic Probabilistic Risk Assessment Submittal, dated March 28, 2018 (RS-18-043) (ML18088A020)
8. USNRC Letter to Exelon Generation Company, LLC, Peach Bottom Atomic Power Station, Units 2 and 3 – Response to Request for Extension of Seismic Probabilistic Risk Assessment Submittal, dated April 24, 2018 (ML18093B511)

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

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

Reference 4 contains the NRC Staff Assessment of the Peach Bottom Atomic Power Station, Units 2 and 3 seismic hazard submittal which concluded that the reevaluated seismic hazard prepared for Peach Bottom Atomic Power Station, Units 2 and 3 is suitable for other activities associated with the NTTF Recommendation 2.1: Seismic.

Reference 5 provided the NRC final seismic hazard evaluation screening determination results and the associated schedules for submittal of the remaining seismic hazard evaluation activities for Peach Bottom Atomic Power Station, Units 2 and 3. Reference 5 indicated that the Peach Bottom Atomic Power Station, Units 2 and 3 Seismic Probabilistic Risk Assessment (SPRA) was expected to be submitted by March 31, 2018. In References 6 and 7, Exelon Generation Company, LLC requested an extension of the Peach Bottom Atomic Power Station, Units 2 and 3 SPRA submittal date to September 28, 2018. This extension request was approved by the NRC in Reference 8.

The enclosure to this letter contains the Peach Bottom Atomic Power Station, Units 2 and 3 SPRA Summary Report which provides the information requested in Enclosure 1, Item (8) B. of the 10 CFR 50.54(f) letter.

This letter closes Regulatory Commitment No. 1 of Reference 3.

This letter contains no new regulatory commitments or revisions to existing regulatory commitments.

If you have any questions regarding this report, please contact David J. Distel at 610-765-5517.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 28th day of August 2018.

Respectfully submitted,

D. P. Helker

David P. Helker
Manager - Licensing & Regulatory Affairs
Exelon Generation Company, LLC

Enclosure: Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3 Seismic Probabilistic Risk Assessment in Response to 50.54(f) Letter with Regard to NTTF 2.1 Seismic, Revision 0, August 28, 2018

cc: Regional Administrator - NRC Region I
NRC Senior Resident Inspector – Peach Bottom Atomic Power Station
NRC Project Manager, NRR – Peach Bottom Atomic Power Station
Mr. Brett A. Titus, NRR/JLD/JCBB, NRC
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Director, Bureau of Radiation Protection – Pennsylvania Department of Environmental Resources
D. A. Tancabel, State of Maryland
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ENCLOSURE

Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3 Seismic Probabilistic Risk Assessment in Response to 50.54(f) Letter with Regard to NTTF 2.1 Seismic

Revision 0

August 28, 2018

(192 Pages)

PEACH BOTTOM ATOMIC POWER STATION (PBAPS) UNITS 2 AND 3 SEISMIC PROBABILISTIC RISK ASSESSMENT IN RESPONSE TO 50.54(F) LETTER WITH REGARD TO NTTF 2.1 SEISMIC

REVISION 0

August 28, 2018

PBAPS UNITS 2 AND 3 SEISMIC PROBABILISTIC RISK ASSESSMENT
SUMMARY REPORT

Table of Contents

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 Peach Bottom Atomic Power Station (PBAPS) 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 NRC [3]. That comparison concluded that the ground motion response spectrum (GMRS), which was developed based on the reevaluated seismic hazard, exceeds the design basis seismic response spectrum in the 1 to 10 Hz range, and a seismic risk assessment is required. A seismic PRA (SPRA) has been developed to perform the seismic risk assessment for PBAPS in response to the 50.54(f) letter, specifically item (8) in Enclosure 1 of the 50.54(f) letter.

This report describes the seismic PRA developed for PBAPS 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 PBAPS, identifying which structures, systems, and components (SSCs) are important to seismic risk, and describing plant-specific seismic issues and associated actions planned or taken in response to the 50.54(f) letter.

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

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

2.0 Information Provided in This Report

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

(1) The list of the significant contributors to SCDF for each seismic acceleration hazard interval, including importance measures (e.g. Fussell-Vesely)
(2) A summary of the methodologies used to estimate the SCDF and LERF, including the following:
   i. Methodologies used to quantify the seismic fragilities of SSCs, together with key assumptions
   ii. SSC fragility values with reference to the method of seismic qualification, the dominant failure mode(s), and the source of information
   iii. Seismic fragility parameters
   iv. Important findings from plant walkdowns and any corrective actions taken
   v. Process used in the seismic plant response analysis and quantification, including the specific adaptations made in the internal events PRA model to produce the seismic PRA model and their motivation
   vi. Assumptions about containment performance
(3) Description of the process used to ensure that the SPRA is technically adequate, including the dates and findings of any peer reviews
(4) Identified plant-specific vulnerabilities and actions that are planned or taken

Note that 50.54(f) letter Enclosure 1 paragraphs 1 through 6, regarding the seismic hazard evaluation reporting, also apply, but have been satisfied through the previously submitted PBAPS Seismic Hazard Submittal [3]. Further, 50.54(f) letter Enclosure 1 paragraph 9 requests information on the Spent Fuel Pool. This information has been submitted separately [60].

Table 2-1 provides a cross-reference between the 50.54(f) reporting items noted above and the location in this report where the corresponding information is discussed.

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

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

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

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

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

The content of this report is organized as follows:

Section 3 provides information related to the PBAPS seismic hazard analysis.

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

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

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

Section 7 provides references.

Section 8 provides a list of acronyms used.

Appendix A provides an assessment of SPRA Technical Adequacy for Response to NTTF 2.1 Seismic 50.54(f) Letter, including a summary of PBAPS SPRA peer review.

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

50.54(f) Letter Reporting Item	Description	Location in this Report
1	List of the significant contributors to SCDF for each seismic acceleration hazard interval, including importance measures	Section 5
2	Summary of the methodologies used to estimate the SCDF and LERF	Sections 3, 4, 5
2i	Methodologies used to quantify the seismic fragilities of SSCs, together with key assumptions	Section 4
2ii	SSC fragility values with reference to the method of seismic qualification, the dominant failure mode(s), and the source of information	Tables 5.4-2, 5.4-3, 5.5-2, 5.5-3 provide fragilities (Am and beta), failure mode information, and method of determining fragilities for the top risk significant SSCs based on standard importance measures such as Fussell-Vesely (FV). Seismic qualification reference is not provided as it is not relevant to development of SPRA.
2iii	Seismic fragility parameters	Tables 5.4-2, 5.4-3, 5.5-2, 5.5-3 provide fragilities (Am and beta) information for the top risk significant SSCs based on standard importance measures such as FV.
2iv	Important findings from plant walkdowns and any corrective actions taken	Section 4.2 addresses walkdowns and walkdown insights.
2v	Process used in the seismic plant response analysis and quantification, including specific adaptations made in the internal events PRA model to produce the seismic PRA model and their motivation	Sections 5.1 and 5.2
2vi	Assumptions about containment performance	Sections 4.3 and 5.5

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

50.54(f) Letter Reporting Item	Description	Location in this Report
3	Description of the process used to ensure that the SPRA is technically adequate, including the dates and findings of any peer reviews	App. A describes the assessment of SPRA technical adequacy for the 50.54(f) submittal and results of the SPRA peer review
4	Identified plant-specific vulnerabilities and actions that are planned or taken	Section 6

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

SPID Section 6.8 Item [1] Description	Location in this Report
A report should be submitted to the NRC summarizing the SPRA inputs, methods, and results.	Entirety of the submittal addresses this.
The level of detail needed in the submittal should be sufficient to enable NRC to understand and determine the validity of all input data and calculation models used	Entirety of the submittal addresses this and identifies key methods of analysis and referenced codes and standards.
The level of detail needed in the submittal should be sufficient to assess the sensitivity of the results to all key aspects of the analysis	Entirety of the submittal addressed this. Sensitivities are discussed in the following sections: • 5.7 (SPRA model sensitivities)
The level of detail needed in the submittal should be sufficient to make necessary regulatory decisions as a part of NTTF Phase 2 activities.	Entirety of the submittal template addresses this.
It is not necessary to submit all of the SPRA documentation for such an NRC review. Relevant documentation should be cited in the submittal, and be available for NRC review in easily retrievable form.	Entire report addresses this. This report summarizes important information from the SPRA, with detailed information in lower tier documentation
Documentation criteria for a SPRA are identified throughout the ASME/ANS Standard [4]. Utilities are expected to retain that documentation consistent with the Standard.	This is an expectation relative to documentation of the SPRA that the utility retains to support application of the SPRA to risk-informed plant decision-making.

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

3.0 PBAPS Seismic Hazard and Plant Response

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

PBAPS is located partly in Peach Bottom Township, York County, partly in Drumore Township, Lancaster County, and partly in Fulton Township, Lancaster County, in southeastern Pennsylvania on the westerly shore of Conowingo Pond at the mouth of Rock Run Creek. The regional and local site geology is described in additional detail in PBAPS NTTF 2.1 Seismic Hazard submittal [3]. The site lies within the Piedmont Upland Section of the Piedmont Physiographic Province of the Appalachian Highlands.

PBAPS is predominantly a firm rock site. The general site conditions consist of residual soils overlying partially weathered rock grading into hard metamorphic sedimentary rocks. Beneath the residual soils, there is 20 ft (6.1m) of firm rock (schist) over Paleozoic or Precambrian hard rock (schist). PBAPS consists of two units (2 and 3) with both reactor buildings supported on the hard rock, with softer rock and soils above this layer having been removed prior to placement of the reactor buildings. Foundations of the remaining important structures also extend to this hard rock layer.

The GMRS at PBAPS is defined at the foundation control point corresponding to the Reactor Building/Turbine Building/Radwaste Building/Main Control Room complex (RB/TB/RW/ MCR complex).

The following four Foundation Input Response Spectra (FIRS) are developed for the structures as summarized:

- *FIRS1 – Reference Rock Hazard (Hard Rock)*: This corresponds to the RB/TB/RW/MCR complex, and used in modeling of the Pump Structure (PS) and Emergency Cooling Tower (ECT). This FIRS has been designated as the GMRS.
- *FIRS2 - Soil Column Outcrop Response at EL 105 ft (top of hard rock) with 20 ft of Compacted Backfill Above*: This is used in modeling the emergency Diesel Generator Building (DGB) which is located at elevation 125 ft but has a pile and shear wall foundation that extends to the top of the hard rock at elevation 105 ft.
- *FIRS3 - Surface Response at EL 136 ft with Moderately Weathered Rock over Hard Rock:* This is used in modeling Yard Equipment.
- *FIRS4 - Surface Response @ EL 117 ft with 40 ft of Compacted Backfill Below*: Used in uncertainty quantification to assess the impact of unbalanced embedment on the PS.

Additional site description and profile development are described in the PBAPS NTTF 2.1 Seismic Hazard submittal [3].

3.1 Seismic Hazard Analysis

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

The seismic hazard analysis determines the annual frequency of exceedance for selected ground motion parameters. The analysis involves use of earthquake source models, ground motion attenuation models, characterization of the site response (e.g. soil column), and accounts for the uncertainties and randomness of these parameters to arrive at the site seismic hazard. Detailed information regarding the PBAPS site hazard was provided to the NRC in the seismic hazard information submitted in response to the NTTF 2.1 Seismic information request [3]. As further discussed below, a supplemental seismic hazard analysis has been performed for PBAPS [6].

3.1.1 Seismic Hazard Analysis Methodology

A probabilistic seismic hazard analysis was performed [6] to support the PBAPS Seismic PRA in lieu of the NTTF 2.1 submittal [3] since the site analysis develops the additional elements required for the Seismic PRA such as FIRS, hazard-consistent strain-compatible properties, and vertical ground motions.

To perform the site response analyses for PBAPS, a random vibration theory approach was employed. This process is consistent with existing NRC guidance and the SPID [2]. The guidance contained in Appendix B of the SPID [2] on incorporating epistemic uncertainty in shear-wave velocities, non-linear dynamic properties and source spectra was followed for PBAPS in addition to development of High Frequency (HF) and Low Frequency (LF) controlling earthquakes (control motions) per recommendations in Regulatory Guide 1.208 [46] for mean annual frequency of exceedance corresponding to 1E-02, 1E-03, 1E-04, 1E-05, and 1E-06.

The GMRS at PBAPS is defined at the base of the RB/TB/RW/MCR complex corresponding to the hard reference rock (shear wave velocity greater than or equal to 9,200 fps). FIRS were developed for additional structures at the elevations described in Section 3.0.

The shear wave velocity profiles were very similar to the NTTF 2.1 Seismic Hazard submittal [3] shear wave velocity profiles, with the exception of FIRS2 soil profile and FIRS4, where the moderately weathered rock was replaced with compacted backfill. The compacted backfill is specified as clean well graded imported sand and gravel or crushed rock with no more than 5 percent passing a #200 sieve with a minimum compaction to 75% relative density [7].

The idealized shear wave velocity profiles for FIRS2, FIRS3, and FIRS4 are presented in Figures 3.1.1-1 to 3.1.1-3, respectively. Note that the GMRS and FIRS1 are at the top of the hard reference rock and there is no variation considered in these rock properties, consistent with the assumption in the NTTF 2.1 Seismic Hazard submittal [3].

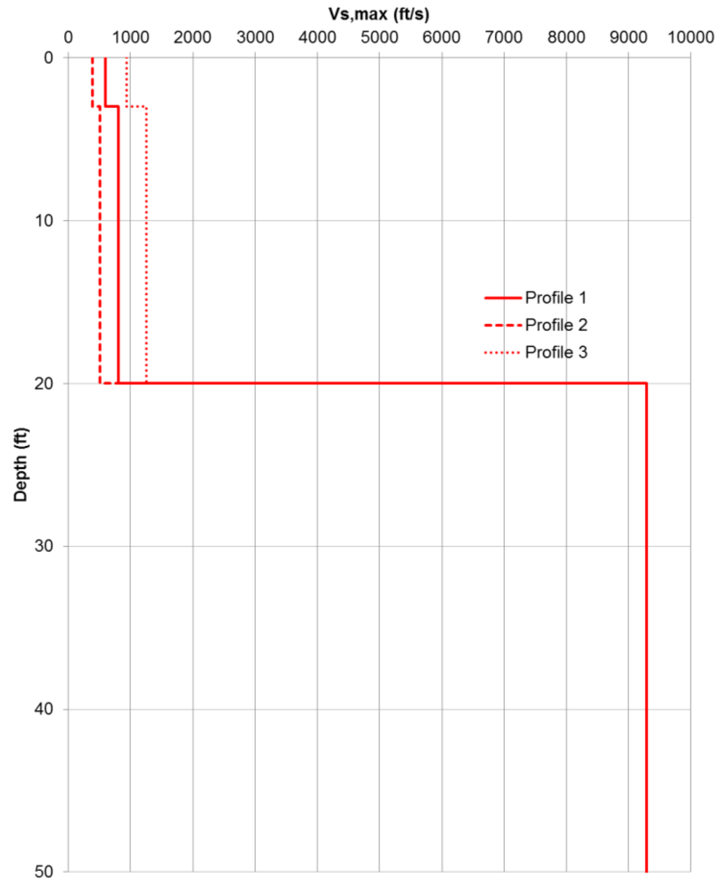

Figure 3.1.1-1. Idealized Shear Wave Velocity (Vs) Profiles Representing Epistemic Uncertainty (FIRS2)

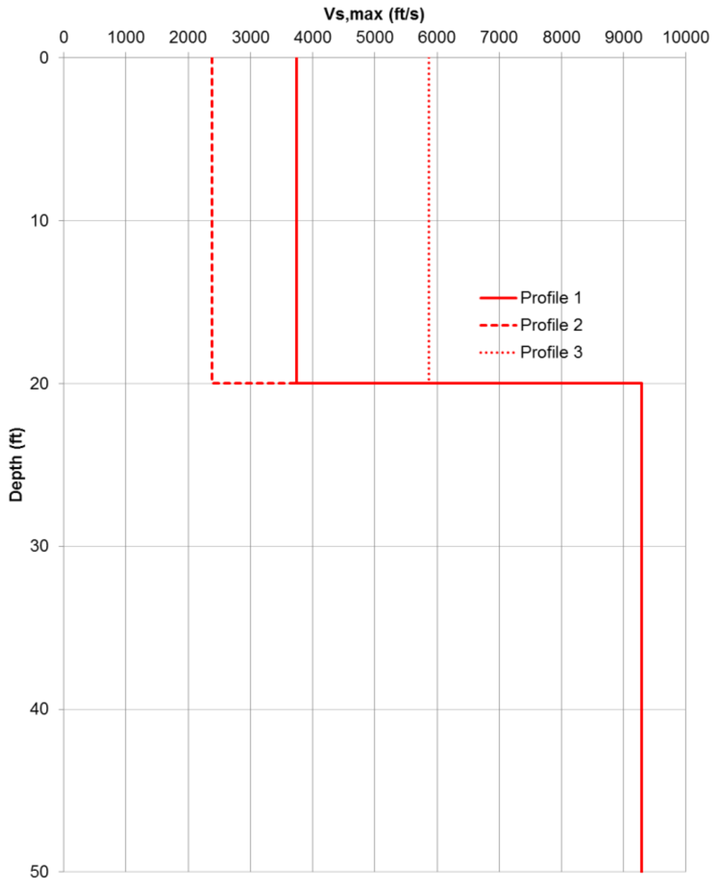

Figure 3.1.1-2. Idealized Shear Wave Velocity (Vs) Profiles Representing Epistemic Uncertainty (FIRS3)

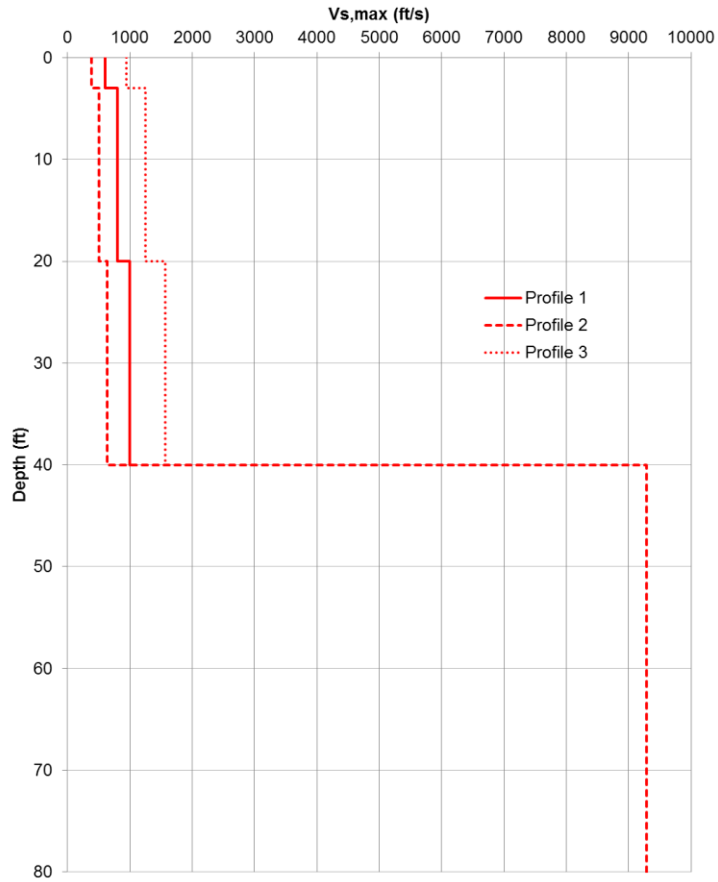

Figure 3.1.1-3. Idealized Shear Wave Velocity (Vs) Profiles Representing Epistemic Uncertainty (FIRS4)

To accommodate the full range in expected dynamic material behavior for the firm rock profile (FIRS3), linear analyses, as well as nonlinear analyses, were included in the site response analyses, with equal weights given to each approach. This approach is consistent with the approach of the NTTF 2.1 Seismic Hazard submittal [3]. Only nonlinear curves were considered in the analyses for the FIRS soil profiles including compacted backfill overlying the hard rock (FIRS2 and FIRS4).

The results of the site response analyses consist of amplification factors which 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 presented 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 [2], a minimum median amplification value of 0.5 was employed in the present analysis. Table 3.1.1-1 and Figure 3.1.1-4 present the mean and fractile exceedance frequencies for hard reference rock at 100 Hz. Sample amplification factors are presented in Figure 3.1.1-5.

Table 3.1.1-1. PBAPS Mean and Fractile Exceedance Frequencies – Hard Reference Rock PGA (100 Hz) Equivalent to GMRS/FIRS1/FIRS2

Amplitude (g)	Mean	Exceedance Frequency				
		0.05	0.16	0.50	0.84	0.95
0.0001	1.456E-01	3.694E-02	7.695E-02	1.528E-01	2.055E-01	2.492E-01
0.00025	8.901E-02	2.179E-02	4.686E-02	9.178E-02	1.262E-01	1.692E-01
0.0005	5.433E-02	1.409E-02	2.960E-02	5.363E-02	7.817E-02	1.060E-01
0.00075	3.915E-02	1.095E-02	2.195E-02	3.676E-02	5.751E-02	7.672E-02
0.001	3.061E-02	9.068E-03	1.758E-02	2.776E-02	4.593E-02	6.056E-02
0.0015	2.130E-02	7.123E-03	1.258E-02	1.851E-02	3.208E-02	4.365E-02
0.002	1.628E-02	5.840E-03	9.782E-03	1.389E-02	2.358E-02	3.420E-02
0.003	1.098E-02	4.285E-03	6.734E-03	9.469E-03	1.499E-02	2.413E-02
0.005	6.581E-03	2.823E-03	4.010E-03	5.741E-03	8.331E-03	1.505E-02
0.0075	4.350E-03	1.891E-03	2.654E-03	3.845E-03	5.397E-03	1.014E-02
0.01	3.231E-03	1.381E-03	1.864E-03	2.833E-03	4.139E-03	7.707E-03
0.015	2.110E-03	8.219E-04	1.114E-03	1.798E-03	2.937E-03	5.240E-03
0.02	1.550E-03	5.542E-04	7.344E-04	1.287E-03	2.204E-03	3.933E-03
0.03	9.922E-04	2.736E-04	4.409E-04	8.015E-04	1.433E-03	2.634E-03
0.05	5.533E-04	1.187E-04	1.949E-04	4.041E-04	8.686E-04	1.613E-03
0.075	3.389E-04	6.301E-05	1.032E-04	2.214E-04	5.445E-04	9.986E-04
0.1	2.345E-04	4.220E-05	6.716E-05	1.504E-04	3.628E-04	6.848E-04
0.15	1.345E-04	2.127E-05	3.651E-05	8.698E-05	2.119E-04	3.860E-04
0.2	8.794E-05	1.253E-05	2.153E-05	5.708E-05	1.497E-04	2.486E-04
0.3	4.584E-05	6.718E-06	1.113E-05	2.984E-05	8.916E-05	1.264E-04
0.5	1.819E-05	2.596E-06	4.090E-06	1.174E-05	3.812E-05	4.898E-05
0.75	7.908E-06	9.729E-07	1.627E-06	4.775E-06	1.632E-05	2.194E-05
1	4.118E-06	4.410E-07	8.203E-07	2.352E-06	8.321E-06	1.229E-05
1.5	1.486E-06	1.158E-07	2.416E-07	7.903E-07	2.862E-06	4.895E-06
2	6.666E-07	3.485E-08	8.668E-08	3.289E-07	1.219E-06	2.325E-06
3	1.898E-07	4.911E-09	1.838E-08	8.039E-08	3.352E-07	7.071E-07
5	3.074E-08	3.723E-11	1.309E-09	8.468E-09	4.647E-08	1.356E-07
7.5	5.917E-09	2.200E-29	7.477E-13	9.979E-10	8.118E-09	2.930E-08
10	1.661E-09	2.200E-29	3.991E-24	1.702E-10	2.228E-09	8.408E-09

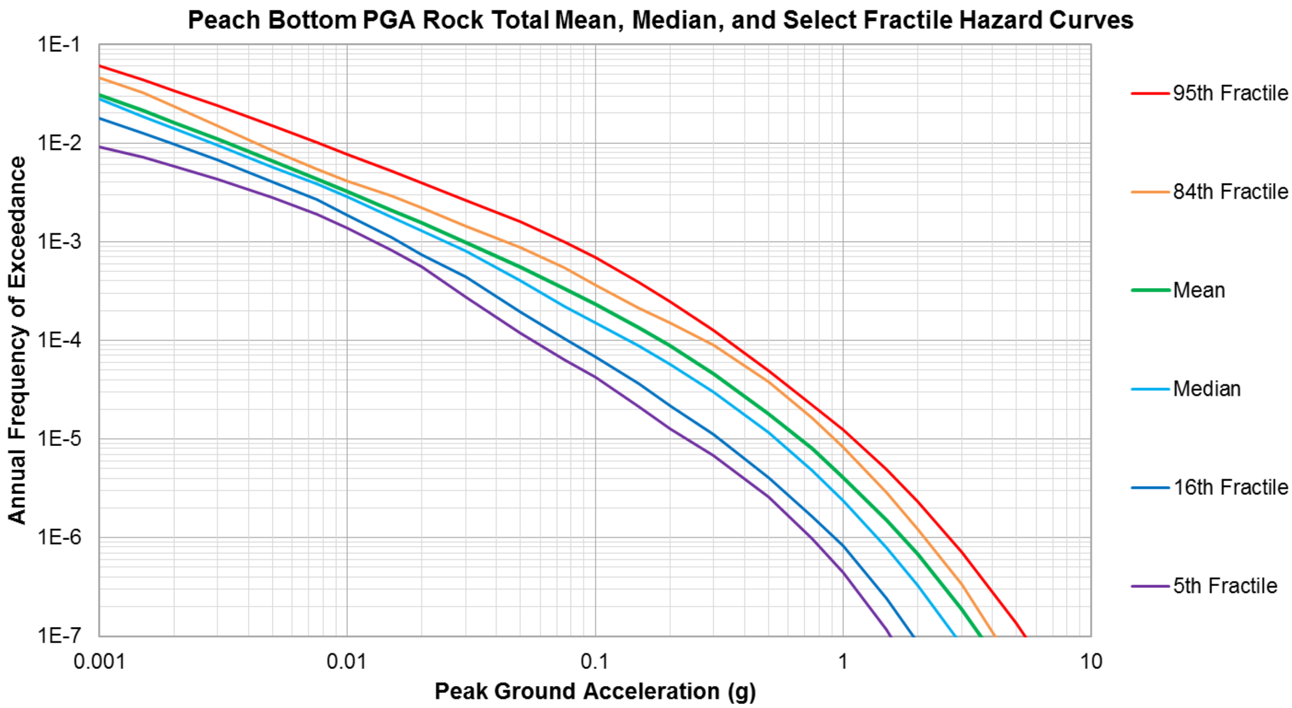

Figure 3.1.1-4. PGA (100 Hz) Fractile Hazard Curves for PBAPS (Hard Reference Rock) Equivalent to GMRS/FIRS1/FIRS2

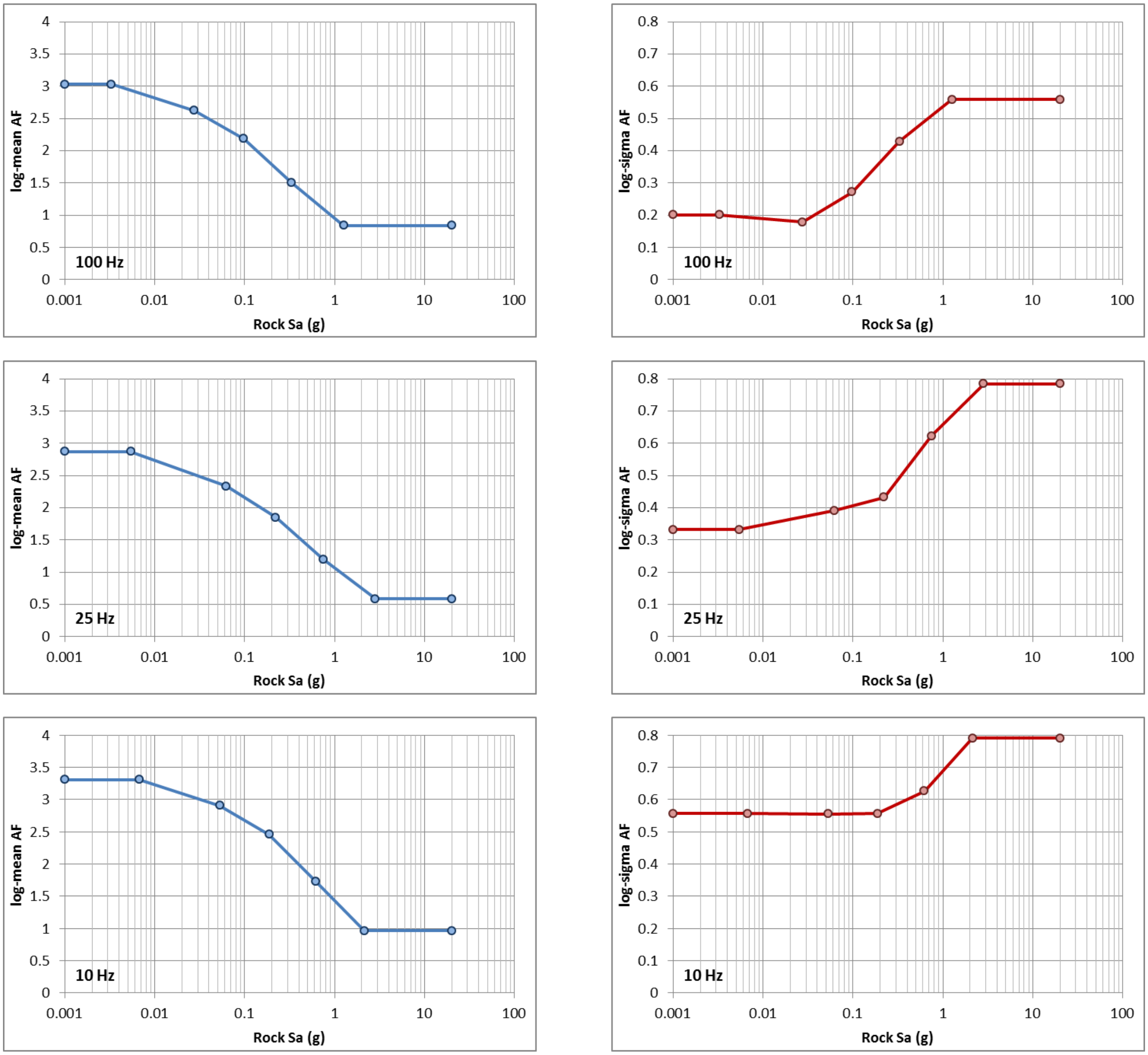

Figure 3.1.1-5. Top of FIRS2 Soil Profile Site Amplification Factor and Logarithmic Sigmas (100 Hz, 25 Hz, and 10 Hz)

FIRS1 is equivalent to the GMRS and corresponds to the hard reference rock. FIRS2 is also equivalent to the GMRS since the FIRS2 control point is defined at the top of the hard reference rock and the site response analyses performed confirmed that there was insignificant impact from the compacted backfill on top of the hard reference rock on FIRS2. FIRS3 and FIRS4 were developed in accordance with Regulatory Guide 1.208 [46]. Sixty randomizations were performed for the site response for each epistemic branch in the soil logic tree, compared to a minimum of thirty recommended in the SPID [2]. The site response analyses were completed using the HF and LF control motions. Site-specific horizontal hazard curves for each of the FIRS (FIRS2 top of soil profile, FIRS3, and FIRS4) site conditions were used and were developed using Approach 3 of NUREG/CR-6728 [8].

The reference earthquake ground motion to which the fragilities are referenced is represented by the horizontal GMRS at the RB/TB/RW/MCR complex foundation control point, which corresponds to the hard reference rock. However, a sensitivity study was performed to determine the effect on the results of the SPRA if a higher reference earthquake level were considered. See Appendix A for further discussion.

Peak Ground Acceleration (PGA) is the ground motion parameter used for the Seismic PRA.

Vertical ground motions were developed by applying Vertical/Horizontal (V/H) ratios to the horizontal GMRS and FIRS. For the GMRS and FIRS founded on hard reference rock, the Central and Eastern United States (CEUS) Rock V/H ratio (PGA in the range of 0.2g to 0.5g) was used directly from NUREG/CR-6728 [8]. For the control point corresponding to the top of the FIRS2 soil profile, FIRS3, and FIRS4, review of multiple V/H ratios including CEUS Rock V/H ratios from NUREG/CR-6728 [8], and Western United States V/H ratios shifted in the frequency domain by a factor of 3 to match the peak in the CEUS V/H ratio was performed, and the CEUS Rock V/H ratio (PGA in the range of 0.2g to 0.5g) was adopted for all FIRS.

Table 3.1.1-2 and Figure 3.1.1-6 provide the horizontal and vertical GMRS/FIRS1/FIRS2.

Table 3.1.1-2. Smoothed Horizontal and Vertical GMRS/FIRS1/FIRS2

Frequency (Hz)	Horizontal GMRS/FIRS1/FIRS2 (g)	Vertical GMRS/FIRS1/FIRS2 (g)
0.1	3.93E-03	2.94E-03
0.125	6.13E-03	4.60E-03
0.15	8.83E-03	6.63E-03
0.2	1.31E-02	9.86E-03
0.3	1.97E-02	1.47E-02
0.4	2.63E-02	1.97E-02
0.5	3.29E-02	2.46E-02
0.6	3.94E-02	2.95E-02
0.7	4.57E-02	3.42E-02
0.8	5.10E-02	3.83E-02
0.9	5.63E-02	4.22E-02
1	6.24E-02	4.68E-02
1.25	7.87E-02	5.91E-02
1.5	9.70E-02	7.28E-02
2	1.28E-01	9.62E-02
2.5	1.54E-01	1.15E-01
3	1.84E-01	1.38E-01
4	2.45E-01	1.84E-01
5	2.94E-01	2.21E-01
6	3.40E-01	2.55E-01
7	3.83E-01	2.87E-01
8	4.23E-01	3.18E-01
9	4.60E-01	3.45E-01
10	4.89E-01	3.67E-01
12.5	5.43E-01	4.31E-01
15	5.80E-01	4.83E-01
20	6.21E-01	5.58E-01
25	6.24E-01	5.94E-01
30	5.94E-01	5.93E-01
35	5.52E-01	5.74E-01
40	5.18E-01	5.57E-01
45	4.90E-01	5.43E-01
50	4.66E-01	5.31E-01
60	4.23E-01	4.82E-01
70	3.84E-01	4.38E-01
80	3.53E-01	4.02E-01
90	3.27E-01	3.73E-01
100	3.10E-01	3.53E-01

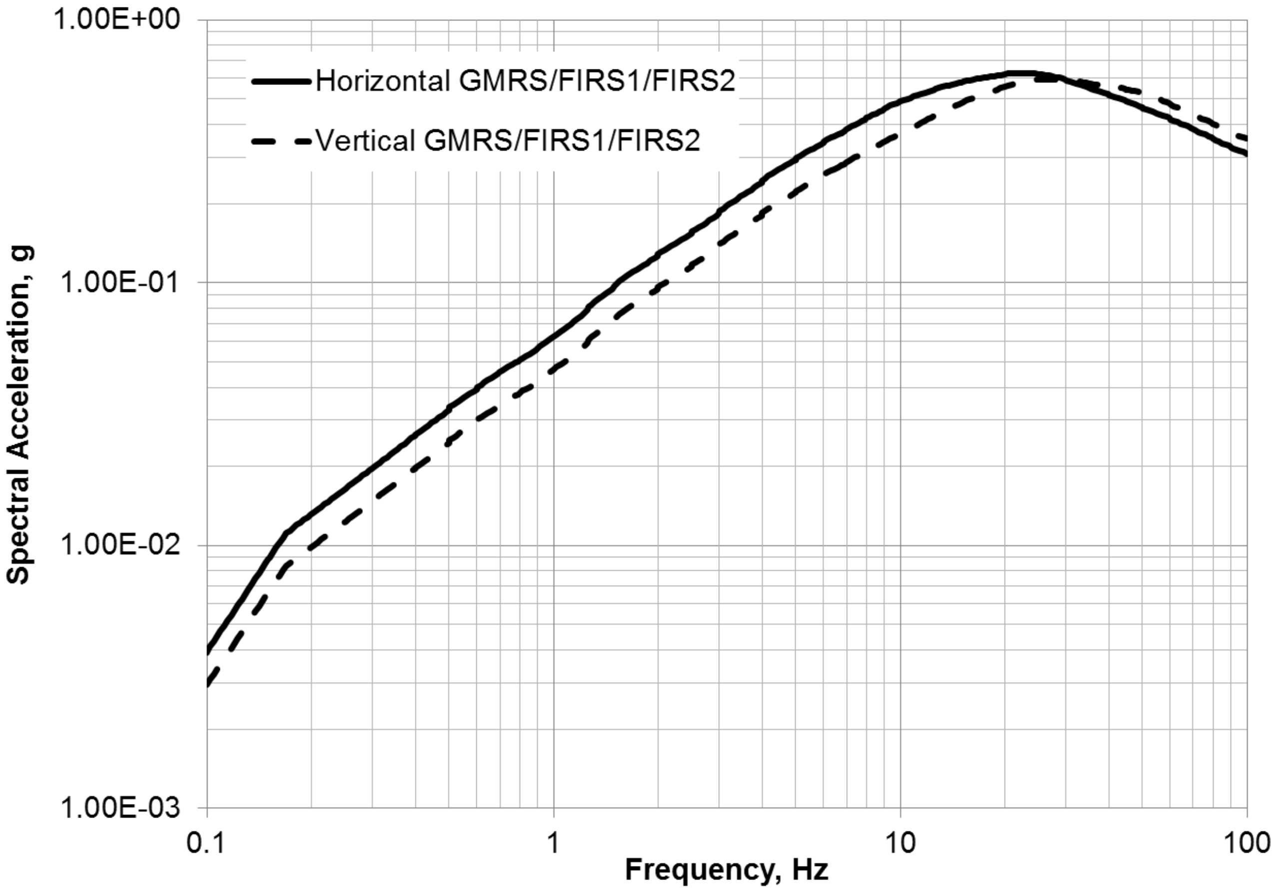

Figure 3.1.1-6. Horizontal and Vertical GMRS/FIRS1/FIRS2

3.1.2 Seismic Hazard Analysis Technical Adequacy

The PBAPS hazard analysis was subjected to an independent peer review against the pertinent requirements in the PRA Standard [4]. The Seismic PRA was peer reviewed relative to Capability Category II for the full set of requirements in the Standard. After completion of the peer review and the disposition of the peer review findings, the full set of supporting requirements was met. The seismic hazard analysis was determined to be acceptable for use in the Seismic PRA.

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

3.1.3 Seismic Hazard Analysis Results and Insights

Table 3.1.1-1 and Figure 3.1.1-4 provide the final seismic hazard results used as input to the PBAPS Seismic PRA, in terms of exceedance frequencies as a function of PGA level for the mean and several fractiles at hard reference rock.

The main contributors to seismic hazard at PBAPS site are the host background source zones, followed by the Charleston and New Madrid Fault System repeated large magnitude earthquake (RLME) sources. Additionally, the New Madrid Fault System contributes more than Charleston at 1 Hz for all mean annual frequency of exceedance (MAFE) levels above 1E-06, and less than Charleston at 5 and 10 Hz for all MAFE levels. At 2.5 Hz, Charleston and New Madrid Fault System contribute about the same at the 1E-02 MAFE, and Charleston contributes more at lower MAFEs. For high frequencies (5 and 10 Hz), the host background source zones are the main contributors to seismic hazard at the PBAPS site. The background source zones are also the main contributors to seismic hazard at PBAPS for low frequencies (1 and 2.5 Hz). The Charleston and New Madrid Fault System RLME sources have small, but noticeable peaks at the 1E-02, 1E-03, and 1E-04 MAFE levels.

Sensitivities of the hard rock hazard to the ground motion models and most significant portions of the seismic source model were performed. The sensitivity analyses indicate a large uncertainty in the rock hazard due to the suite of ground motion models. Also, the sensitivity analyses indicate that the ground motion models for the background seismic source zones and the seismicity rates for the dominant background zone contribute the most to the uncertainty for spectral frequencies corresponding to the PGA (100 Hz) and 1 Hz.

The Central and Eastern United States Seismic Source Characterization (CEUS-SSC) [9; 10] concluded its data gathering efforts in 2008. As a result, a literature search of published and unpublished data was completed to identify any data that may have an impact on the SSC or any other site-specific modifications based on new information. The CEUS-SSC [9] developed comprehensive up-to-data databases including a comprehensive earthquake catalog through December 31, 2008 and a compilation of paleo-seismic data. For the CEUS-SSC Project, comprehensive Data Evaluation Tables were prepared. Literature that post-dates the CEUS-SSC was evaluated to confirm the lack of local seismic sources. An updated earthquake catalog post-dating the CEUS-SSC through January 31, 2015 was developed along with induced seismicity. After the review and studies of new information, it was concluded that the CEUS-SSC model did not require an update.

The PSHA performed incorporated the entire CEUS-SSC logic tree published in NUREG-2115 [9] with its revisions published in 2015 [10]. 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 [9].

3.1.4 Horizontal and Vertical GMRS

This section provides the control point horizontal and vertical GMRS.

The horizontal and vertical GMRS at the control point is tabulated in Table 3.1.1-2 and presented in Figure 3.1.1-6. The development of the control point response spectra is summarized in Section 3.1.1 and further described in detail in the PBAPS PSHA report [6].

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

4.1 Seismic Equipment List

For the PBAPS SPRA, a seismic equipment list (SEL) was developed that includes those SSCs that are important to achieving safe shutdown following a seismic event, and to mitigating radioactivity release if core damage occurs, and that are included in the SPRA model. The methodology used to develop the SEL is consistent with the guidance provided in EPRI 3002000709 [11].

4.1.1 SEL Development

The PBAPS SPRA SEL is developed consistent with the requirements and guidance identified in the following industry references:

- Part 5 (Addenda B) of the American Society of Mechanical Engineers (ASME) / American Nuclear Society (ANS) PRA Standard (RA-Sb-2013) [4]
- Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic, EPRI Report 1025287 [2]
- Seismic Probabilistic Risk Assessment Implementation Guide, EPRI Report 3002000709, December 2013 [11]

The EPRI 2013 Seismic PRA Implementation Guide (SPRAIG) [11] provides the following general guidance as one method to develop an initial SEL:

1. Identify SSCs important to safe shutdown from Full-Power PRA Models
2. Identify SSCs from Review of Seismic evaluation performed for the IPEEE
3. Identify structures and passive components important to seismic response (including identification of SSCs from secondary hazard considerations)
4. Identify Additional SSCs from Plant Walkdown
5. Disposition SSCs on SEL
6. Review and document SEL

The above EPRI approach is followed for the PBAPS SPRA SEL development.

The PBAPS SPRA SEL is developed by using the PBAPS existing full-power PRA models as the starting point. Use of the PRA models as a starting point for SSCs to consider for fragility analysis is a rational starting point as the PRA models have already identified and modeled SSCs that cover all the critical safety functions and are appropriate for modeling in PRA core damage frequency (CDF) and release frequency models. Basic events in the PRA models are used as the vehicle to identify the starting list of SSCs and operator action pathways to walkdown.

The PRA model files used as input for the SEL development are the PBAPS full-power internal events PRA (which also includes internal flooding models) and the internal fires PRA. These models include both Level 1 (core damage frequency) and Level 2 (large early release frequency) full power PRA related equipment. These PRAs do not cover spent fuel pool related functions; this is acceptable as the PBAPS SPRA is for reactor core postulated accidents and not spent fuel pool accidents (this is consistent with Reference [2]).

In addition to internal flooding and internal fires, these PRA models used as input for the initial phases of the SEL development cover the following types of initiating events:

- Transients
- Loss of support systems (e.g., loss of DC bus, loss of AC bus, loss of instrument air, etc.)
- Loss of offsite power (LOOP)
- Loss of coolant accidents (LOCA) inside primary containment (including excessive LOCA)
- Interfacing Systems LOCAs (ISLOCA)
- Loss of coolant accidents outside primary containment (BOC)

All these initiated states are included in the PBAPS SPRA with seismic-induced SSC failures, with the exception of transients. Given the low capacity of offsite power, seismic-induced transients (i.e., offsite power remains intact) are not explicitly modeled in the SPRA as the plant likely would remain at power (not trip) or if a trip did occur the likelihood of seismic-induced failure of significant mitigation equipment is very low. As such, equipment on the initial SEL that is powered only from non-emergency AC power is screened from further consideration (except for SSCs that have the potential for secondary hazards or impacting operator action pathways).

The Very Small LOCA initiator is added to the SEL and included in the PBAPS SPRA model. The excessive LOCA is addressed by a fragility for the RPV supports. The RPV recirculation pumps were added to support fragility evaluation for seismic-induced Large LOCA. Failure to scram (ATWS) is addressed by fragility calculations of the RPV internals.

Initiating events for plant shutdown configurations (e.g., loss of SFP Cooling) are not covered by these models and this is consistent with the scope of this full-power PBAPS SPRA (and consistent with Reference [2]).

These PRA models also cover all the requisite Level 1 and Level 2 critical safety functions:

- Reactivity control
- Reactor pressure control
- Reactor coolant inventory control (including RPV depressurization)
- Containment pressure control (including vapor suppression)

- Primary and secondary containment isolation

The frontline systems modeled in the PBAPS SPRA as a function of critical safety function are summarized in Table 4.1.1-1. The support systems used in the PBAPS SPRA are not listed in Table 4.1.1-1. The support systems modeled in the PBAPS SPRA are (room cooling has been evaluated and is not required in the PRA for any of the frontline or support systems) [54]:

- Emergency AC (including EDGs)
- 125V and 250V safety DC
- Emergency Service Water (ESW)
- Emergency Cooling Water (ECW)
- Reactor Building Closed Cooling Water (RBCCW)
- High Pressure Service Water (HPSW)
- Pneumatic supplies (e.g., SGIG)
- Condensate Transfer (e.g., CST)

Table 4.1.1-1 PBAPS SPRA Frontline Systems per Safety Function [41]

Critical Safety Function	Systems (1),(2),(3)
Reactivity Control	RPS ARI RPT SLC
RPV Pressure Control	ADS and non-ADS SRVs RPT
RPV Coolant Inventory Control (High Pressure)	HPCI RCIC CRD
RPV Coolant Inventory Control (Low Pressure)	LPCI mode of RHR LPCS HPSW through RHR
RPV Depressurization	ADS and non-ADS SRVs
Containment Pressure and Temperature Control	Suppression pool cooling mode of RHR Containment Spray (DWS) Venting
Vapor Suppression	WW-DW Vacuum Breakers Drywell Spray mode of RHR SRVs
Containment Isolation	Primary Containment Isolation System and associated valves Primary containment structure Reactor building structure

Notes to Table 4.1.1-1:

1. Systems/functions reliant on auxiliary AC power for success are not credited in the PBAPS SPRA.
2. Support systems (e.g., electric power) are not listed in this table.
3. Some of the critical safety functions also are modeled with FLEX equipment. FLEX can supply emergency AC power to various functions and FLEX is used as an alternative injection system in the SPRA.

In addition to the initial development stages described above, the SEL development is supplemented by the following efforts:

- Review of system drawings to identify items not explicitly included in the PRA models
- Review of the internal flooding PRA to identify internal flooding sources of potential significance
- Review of plant drawings and Human Reliability Analysis to identify operator action pathways
- Identification of block walls in buildings containing SPRA equipment
- Identification of flammable sources (e.g., hydrogen, fuel oil, lube oil)
- Identification of potential seismic-induced electrical fire sources (including non-safety electrical, with the assumption that arcing may occur prior to loss offsite power)
- Component chatter assessment (separate topic discussed below)
- Identification of buildings of interest to SPRA
- Identification of above ground tanks
- Identification of buried items
- Plant walkdowns

Structures that house or spatially interact with identified SSCs, as well as those that involve ex-Control Room actions credited in the SPRA, are included in the SEL for fragility consideration. A disposition of all structures on the site is performed and documented in the SEL report. The following buildings and structures were identified for inclusion on the SEL (no earthen structures were identified for inclusion on the SEL):

- Drywell, Vents, Torus, and Penetrations (Primary Containment): Houses NSSS and key equipment in the SPRA. NSSS line items included separately on SEL for RCS piping (LOCAs) and RPV supports.
- Reactor Buildings: Houses key equipment in the SPRA (e.g., RHR pumps and heat exchangers).
- Reactor Vessel Support Pedestal: Houses RPV and control rods.
- Main Control Room Complex: Houses key plant equipment in the SPRA. The main control room and all of the safeguard AC and DC buses are in this area.
- Radwaste Building: Houses key plant equipment in the SPRA. The adjoining reactor auxiliary bay (which is considered part of the Radwaste building structure) houses the HPCI and RCIC turbine driven pumps. The Radwaste Building Complex and Main Control Room Complex are structurally connected to each other.
- Diesel Generator Building: Houses key plant equipment in the SPRA (i.e., the EDGs).
- Pump Structure (Seismic Class I Portion): Houses key plant equipment in the SPRA (i.e., ESW and HPSW pumps).

- Emergency Heat Sink Facility: Houses key plant equipment in the SPRA (i.e., ECW pump and Emergency Cooling Towers).
- Liquid N2 Tank Building: Houses key plant equipment in the SPRA (i.e., the CAD Tank).
- Turbine Building: The pipe tunnel portion houses piping and cables for key plant equipment in the SPRA (e.g., ESW and HPSW pumps). The normal egress for the Main Control Room is via Turbine Building Elevation 165'.
- Miscellaneous Switchyard Areas and related Switchgear Buildings: The switchyard and the miscellaneous outdoor switchgear structures are addressed by the "Offsite Power" line item on the SEL.
- Station Blackout Structure: Houses key plant equipment in the SPRA (i.e., SBO Line equipment to support alternate power from the Conowingo dam hydro-electric station).
- Conowingo Dam: The downstream dam supports maintaining adequate river level as the normal (i.e. ultimate) heat sink to support normal suction to the Circulation Water Pump Structure. Downstream hydro-electric station also provides an alternate AC power source to PBAPS.
- FLEX Equipment Building: Houses FLEX equipment.

The following buried items were identified for inclusion on the SEL:

- Buried HPSW piping
- Buried ESW piping
- Buried ECW piping
- Buried EDG Fuel Oil Transfer piping
- EDG Fuel Oil Storage tanks

Every cable tray, pipe and HVAC duct in the plant was not specifically itemized; the PBAPS SPRA used fragility walkdowns to search for outliers, to assess the ruggedness of these distributed systems, and to calculate fragilities in certain cases (e.g., safety piping in the reactor building).

In addition to the above, SSCs from the previous seismic related assessments were added to the PBAPS SEL for consideration:

- PBAPS Safe Shutdown Equipment List [35] [65]
- PBAPS NTTF 2.3 Seismic Walkdown Equipment List (SWEL) [37,38]
- PBAPS FLEX ESEL [36]
- PBAPS Initial Seismic PRA Model – An earlier "Phase I" seismic PRA performed for PBAPS in 2012 [49]. Initial "Phase I" seismic PRA models were developed for Exelon sites following the events at Fukushima in anticipation of potentially developing more detailed seismic PRA models in the future.

The total number of line items on the SEL is approximately 9000 and covers basic events, initiating events, operator actions, various basic event types, and specific pieces of equipment and structures. A disposition process of each line item is used to identify those line items that can be screened and those that are to be carried forward for SSC fragility evaluation. The following disposition codes are used to disposition the PBAPS SEL line items:

- S0a: Non-applicable initiating event to SPRA (e.g., Loss of Feedwater)
- S0b: Type A and B HEPs
- S0c: Type C dependent HEP (Type C independent HEPs already provide the necessary information on action pathways)
- S0d: Function Recovery and Repair basic events
- S0e: Test and maintenance basic events
- S0f: Common Cause Failure (CCF) basic events
- S0g: Flag basic events (i.e., PRA basic events set to TRUE or FALSE to model specific plant conditions)
- S0h: Other basic events that need not be carried forward in the SEL development process for the identification of SSCs (e.g., plant configuration probabilities, phenomena events).
- S0x: Additional Failure Mode basic events that can exist in the PRA models for a given SSC that need not be carried forward in the SEL development process.
- S1: SSCs not included in SPRA model
- S2: Post-initiator operator actions performed in Main Control Room (Main Control Room structure and control panels already included on SEL)
- S3a: Inherently rugged SSC
- S3b: Rugged SSC based on observation
- S4a: Subsumed into fragility component boundary – circuit breakers
- S4b: Subsumed into fragility component boundary – relays
- S4c: Subsumed into fragility component boundary – misc. instrument and control items
- S4d: Subsumed into fragility component boundary – rule of the box
- F1: SPRA post-operator actions performed outside Main Control Room (these define the operator action pathways that need to be investigated)
- F2: SSC requiring fragility evaluation
- F2-S3b: SSCs that were originally dispositioned as S3b (e.g., valves identified as rugged based on observations). The PB SPRA Fragility Team calculated fragilities for SSCs that need to change state (e.g., MOVs, AOVs).

The disposition codes beginning with the letter "S" indicate SEL line items that need not be carried forward for fragility calculations for the variety of reasons indicated (e.g., other line items already capture that SSC; or that line item is within the fragility component boundary of another line item on the SEL, etc.). The line items with the "S3b" disposition code were walked down to determine if they can be properly classified as rugged and not require a fragility evaluation. However, fragility evaluations were performed for approximately 400 SEL items with the "F2-S3b" disposition code covering both units where the SSC needs to actively change state (e.g., MOV or AOV needs to open or close to support the system mitigation suction in the PRA model). The SEL line items with the "F2" disposition code identify the SSCs requiring fragility evaluation. There are over 500 "F2" SSC line items (i.e., not including the "F2-S3b" line items) on the PBAPS SEL covering both units. Of the approximately 9000 line items on the SEL covering both units, fragility data was provided for over 900 SSCs associated with the SSCs with disposition code "F2-S3b" that are identified to need to change state and SSCs with disposition code "F2".

4.1.2 Relay Evaluation

During a seismic event, vibratory ground motion can cause relays to chatter. The chattering of relays potentially can result in spurious signals to equipment. Most relay chatter is either acceptable (does not impact the associated equipment), is self-correcting, or can be recovered by operator action. An extensive relay chatter evaluation was performed for the PBAPS SPRA, in accordance with SPID [2], Section 6.4.2 and American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA Standard, Section 5-2.2 [4] and is documented in reference [58]. Note that relay is used in sections of this report to mean relays as well as other contacts and contact devices that have the potential to chatter, including circuit breakers and motor starters. The term relay should be taken to mean any or all of these different electrical devices that are potentially sensitive to chatter. The evaluation resulted in most relay chatter scenarios screened from further evaluation based on no impact to component function. The relays, circuit breakers and other contact devices that were not screened are listed in Table 4.1.2-1, along with their function and disposition in the SPRA with appropriate seismic fragility or operator action.

The unscreened contact chatter scenarios provided in the contact chatter evaluation [58] (over 400) are considered and evaluated for inclusion in the SPRA model based on the identified system impact (e.g., divisional diesel fails to start or load). Given this high number of unscreened contact chatter scenarios, not all contact chatter scenarios are explicitly included in the SPRA model. Initial SPRA model quantifications helped identify the risk impact of individual or correlated contact chatter scenarios based on associated system impact and fragility value. Table 4.1.2-1 lists the contact chatter scenarios which were not screened via the chatter evaluation [58] and are explicitly modeled in the SPRA.

Table 4.1.2-1 Summary of Disposition of Unscreened Relays

Relay	Function	Disposition
150G ground fault relays	Chatter would render unavailable the 4 kV switchgear	**Modeled in SPRA** due to calculated risk impact based on initial quantification results. Modeled as seismic induced correlated relay chatter unavailability of all eight (8) 4 kV switchgears based on the limiting fragility for all (8) 4 kV switchgears. Credit for potential operator recovery of the seismic induced relay chatter event is based on insights from plant specific operator interviews and detailed Human Reliability Analysis (HRA).
127X undervoltage relays	Diesel generator loading may be out of sequence	**Modeled in SPRA** due to calculated risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of the four (4) EDGs. Credit for potential operator recovery of the seismic induced relay chatter event is based on insights from plant specific operator interviews and detailed Human Reliability Analysis (HRA).
151N neutral overcurrent relays	Divisional diesel unavailable	**Modeled in SPRA** due to calculated risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of individual EDGs. Credit for potential operator recovery of the seismic induced relay chatter event is based on insights from plant specific operator interviews and detailed Human Reliability Analysis (HRA).
TD5, 5, TD3, CC1, EOSX12, ESR SDR, TD4, TD2, CP1, CT1, FP1, IP1, OP1, OT1, SFR, PE2 protective relays	Divisional diesel unavailable	**TD2 and TD5 relays modeled in SPRA** due to calculated risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of the four (4) EDGs. No credit for potential operator recovery of the seismic induced relay chatter event is assumed based on the estimated time required to perform a local operator action.

Table 4.1.2-1 Summary of Disposition of Unscreened Relays

Relay	Function	Disposition
3-23A-K035 HPCI STEAM LINE HIGH DIFFERENTIAL PRESSURE RELAY	HPCI auto isolation	**Modeled in SPRA** due to calculated risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of HPCI due to auto isolation. Over one hour will be required to recover HPCI operation. Limited credit for potential operator recovery of the seismic induced relay chatter event is based on insights from plant specific operator interviews (e.g., time required to recover HPCI) and detailed Human Reliability Analysis (HRA).
2-23A-K027 HPCI - AUTO ISOLATION RELAY	HPCI fails to inject – Inboard steam supply valve spuriously closes	**Modeled in SPRA** due to calculated risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of HPCI due to spurious valve closure. Recovery of HPCI operation is not likely. The SPRA assumes no credit for operator recovery.
150/151 time-phased overcurrent relays	Interruption of diesel sequencing and potential diesel overload	**Modeled in SPRA** due to calculated risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of the four (4) EDGs. Credit for potential operator recovery of the seismic induced relay chatter event is based on insights from plant specific operator interviews and detailed Human Reliability Analysis (HRA).
SI-Overcurrent relays	Lockout of divisional switchgear	**Modeled in SPRA** due to calculated risk impact based on initial quantification results. Modeled as seismic induced correlated relay chatter unavailability of all eight (8) 4 kV switchgears based on the limiting fragility for all (8) 4 kV switchgears. Operator recovery assumed unlikely and not credited due to lockout of divisional switchgears.

Table 4.1.2-1 Summary of Disposition of Unscreened Relays

Relay	Function	Disposition
3-13A-K033 RCICS-STEAM LINE HIGH DIFFERENTIAL PRESSURE - STEAM LINE BREAK	RCIC automatic isolation	**Modeled in SPRA** due to calculated risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of RCIC due to auto isolation. Over one hour will be required to recover RCIC operation. Limited credit for potential operator recovery of the seismic induced relay chatter event is based on insights from plant specific operator interviews (e.g., time required to recover RCIC) and detailed Human Reliability Analysis (HRA).
2-13A-K012 RCICS - AUTO ISOLATION SIGNAL, CONTROL RELAY	RCIC fails to inject – Inboard steam supply valve spuriously closes	**Modeled in SPRA** due to calculated risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of RCIC due to spurious valve closure. Recovery of RCIC operation is not likely. The SPRA assumes no credit for operator recovery.
0-33-102-1706 ESW 'B' PUMP DIESEL LOAD SEQUENCE TIME DELAY	Significant overloads could cause the EDG to stall	**Modeled in SPRA** due to calculated risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of EDG C. No credit for potential operator recovery of the seismic induced relay chatter event is assumed based potential exceedance of the EDG load rating.
0-33-163-1603 ESW 'A' PUMP START ON LOSS OF ESW 'B' PUMP DISCHARGE PRESSURE	Significant overloads could cause the EDG to stall	**Modeled in SPRA** due to calculated risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of EDG B and C. No credit for potential operator recovery of the seismic induced relay chatter event is assumed based potential exceedance of the EDG load rating.
0-33-163-1706 ESW 'B' PUMP START ON LOSS OF ESW 'A' PUMP DISCHARGE PRESSURE	Significant overloads could cause the EDG to stall	**Modeled in SPRA** due to calculated risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of EDG B and C. No credit for potential operator recovery of the seismic induced relay chatter event is assumed based potential exceedance of the EDG load rating.

Table 4.1.2-1 Summary of Disposition of Unscreened Relays

Relay	Function	Disposition
MO-3-23-015 Steam Line to HPCI Turbine Inboard Isolation Valve Motor Starter	Chatter of the starter seal-in contact may cause the valve to spuriously close and stop HPCI. AC-power would be needed to re-open the valve.	**Modeled in SPRA** due to calculated risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of HPCI due to potential damage to the HPCI pump. The SPRA assumes no credit for operator recovery.
MO-3-13-015 Steam Line to RCIC Turbine Inboard Isolation Valve Motor Starter	Chatter of the starter seal-in contact may cause the valve to spuriously close and stop RCIC. AC-power would be needed to re-open the valve.	**Modeled in SPRA** due to calculated risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of RCIC due to potential damage to the RCIC pump. The SPRA assumes no credit for operator recovery.
MO-2-13-015 Steam Line to RCIC Turbine Inboard Isolation Valve Motor Starter	Chatter of the starter seal-in contact may cause the valve to spuriously close and stop RCIC. AC-power would be needed to re-open the valve.	**Modeled in SPRA** due to calculated risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of RCIC due to potential damage to the RCIC pump. The SPRA assumes no credit for operator recovery.

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]. Walkdowns of those SSCs included on the seismic equipment list were performed to assess the as-installed condition of these SSCs for use in determining their seismic capacity and performing initial screening.

Several previous seismic walkdowns for PBAPS have been documented. The information gathered during these previous walkdowns and the results and conclusions contained in the walkdown information was used where applicable to supplement plant drawings and calculations and to reduce the scope of walkdowns performed specifically for the SPRA as discussed in this report. These previous walkdowns include:

- SQUG/IPEEE – Performed in 1995-97 time frame in support of the Individual Plant Examination of External Events (IPEEE) and in response to Unresolved Safety Issue (USI) A46, using the methodology developed by the Seismic Qualification Utility Group (SQUG) and contained in the SQUG Generic Implementation Procedure (GIP) [30] and the guidelines contained in EPRI NP-6041-SL [14].
- NTTF 2.3, seismic – Performed in response to Near-Term Task Force (NTTF) Recommendation 2.3, Seismic. This walkdown was completed in late 2012 [37][38].
- ESEP – Performed during 2014 and 2015 in support of the Expedited Seismic Evaluation Process (ESEP) [36].
- Seismic PRA – Performed to develop input to an earlier “Phase I” seismic PRA performed for PBAPS in 2012 [49]. Initial “Phase I” seismic PRA models were developed for Exelon sites following the events at Fukushima in anticipation of potentially developing more detailed seismic PRA models in the future.

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 walked down previously, all components on the SEL, including those walked down previously were included in the scope of the current SPRA walkdowns. However, for components which had been walked down previously and for which sufficient information was available to permit development of a fragility, the walkdown was limited to a walk-by of the individual components.

Detailed walkdowns were performed for all components which had not been walked down previously. During a detailed walkdown, the caveats from the SQUG

GIP [30] were verified and sufficient information was gathered to allow a fragility to be developed. 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 which had been walked down previously. During walk-bys, the walkdown team inspected the component to ensure that there were no obvious changes to the component since the previous walkdown that would adversely impact the seismic capacity of the component. In particular, the walkdown team focused on potential seismic interaction concerns and conditions that might adversely impact the component. In general, walk-bys were less detailed and less intrusive than walkdowns.

The walkdowns were performed in accordance with Table 6.5 of the SPID [2]. Information contained in the SQUG GIP [30] and EPRI NP-6041 [14] was used to supplement the guidance provided in the SPID. The SPRA walkdown meets or exceeds the requirements for a Capability Category 2 SPRA established in the current ASME/ANS risk assessment standard updated through ASME/ANS RA-Sa-2009 [4]. This standard is endorsed by the United States Nuclear Regulatory Commission (USNRC) Regulatory Guide (RG) 1.200, Revision 2 [15], for seismic risk analysis.

During the course of the seismic PRA, a number of different walkdowns were performed. These included a plant familiarization walkdown and walkdowns or walk-bys of all components on the SEL. 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 Diverse and Flexible Coping Strategies (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 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.

During the walkdowns, the walkdown team focused on seismic issues that could potentially affect the assignment of a seismic capacity to individual components. This included anchorage details, compliance with the caveats contained in the SQUG GIP [30] associated with each equipment class, seismic interaction due to falling or displacement, existence of block walls in proximity to the components and potential for seismically induced flood and fire. Walkdown documentation for equipment and structures consisted of noting the existing conditions, taking photographs and recording findings, if any.

4.2.1 Significant Walkdown Results and Insights

Consistent with the guidance from NP-6041-SL [14], no significant findings were noted during the PBAPS seismic walkdowns. Observations make during the walkdowns are documented in the walkdown reports.

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

4.2.2 Seismic Equipment List and Seismic Walkdowns Technical Adequacy

The PBAPS SPRA SEL development and walkdowns were subjected to an independent peer review against the pertinent requirements (i.e., the relevant SFR and SPR requirements) in the PRA Standard [4]. The peer review was performed relative to Capability Category II for the full set of requirements in the PRA Standard.

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

4.3.1 Fixed-base Analyses

Although PBAPS is a hard rock site, fixed-base analyses were not performed, i.e., 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 SSI models [16], [17], [18], [19].

4.3.2 Soil Structure Interaction (SSI) Analyses

SSI analyses considering ground motion incoherence were performed for the Reactor Building (RB) Complex (which includes the Reactor Buildings, Turbine Building, and Radwaste Building and Main Control Room), Diesel Generator Building (DGB), Emergency Cooling Tower (ECT), and Pump Structure (PS). For each, structural and soil properties are defined consistent with their response at a representative acceleration hazard range of interest selected via coordination with fragility and PRA analysts. This hazard range of interest was selected to be the GMRS level based on insights from incremental risk quantifications, especially regarding the relative risk-significance of different acceleration intervals and individual components.

A baseline set of SSI analyses with the assumption of cracked concrete elements (i.e., simulated with reduced stiffness and increased damping per ASCE 43-05 [22]) was originally performed for all the structures. Subsequently, a second, supplemental set of SSI analyses with the assumption of uncracked concrete was performed for the RB complex, DGB, and PS. Structural response results from both sets of analyses were considered during component fragility analysis with the intent that the seismic demands of risk-significant components should be reasonably consistent with their failure levels.

The RB Complex, ECT, and PS SSI analyses consider structural property variation via use of best estimate (BE), lower bound (LB), and upper bound (UB) structure models. This method for consideration of structural property variation was also used for the DGB supplemental SSI analyses. For the DGB baseline SSI analyses, structural frequency variation is considered by peak broadening. The RB Complex, ECT, and PS are founded on hard rock (Vs = 9285 fps) so soil variability is not considered; ground motion variability is considered via use of five independent sets of time histories. One set of time histories was provided as part of the PSHA [6] and the remaining four were developed in [20]. The DGB is founded on and within soil, with soil variability considered via use of separate BE, LB, and UB soil cases where the soil properties for each layer of each case is defined based on the results of the probabilistic site response analysis performed with the PSHA [6].

The SC-SASSI analysis code [31] was used to perform the SSI analyses. Cutoff frequency for the SSI analyses was chosen to be 50 Hz, and the SSI models were sufficiently refined to transmit frequencies up to at least 50 Hz through the soil/rock-foundation interface. All SSI analyses utilized the SASSI Direct Method and the analyses in the three spatial directions are performed simultaneously.

A list of structures and description of relevant parameters is listed in Table 4.3-1.

SSI analysis documentation can be found in references [16], [17], [18], [19].

4.3.3 Structure Response Models

Seismic models were developed based on industry codes and standards (ASCE 4-98 [21] and ASCE 43-05 [22]) to obtain median-centered response analyses including SSI effects at the GMRS hazard level. Models were sufficiently refined to capture building torsion effects, out-of-plane floor response, and in-plane floor diaphragm stiffness. Mass sources included self-weight, equipment, distributive systems, and seismic live load. Concrete and steel material properties were building-specific and based on plant data. The existing (e.g., design basis, IPEEE) structural models were evaluated against the SPID [2] Section 6.3.1 requirements for structural modeling. Furthermore, they were evaluated to determine whether generating a new model would be beneficial for fragility considerations given low-capacity and/or risk-significant components or structures.

For the RB Complex, it was determined that the existing lumped mass stick models (LMSM) of the individual structures did not meet the SPID criteria [2] and could not reasonably capture local responses that could be important to component fragilities, given the relative complexity of the structural configurations. Furthermore, the individual structures shared a common large foundation, and had partially shared load paths between individual buildings. Therefore, a new, combined RB Complex detailed 3D finite element (FE) model was generated to consider structural interfaces and the large common foundation beneath the entire complex. In the baseline analyses which considered cracked concrete, structural damping was varied; for concrete, the damping value used (as a percentage of critical damping) was 7% for LB (paired with UB structural stiffness), and then varied according to a Coefficient of Variation (COV) of 0.35 for BE and UB properties. For the supplemental analyses considering uncracked concrete, concrete damping was held constant across the models (while stiffness was varied), with 4% of critical damping assigned for concrete elements.

For the DGB superstructure, it was determined that although the existing LMSM did not directly satisfy the SPID criteria [2], relatively minor enhancements (e.g., addition of oscillators for capturing floor response and outriggers for response at building corners) were feasible to upgrade the existing model to a satisfactory level. A check was performed to confirm that the appropriate level of discretization existed in order to sufficiently capture the effect of higher modes for the superstructure LMSM. However, since the structure (a) is founded on and within soil, (b) has a relatively complex foundation system, and (c) supports most credited equipment from the ground floor, the seismic demand on the credited equipment is governed by soil and foundation response. Therefore, a new FE model of the foundation system (slab, shear walls and bearing piles) was developed to consider the effects of embedment and incoherency, with the superstructure LMSM attached to the slab at grade. In the baseline analyses which considered cracked concrete, 10% of critical damping was assigned for concrete elements, and in supplemental analyses considering uncracked concrete, 4% of critical damping was assigned for concrete elements.

For the ECT, preliminary review identified the structure as potentially low-capacity and risk-significant as compared to other plant structures. The governing failure mode was identified as the soft-story columns and it was determined that the expected low fragility was at least partially a result of over-conservative force distribution resulting from the existing LMSM. It was also understood that the ECT is a redundant system credited for decay heat removal only in the scenario that the normal (i.e. ultimate) heat sink is lost, such as in the event of the failure of the downstream Conowingo Dam. Because this dam was potentially risk-significant, credit for the ECT was believed to be a potentially important scenario to realistically consider in the risk assessment. Therefore, a detailed 3D FE model was developed in order to reduce potential conservatisms in structural fragilities developed from a LMSM. For concrete, the damping values used (as a percentage of critical damping) was 7% for LB (paired with UB structural stiffness), and then varied according to a COV of 0.35 for BE and UB properties. A supplemental analysis assuming uncracked properties was not performed for this structure due to the fact that significant cracking of this structure is expected at the GMRS level and due to the fact that the structure contains relatively few components that are potentially risk significant.

For the PS, it was determined that although the existing LMSM did not directly satisfy the SPID criteria [2], relatively minor enhancements (e.g., addition of oscillators for capturing floor response, outriggers for response at building corners, and additional discretization of the LMSM) were feasible to upgrade the existing model to a satisfactory level. Therefore, the existing LMSM was enhanced and connected to a flat foundation slab for SSI analyses. The structure was analyzed as surface-founded, and consideration was given to the foundation configuration in an uncertainty quantification study that assessed the possible range of response differences caused by unbalanced embedment effects. In the baseline analyses which considered cracked concrete, structural damping was varied; for concrete, the damping values used (as a percentage of critical damping) was 7% for LB (paired with UB structural stiffness), and then varied according to a COV of 0.35 for BE and UB properties. For the supplemental analyses considering uncracked concrete, concrete damping was held constant across the models (while stiffness was varied), with 4% of critical damping assigned for concrete elements.

Structural model verification was performed by comparing the total mass and fixed-base fundamental frequencies to the existing LMSMs, as well as performing static analyses considering 1g acceleration forces in the vertical and two horizontal directions to confirm reasonable structural behavior. SSI model verification was performed by mass comparison and careful review of transfer functions in all directions and all structure/soil cases. Transfer function review included, for example, confirmation that low frequency response approached 1.0 for on-axis directions and 0.0 for off-axis directions, reasonableness of amplification with increased building elevation, and comparison of resonant peaks to fixed-base frequency analyses of the structure and site response analyses of the soil column.

Following the frequency-domain SSI analyses, the in-structure response spectra (ISRS) for structures considered in the seismic PRA were developed using spectrally matched (to the GMRS) time-histories. Both horizontal and vertical ISRS were computed from the time-history motions at various floor levels and other important locations. Selection of the locations at which response was calculated was based on equipment location within the buildings, with node specificity a function of component risk-significance. For the baseline ISRS, small plant areas / rooms were defined to capture each component location, and the responses at representative nodes within each area were included in the response at that area. For the supplemental ISRS, component-specific responses at the equipment footprint and/or anchor points were provided.

The ISRS were calculated in the frequency range of 0.1 Hz to 100 Hz and are the algebraic sum of the response obtained for each of the three directions of input ground motion. For the RB Complex and PS supplemental ISRS, highly amplified narrow frequency content was clipped for comparison to broad-banded test response spectra typical of most nuclear power plant (NPP) components. Two separate methods based on guidance provided in (1) EPRI TR-103959 [23] and (2) EPRI NP-6041-SL [14] were used for the peak-clipping process in the supplemental spectra.

For the PBAPS dynamic analyses, both median (~50th%) and conservative (~80th%) estimates of ISRS were developed from a series of structural response analyses which separately considered variability in structural properties, soil properties, and ground motion characteristics. The separate analysis cases were combined to capture the collective effect of such independent variabilities on the median and conservative response. For the RB Complex, ECT, and PS, a multi-case deterministic approach was used where the structural properties (frequency and damping) were varied. Since these structures are founded on hard rock, soil variability was not considered. For the baseline analyses, the BE case was analyzed using five time-histories, and the LB and UB cases were analyzed with the single time history that best represented the middle of the five BE responses. The median ISRS were developed by averaging the response from the individual cases, and the conservative ISRS were developed by enveloping the individual responses. For the supplemental analyses a single time-history (best representing the median) was used, and the average/envelope methodology for median/conservative response was maintained. For the DGB baseline analyses, the structural properties were not varied, but the soil properties were, resulting in similar BE, LB, and UB analyses. These were analyzed with a single time-history developed in the PSHA for the DGB-specific FIRS. Peak broadening of ±15% was included for the BE-soil case in order to consider structural frequency variability, and then the BE, LB, and UB cases were averaged/enveloped to obtain median/conservative estimates of ISRS. For the DGB supplemental analyses, structural properties were varied in addition to soil properties, resulting in five SSI analysis cases: BE_{soil}-BE_{struc}, LB_{soil}-BE_{struc}, UB_{soil}-BE_{struc}, BE_{soil}-LB_{struc}, BE_{soil}-UB_{struc} which were analyzed with the same single time-history as used in the baseline analyses. Averaging/enveloping was performed as discussed previously to obtain median/conservative estimates of ISRS.

The FLEX storage building is a single-story concrete shear wall structure that is located at the ground surface and founded on piles that extend to the hard reference rock. It is used to house FLEX components that are stored in the building. These components are not in operation while they are stored in this structure and are not permanently anchored to the floor. Building responses were therefore not calculated for this structure. Since the building is surface mounted, FIRS 2 was used as the input to evaluate the fragility of SSCs within the FLEX storage building.

Table 4.3-1 summarizes the type of analysis and model used for each of the major structures modeled in the SPRA. Unless otherwise specified, the same approach was used for both the baseline and supplemental analyses for each structure.

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

Structure	Foundation Condition	Type of Model	Analysis Method	Comments/Other Information
Reactor Buildings, Turbine Building, Radwaste Building and Main Control Room Complex	Rock	Combined FE model	Multi-case Deterministic SSI	Shear Wave velocity = 9285 ft/sec; SSI analysis performed with incoherence, 3 structure cases used, 5 time-histories (T-H) for BE
Diesel Generator Building	Foundation shear walls and bearing piles embedded in ~20 ft. of soil down to rock	LMSM superstructure with FE foundation	Baseline: Deterministic SSI	SSI analyses performed with incoherence, 3 soil cases used, peak-broadening for BE case
			Supplemental: Multi-case Deterministic SSI	SSI analyses performed with incoherence, 5 cases (BE_{soil}-BE_{struc}, LB_{soil}-BE_{struc}, UB_{soil}-BE_{struc}, BE_{soil}-LB_{struc}, BE_{soil}-UB_{struc})
Emergency Cooing Tower	Rock	FE	Multi-case Deterministic SSI	Shear Wave velocity = 9285 ft/sec; SSI analysis performed with incoherence, 3 structure cases used, 5 T-H for BE
Pump Structure	Rock	LMSM superstructure with representative FE foundation	Multi-case Deterministic SSI	Shear Wave velocity = 9285 ft/sec; SSI analysis performed with incoherence, 3 structure cases used, 5 T-H for BE. Uncertainty quantification for embedment condition.
FLEX Storage Building	Surface founded on piles down to rock	N/A	N/A	Building response not calculated. FIRS2 used for response.

4.3.4 Seismic Structure Response Analysis Technical Adequacy

The PBAPS SPRA Seismic Structure Response and Soil Structure Interaction Analysis were subjected to an independent peer review against the pertinent requirements in the PRA Standard [4]. The peer review was performed relative to Capability Category II for the full set of requirements in the PRA Standard [4].

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

4.4 SSC Fragility Analysis

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

This section summarizes the fragility analysis methodology, presents a tabulation of the fragilities (with appropriate parameters *(e.g.,* A_m, βr, βu*)*, and the calculation method and failure modes) for those SSCs determined to be sufficiently risk important, based on the final SPRA quantification (as summarized in Section 5). Important assumptions and important sources of uncertainty, and any particular fragility-related insights identified, are also discussed.

4.4.1 SSC Screening Approach

The Seismic PRA approach used at PBAPS initially utilized three quantifications in support of the Near-Term Task Force (NTTF) Recommendation 2.1, Seismic submittal to the NRC. In addition to these formal quantifications, various sensitivity studies were performed during the course of 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 in an attempt to improve their calculated fragilities in order to reduce their risk significance. This approach has been successfully implemented at several plants and is in compliance with the ASME Standard [4] and the SPID [2]. All three quantifications and numerous sensitivity studies were performed prior to the peer review. Subsequent to the peer review and in an effort to address peer review comments, two additional quantifications were performed. After each quantification, the results were reviewed to determine if additional insights were obtained and to determine if further refinement of fragilities associated with top risk contributors would improve the results and yield a more realistic model.

For the first quantification, a representative fragility was provided for all items on the SEL that were identified to require a fragility calculation (e.g., SSCs with disposition code “S3b” that are identified to need to change state and SSCs with disposition code “F2”) as discussed in Section 4.1.1. Representative fragilities were site-specific and were based on scaling existing calculations and/or performing simplified analyses. This included fragility data for approximately 900 SSCs, including structures and components. In addition, fragilities were provided for approximately 370 relay chatter scenarios. The PBAPS Fragility Team did not screen any components from the SEL based on judged seismic capacity. The only items removed from the preliminary SEL were items identified as inherently rugged. This included manual valves, check valves, hand switches, reset pushbuttons, cables and other items per available industry guidance (e.g. 2013 EPRI SPRA Implementation Guide [11]) as permitted by the SPID [2].

After the first quantification, the Fussell-Vesely Importance measure (FV) was provided for each of the components on the SEL that are included in the PRA model. The top risk contributors were determined to be those with the highest FV numbers. The FV number is an estimate of the amount that either the Seismic Core Damage Frequency (SCDF) or the Seismic Large Early Release Frequency (SLERF) would improve if the fragility of the component were improved to a point where seismic failure for the component would not occur (failure rate = 0). In addition to the use of the FV ranking to identify SSCs for further fragility refinement, a challenge meeting was held to assess the SSCs on the list and the scenarios that led to their relative importance. This challenge meeting included representatives from site engineering, and the fragility, structural modeling and PRA teams. Each SSC that was determined to be a high-risk contributor was discussed to ensure that the scenarios included in the PRA model were accurate and appropriate and reflected actual plant operation. The reported FV numbers from the first quantification along with results of various sensitivity studies and the inputs from the challenge meeting were used to develop a list of components for which enhanced fragilities would be developed for input to the second quantification. Both SCDF and SLERF were considered and, to obtain a sufficiently large list of components for which enhanced fragilities would be provided, a cutoff threshold FV value of 1E-06 was used.

Enhanced fragilities were developed for all items determined to be high-risk components with the exception of items such as Loss of Offsite Power (LOOP) for which there was no basis to improve fragilities. The second quantification incorporated the enhanced fragilities developed using detailed Conservative Deterministic Failure Margin (CDFM) calculations. For the second quantification:

- Approximately 100 SSC fragilities (excluding relays) were updated with enhanced fragility values.
- Approximately 280 relay fragilities were updated.

For the third quantification, SSCs that were dominant risk contributors were identified based on the risk insights from the second quantification and associated sensitivity studies. The process was the same as the process used following the first quantification. This process was used to develop a list of components for which detailed fragilities would be calculated for input to the third quantification.

For the third quantification, “detailed” fragilities were developed for the dominant risk contributors using the Separation of Variables (SOV) approach to directly calculate the Median fragility. Approximately 40 SSC fragilities and 180 relay chatter fragilities were updated with detailed fragilities. The third quantification incorporated the detailed fragilities developed using the SOV calculations. Again, dominant risk contributors for which there was no basis to improve fragilities such as LOOP were not improved. Note that in some cases, where justifiable, fragilities developed for the third quantification were generated by further refining the CDFM calculations used to develop fragilities for the second quantification. Thus, the final quantification included detailed fragilities based on the SOV approach for dominant risk contributors, enhanced fragilities based on the CDFM approach for high-risk contributors and representative fragilities for low-risk contributors as well as generic fragilities for items such as LOOP.

After each quantification, sensitivity studies were performed to determine the relative importance of specific items, to determine the result of improvements in fragilities and to determine if improving the fragility of items such as LOOP would lead to additional risk insights. The results of the sensitivity studies were used to help identify high-risk and dominant risk components and to determine the approach to be used to refine the fragility of specific components.

As stated, subsequent to the peer review, additional quantifications of the SPRA model were performed. These quantifications used additional improved component fragilities as well as inclusion of FLEX and additional operator actions and improved human reliability estimates. The process used to determine components for which improved fragilities would be provided was the same as the process used after the first and second quantifications.

4.4.2 SSC Fragility Analysis Methodology

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

For the PBAPS SPRA, the computation of seismic fragilities for SSCs included in the SPRA is performed in accordance with or consistent with the requirements in ASME/ANS RA-Sb-2013 [4] and is based on the methodology provided in various EPRI guidelines. Specifically, the strategy for developing the fragilities for the complete set of SSCs on the SPRA SEL follows the recommendations of EPRI NP-6041-SL [14], EPRI-1019200 [24], and EPRI-103959 [23]. As mentioned in Section 4.4.1, the computation starts with “representative” fragilities in the first quantification and proceeds progressively to more detailed fragilities in subsequent quantifications as top contributors to risk are identified.

For the first quantification, site specific representative fragilities (referred to as ‘representative’ throughout) were typically developed by scaling existing design basis calculations to account for available margins in the design. This is the margin between allowable values associated with design requirements and values associated with High Confidence of a Low Probability of Failure (HCLPF) evaluations. These margins were used to develop a Safety Factor which is anchored to the GMRS to estimate a HCLPF. The generic values of aleatory variability and epistemic uncertainty from the SPID [2] 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. As discussed in Section 4.4.1, the top risk contributors were determined based on the FV numbers from the initial quantification and subsequent sensitivity studies. The quantifications and sensitivity studies are described in detail in the PRA notebook [45]. The fragilities were calculated using the CDFM method to determine the HCLPF. The generic uncertainty values, as recommended in Table 6.2 of the SPID [2] 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 from the supplemental analyses at the location of the individual components.

Fragilities for the third quantification were developed for the dominant risk contributors as identified during the second SPRA quantification. When beneficial, the fragilities for the final quantification were computed using the SOV approach where the median capacity is calculated directly, and a specific set of uncertainties are computed for each SSC. ISRS from the supplemental analyses at the location of the individual components were used as inputs. The number of components with detailed fragilities (SOV calculations) is consistent with the requirements of the ASME Standard [4] and the SPID [2].

4.4.2.1 Structures Fragility

The seismic fragilities for the following structures were determined:

- Diesel Generator Building (DGB)
- Radwaste / Main Control Room Complex
- Turbine Building Sub-Structure and Super-Structure
- Emergency Cooling Tower
- Reactor Building
- Pump Structure
- FLEX Building
- Conowingo Dam

Site specific representative structural fragilities were developed using a scaling approach. This approach identifies a scale factor on the effective structure input motion necessary for the median seismic demand to reach median seismic capacity from previous estimates. The scaling is done based on a weighted contribution at the spectral frequencies of active structure response modes, or ZPA response at different floor elevations, and is based on results from the response as discussed in the Fragility Report [25]. Reference-level seismic demand is the PBAPS GMRS, which is equivalent to the FIRS used for SSI analysis of the structures. A simplified structural fragility for the FLEX Building was developed based on the appropriate FIRS using a simplified approach taking input from an existing FLEX Building calculation [63]. A simplified structural fragility for the Conowingo Dam was developed by an analysis of the controlling structural feature of the dam determined from a review of existing analysis and drawings. Because none of these structures were found to be top contributors to seismic risk, more detailed fragilities are not required.

4.4.2.2 Component Fragility

For the first quantification, representative fragilities were typically developed by scaling existing design basis calculations or calculations performed during the SQUG/IPEEE effort to account for available margins in the design. The design margins were used to develop the Safety Factor which is anchored to the PGA of the GMRS to estimate a HCLPF. Generic aleatory variability and epistemic uncertainty values were assigned to obtain the Median fragility values as discussed in the SPID [2]. For special cases such as the Nuclear Steam Supply System (NSSS), a unique set of uncertainties was considered.

In general, the seismic evaluation of SSCs consists of two parts:

- Seismic Response Evaluation
- Seismic Capacity Assessment

The capacity of the SSC is defined with four (4) major failure modes:

- Functional failure
- Structural failure including anchorage failure (structure integrity)
- Nearby structural or spatial interactions (II/I)
- Seismically induced flood / spray interactions

In addition to developing representative fragilities for components, representative fragilities were developed for block walls and other items which could cause adverse seismic interactions. In addition, more detailed fragilities were developed for specific items such as the Condensate Storage Tanks (CST) and NSSS components. More detailed calculations were performed for these items because there were no existing calculations or because the existing calculations could not be scaled due to lack of information or lack of rigor and because these items were known to be relatively important to seismic risk.

Fragilities for the second quantification were in general developed using the CDFM approach. Instead of scaling existing analysis or using a simplified approach, detailed calculations were performed for each item using item specific information. The information needed to perform the evaluations was obtained from existing plant documentation or information gathered during the various walkdowns. While existing calculations were not scaled in the development of

CDFM calculations, information and insights such as controlling failure mode were obtained from these calculations.

Fragilities for the third quantification were in general developed using the SOV approach. Detailed SOV calculations, or in some cases more detailed CDFM calculations, were performed for each item using item specific information. Where needed, additional walkdown information was gathered to obtain more detailed input on items such as nozzle loads and location of center of gravity. Based on the results of the second quantification and the sensitivity studies, approximately 40 items were selected for further refinement. In addition, refined fragilities were developed for approximately 180 relays. The fragilities for these selected components were refined using the CDFM and/or SOV approach as judged beneficial. The number of SOV calculations performed meets the requirements of the ASME Standard [4] and the SPID [2].

Seismic demand was determined based on in-structure response spectra (ISRS) developed for the SPRA and in-cabinet amplification factors. The amplification factors for evaluating relays were determined by adjusting the generic amplification factors from EPRI NP-6041 [14] as applicable to account for exact location and orientation within the cabinets and the construction of the cabinets.

The component capacity was obtained from either Generic Equipment Ruggedness Spectra (GERS), EPRI NP-6041 [14], the EPRI high-frequency study, or other test reports. Appropriate knock-down factors and clipping per EPRI guidance (NP-6041-SL [14] for CDFM method or TR-103959 [23] for SOV method) was used as applicable for each scenario. Consistent with the guidance, clipping was only performed when comparing to broad-band test spectra.

Subsequent to the peer review, additional quantifications were performed to further refine the SPRA model and to respond to peer review comments. These quantifications are described in Section 5 of this report. To support these quantifications, additional refined fragilities were developed using either the CDFM or SOV approach as appropriate. Table 4.4.2.2-1 provides a summary of the number of components for which fragilities were developed for each quantification. Note that the number of SSCs included in the SPRA model was not reduced to the numbers shown in this table for the second quantification onward. Fragilities that were not improved were carried over from one quantification to the next. Also note that in some cases, refined fragilities were provided for certain SSCs for use in various sensitivity studies. In some cases, these refined fragilities were developed based on estimates and maximum potential improvements in order to determine the impact and benefit of developing more detailed fragilities for these items.

Table 4.4.2.2-1 Approximate Numbers of Refined SSC and Relay Fragilities for Each Risk Quantification

Quantification	Count of SSC (non-relay)	Count of Relays	Notes
Q1	900	370	Representative
Q2	100	280	CDFM: 79 components SOV: 16 components Relays: All CDFM
Q3	40	180	Representative: 4 components CDFM: 9 components, SOV: 29 components Relays: All SOV
Post Peer Review	40	360	CDFM: 36 components, SOV: 2 components Relays: All SOV

The following sections of the report provide additional detail on the methodology used to develop component fragilities. In general, the methodology described was applied to all three approaches used to develop fragilities (representative, CDFM and SOV). When the methodology differed from one approach to the other, the difference is discussed.

Use of Structural Response Results

The seismic demand for the component fragilities is the ISRS for the locations where the component is installed. The ISRS developed using the baseline (fully cracked) analyses were used to develop representative fragilities for the initial quantification. Based on the results of the initial quantification it was determined that the ISRS for subsequent quantifications should be based on the supplemental (un-cracked) analyses. Following the peer review, a sensitivity study was performed to determine the impact on the SPRA results if a higher earthquake level that would result in a fully cracked structure were considered. The results of this sensitivity study are discussed in section 5.7 of this report.

Frequency Range of Interest for Development of Component Fragilities

For SSC fragility development, the frequency of the SEL component was estimated or calculated using available industry information such as EPRI NP-6041-SL [14], the SQUG GIP [30] or other available documentation, including existing calculations or test reports. The natural frequency was varied +/- 20% if the ISRS was developed from a stick model and +/- 10% if it was developed from a finite element model. This frequency range was used to calculate the anchorage forces and moments to evaluate the load path and the anchorage.

For functional capacity, the natural frequency of the component was determined using the same approach as for the anchorage assessment. Then, the following process was followed:

- Regardless of the natural frequency of the component, the frequency of internal components located within the component could have a different frequency. Based on testing, it has been determined that the horizontal frequency range of interest for typical components and sub-components in nuclear power plants is in the range of 4 to 20 Hz unless the component frequency is known to be above 20 Hz. EPRI high-frequency testing [32] has shown that there are no components that are sensitive to seismic motion above 20 Hz that are not also sensitive to the same motion below 20 Hz. Thus, the frequency range of interest is taken as 4 Hz to 20 Hz unless available information indicates that a different frequency range should be considered.
- For components whose frequency is known to be above 20 Hz., the acceleration at the ZPA was typically used. Based on industry consensus, peaks in the response spectra that occur beyond about 20 Hz are not typically considered. However, accelerations that are obtained by linear interpolation between the acceleration that occurs at 20 Hz. and the acceleration at the ZPA were considered in cases where the fragility could be affected by accelerations at these frequencies.
- If the internal sub-components are known to be rigid, the assessment was performed only at the fundamental frequency of the component, varied as discussed previously.
- If the component is rigid and the internal components are known to be rigid or seismically insensitive, then the assessment was performed only at the ZPA.
- Consistent with industry consensus, functional fragility assessments were performed only with respect to horizontal input motion unless the component was determined to be sensitive to vertical input motion.

Clipping

The 80th percentile and Median Non-Exceedance Probability ISRS obtained from the structural analysis of the various buildings were used as input to the development of refined fragilities for the second and third quantifications respectively. In general, testing and evaluations utilize a broad-band spectrum as input to capacity determinations. Since ISRS typically have relatively sharp peaks, these peaks need to be clipped to provide a meaningful comparison to the capacity spectra. Clipping is done prior to the determination of demand (that is, the raw spectra are clipped) and is performed for both anchorage evaluations and functionality evaluations as discussed previously.

For anchorage evaluations, clipping was performed using the clipping recommendation from ASCE 4-16 [26]. This clipping approach was applied to all

ISRS that were obtained from finite element structural models. ISRS that were obtained from stick models were not clipped for anchorage evaluations. For functionality evaluations, the clipping factors contained in EPRI 6041-SL [14] are used. This approach to clipping is in accordance with industry positions and latest practices and was used to develop detailed fragilities for the second and third quantifications. The clipping was applied to the raw spectra per industry guidance. For the development of representative fragilities for the first quantification, a more simplified conservative approach was used for clipping the composite spectra.

Seismic Demand and Capacity

For purposes of determining functionality fragilities, the envelope of the horizontal spectra in the two horizontal directions was used (after clipping) to obtain the demand. Consistent with latest industry practice, vertical spectra were not considered in performing functionality evaluations when the vertical motion is not controlling, and the component is not sensitive to the vertical input motion.

Functional capacity of a component is based on the available seismic capacity information including component-specific test data, SQUG GERS, or seismic capacity based on earthquake experience (i.e., SQUG bounding spectrum from the SQUG GIP [30] or Table 2-4 of EPRI Report NP-6041-SL [14]). Very few site specific seismic tests or narrow-band test response spectra were used to determine the capacity for any components. However, in the very few cases where site-specific test data was available, the capacity from the test data was used. Clipping was adjusted or eliminated to account for any narrow-band test response spectra.

As stated, the results of the EPRI high frequency study [32] showed that components are not sensitive to seismic inputs above 20 Hz unless they are also sensitive to the same inputs below 20 Hz. Therefore, irrespective of the source, the peak of the spectrum used to establish capacity was extended to higher frequencies at the acceleration at the peak. The clipped ISRS at the location of the component was used to determine the demand. The 5% damped spectra were used unless a different damping is specified in the reference document used to obtain the capacity.

For anchorage evaluations, the spectra from the three orthogonal directions were applied with respect to the actual equipment layout in the plant. The spectra for the three orthogonal directional components of the earthquake were combined using the 100-40-40 rule. When it was obvious that one direction would control (for example, the component is much narrower in one direction than in the other), the 100% load was applied in the controlling direction. In cases where it was not obvious which direction controls, each of the possible combinations were considered.

Capacity information used for anchorage calculations is dependent on the controlling item. All structural elements (not just the anchorage) in the load path were considered. However, in most cases, the controlling structural element was determined by engineering experience or a review of existing calculations rather than through an explicit analysis of all structural elements. In each case, the capacity of the controlling structural element was determined using the latest available information as referenced in the individual calculations and/or as obtained through the walkdowns. The controlling structural element in each case is identified in the individual calculations.

Correlation

Recommendations related to correlation are provided in the fragility calculations for each component or group of components. The correlations between different components were established based on the following parameters. Essentially, for components to be correlated, all the following caveats must be met unless justification is provided for keeping components correlated that do not meet one or more of these caveats:

- Failure mode: The components have the same failure mode
- Similarity: The components are similar with respect to dimensions, weight, equipment class and function
- Orientation: The components are oriented in the same direction or the components are symmetrical with respect to the two orthogonal directions.
- Location: The components are located in the same building at the same location or at locations with virtually identical seismic inputs.
- Fragility: The fragilities of the components are identical or nearly identical.
- Other seismic concerns: There are no unique characteristics associated with one or more of the otherwise correlated components. This would include seismic interactions such as block walls that have a fragility less than other failure modes and which affect only a subset of the components or situations where one or more of the bounding spectrum caveats were not met for a subset of the otherwise correlated components.

In addition to correlation of components, relays are also correlated in certain cases. In general, relays are correlated if all the following caveats are met:

- The relays are in the same cabinet or in identical cabinets located in the same area and orientated in the same direction. Where the cabinets are in a large area, they must be within a portion of the floor that has the same seismic response to be correlated. That is, they must be on the same structural element (common slab between walls or beams that support the slab). If the cabinets are not correlated, the relays are not correlated.
- The relays are the same model number and have the same capacity.
- The relays are oriented in the same direction with respect to north-south and east-west directions.

- The relays are located at similar locations within the panel. That is, they are located such that the in-cabinet amplification factors are similar.
- There are no unique characteristics that would affect a subset of components that would otherwise be correlated.

Lateral and Vertical In-Cabinet Amplification Factors for Relay Fragility Analysis

As part of the development of fragilities for relays, it is necessary to develop in-cabinet amplification factors to amplify the ISRS. Generic conservative amplification factors are provided in the SQUG GIP [30] and in EPRI NP-6041-SL [14]. These amplification factors are applicable for the most severe locations within the various types of cabinets. However, they are conservative for other locations. Thus, an approach to determine less conservative but defendable amplification factors for relays that are not located in the most severe locations was developed and used to reduce the generic amplification values for some relays. This approach is described in detail in Appendix 6 of the Fragility Report [25].

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 5.4-2 and 5.4-3 for SCDF and Tables 5.5-2 and 5.5-3 for SLERF. Detailed (separation of variables, SOV) calculations have generally been performed for the highest risk significant SSCs, as well as for selected other components.

4.4.4 SSC Fragility Analysis Technical Adequacy

The PBAPS SPRA SSC Fragility Analysis was subjected to an independent peer review against the pertinent requirements in the PRA Standard [4]. The peer review was performed relative to Capability Category II for the full set of requirements in the PRA Standard.

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

5.0 Plant Seismic Logic Model

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

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

5.1 Development of the SPRA Plant Seismic Logic Model

The PBAPS seismic response model was developed by starting with the 2014 PBAPS internal events at power PRA model of record, as updated with the Application Specific Model (ASM) [61] as of February 28, 2018, and adapting the model in accordance with guidance in the SPID [2] and PRA Standard [4], including adding seismic fragility-related basic events to the appropriate portions of the internal events PRA, eliminating some parts of the internal events model that do not apply or that were screened-out, and adjusting the internal events PRA model human reliability analysis to account for response during and following a seismic event.

For the PBAPS SPRA, the following sections discuss the methods used to develop the seismic plant response model. The elements of the analysis are as follows:

- The seismic initiators are derived from the site specific seismic hazard analysis.
- The seismic accident sequences are developed by using a Seismic Initiating Event Tree (SIET) and, a set of Level 1 (core damage) and Level 2 (post-core damage) accident sequence event trees based on the PBAPS specific FPIE PRA model.
- The seismic system fault trees that support the event tree quantification based on the PBAPS specific FPIE PRA model.
- The fragility analysis that is performed to characterize the seismic induced failure modes of SSCs is used to model seismic induced system failure modes in the event tree and fault tree models.
- The interface of the operators with accident mitigation systems is incorporated into the seismic system fault trees as modified by the fragility analysis.
- The software used to process the above information into a cohesive framework and quantify the models. (See Section 5.3.1)

Initiating Events

The frequency of earthquakes at the PBAPS site is based on site-specific probabilistic seismic hazard analysis developed by Fugro [6]. The mean hazard curve is divided into eight ground motion ranges (seismic hazard intervals) for use in developing and quantifying the SPRA. Each seismic hazard interval initiator in the PBAPS seismic evaluation is assigned an initiator ID (e.g., %G4, “Seismic Initiating Event: 0.4g to 0.5g PGA”) and an initiator frequency. The frequency for the seismic hazard interval initiator is calculated as the exceedance frequency of the beginning point of the ground motion range minus the exceedance frequency of the end point of the ground motion range. The frequency of the last (highest) ground motion interval is the exceedance frequency at the beginning point of that interval. The seismic initiating events developed for the PBAPS SPRA are documented in the PBAPS Seismic PRA Initiating Event Notebook [40].

Accident Sequences

Event trees and fault trees are used to model the SPRA accident sequences. The accident sequence model accounts for the unique failure modes caused by seismic induced ground motion in addition to combinations of non-seismic failure modes. The sequence models address all the mitigation responses necessary to bring the plant to a safe shutdown. Event trees are a useful logic tool for displaying the seismic accident sequences.

The SPRA model process uses a seismic pre-tree, i.e., the Seismic Initiating Event Tree (SIET), to sort out the more pervasive effects of a seismic event that can lead directly to core damage or to a degraded plant condition (e.g., induced LOOP, induced large LOCA). The second tier of the event trees are systemic event trees (identical to those in the Level 1 internal events PRA) that evaluate the plant response and mitigation capability given the preconditions established in the SIET. Sequence logic transfers directly from the SIET into the systemic event trees to ensure that no information is lost in these transfers. The event trees are used to define the accident sequence progression and the assigned end state of the Level 1 events.

The methodology to group and transfer core damage sequences from the Level 1 event trees to the Level 2 Containment Event Trees (CETs) is identical to the FPIE PRA methodology. In addition, the seismic PRA is judged to create no unique Level 2 accident scenarios such that the SPRA Level 2 CETs are also identical to the FPIE CETs. The SPRA Level 2 CETs employ the identical definition for LERF timing and radionuclide release categories as the FPIE CETs. The Level 1 and Level 2 seismic accident sequence evaluation is documented in the PBAPS Seismic PRA Event Tree Notebook [41]. A sensitivity study has been performed to evaluate the potential risk impact on the SPRA results if the LERF definition is revised to consider seismic induced impacts of sheltering and evacuation offsite.

<u>System Fault Trees</u>

The SPRA system models reflect the as-built and as-operated plant. The internal events system fault tree models derived directly from the internal events model are used as a starting point for development of the SPRA system fault tree models.

The internal events PRA system fault trees are modified to reflect the unique aspects of the seismic hazard challenge. Therefore, both seismic and random SSC failures are accounted for in the SPRA model. These seismic response modifications include the following specific seismic attributes:

- Seismic hazard interval initiating events are inserted as the initiating event logic of the SIET sequences, as well as into system fault tree structures.
- SSC fragilities that would lead to a system or train failure are added to the system models.
- Effects on operator error probabilities due to the seismic induced changes to performance shaping factors are incorporated in the HEP calculations.
- Each of the above effects varies with seismic hazard intensity, i.e., varies by seismic hazard interval initiating event.

Specific aspects of the SSC fragility modeling and impacts include, but are not limited to, the following:

- For seismic induced LOOP events, recovery of offsite power is not credited for any hazard interval (e.g., failure of ceramic insulators).
- A fragility for seismic induced Very Small LOCA is explicitly modeled (e.g., potential to preclude credit for adequate RPV makeup from low flow CRD pumps)
- The unscreened contact chatter scenarios provided in Table 4.1.2-1 are considered and evaluated for inclusion in the SPRA model based on the identified system impact (e.g., divisional diesel fails to start or load). Given the high number of unscreened contact chatter scenarios (i.e., over 400), not all contact chatter scenarios are explicitly included in the SPRA model. Initial SPRA model quantifications helped identify the risk impact of individual or correlated contact chatter scenarios based on

associated system impact and fragility value. In addition, a Human Reliability Analysis (HRA) is performed to evaluate the potential credit for operator recovery of the contact chatter scenario (e.g., locally reset diesel) in the SPRA model. The PBAPS Seismic PRA Fragility Modeling Notebook [43] and the PBAPS Seismic PRA Methodology Notebook [39] provide further details on the methodology for including contact chatter events in the SPRA model.

- One of the aspects of the seismic hazard is that it could induce either a fire or a flood event. Because of this possibility, an assessment of these induced hazards is needed. The PBAPS SPRA approach to identification and assessment of postulated seismic-fire and seismic-flood interactions follows the SPRA Implementation Guide 3002000709 [11] and ASME/ANS RA-Sb–2013 [4] Supporting Requirements SFR-E4, SFR-E-5 and SPR-B9. This includes use of PBAPS fire PRA and internal flooding PRA information as well as plant walkdowns and drawing reviews to identify sources for consideration. The postulated seismic-induced sources for assessment includes non-safety electrical cabinets (although these are powered by offsite AC it may be postulated that such cabinets may experience seismic-induced arcing prior to seismic-induced loss of offsite power). Walkdowns were performed to identify additional sources as well as to assess the sources of seismic induced fire or flood events and to characterize their potential risk for inclusion in the seismic PRA model [33]. Hazards identified in the internal flood study and the internal fire analysis were considered by the walkdown team.

  Seismic-induced flooding from tanks and piping systems were assessed. Those of potential significance to the SPRA include piping systems with a significant suction source volume, can cause flow without auxiliary power and flood areas with equipment used in the SPRA. The assessment also considered seismic-induced actuation of fire suppression systems that could cause flooding. It was determined that seismic-induced failure of sprinkler heads, coupled with the potential for inadvertent actuation of the Fire Protection system caused by seismic-induced introduction of dust particles in the air, would create a flood that would cause loss of all batteries and switchgear along with other electrical components in the Control Building complex. This scenario is explicitly modeled in the SPRA. Other potential scenarios were investigated and determined to be non-significant risk contributors either due to limited consequences or piping with sufficiently high seismic-capacity.

- SSCs with a potential impact on containment integrity (e.g., containment bypass scenarios) were also evaluated and modeled accordingly for the Level 2 LERF model.

The PRA fault tree models contain the basic Boolean logic regarding SSC failure modes and their associated probabilities. For the PBAPS Seismic PRA, three types of fault tree models are developed:

- System Fault Trees
- Event Tree Nodal Fault Trees
- Integrated Fault Tree to model CDF and LERF accident sequences

Fragilities

Seismic fragility of a structure or equipment item is defined as the conditional probability of its failure at a given value of the seismic input or response parameter (e.g., PGA, stress, moment, or spectral acceleration). Seismic fragilities are needed in an SPRA to estimate the conditional seismic-induced failure probabilities of structures and mitigating systems (including their support systems) given a seismic initiating event. The fragilities are calculated using the methodologies discussed in Section 4.4.

SSC's of the same type that also possess the same location, elevation, and orientation are assigned to a single, correlated group as discussed further in Section 4.4. Due to the widespread nature of a seismic event, if a single SSC in a group were to fail, it can be assumed that all SSC's in the group would fail. This is consistent with the current state of practice.

Over 100 fragility groups are modeled in the PBAPS SPRA. Of the over 100 fragility groups, approximately 60 involve correlated fragility groups. Fully correlated response of the same or very similar equipment in the same structure and elevation is assumed. The SPRA does not model any partial correlation of fragility groups. Some of the risk significant correlated fragility groups include, but are not limited to, the following:

- All 125 VDC safety-related Battery Racks (8 SSCs)
- Main Control Room Emergency Relay Protection Boards (4 SSCs)
- Relay Chatter caused lockout of all safety-related 4kV Buses (8 SSCs)
- Relay Chatter caused unavailability of all EDGs (4 SSCs)

The development of the fragility groups, fragility correlation groups, and how they are incorporated into the SPRA model is documented in the PBAPS Seismic PRA Fragility Modeling Notebook [43].

Human Reliability Analysis

The scope of the Seismic PRA HRA is focused on the post-initiator operator actions. The pre-initiator Human Interactions (HI) are performed prior to a seismic event and are therefore not affected by the seismic event. Therefore, the assessment of the pre-initiator HIs remain the same as in the Internal Events PRA HRA (the pre-initiator HEPs existing in the FPIE system fault tree models propagate through the SPRA accident sequence logic and quantification).

The PBAPS Internal Events PRA uses a systematic approach for the identification and evaluation of operator actions in response to postulated accidents. The methods used are well established and are applied appropriately to the internal events models through use of the EPRI HRA Calculator® [50]. The seismic HRA uses these operator actions and these base calculations of Human Error Probabilities (HEPs) as input to the seismic HRA. PBAPS uses the EPRI HRA Calculator® for the internal events PRA, the fire PRA and the seismic PRA.

The human actions that are modeled in the Level 1 and Level 2 internal events PRA are included as basic events in the fault trees. The Human Error Probabilities (HEPs) generated from the Human Reliability Analysis (HRA) have been assigned unique basic event names. Additional actions specific to seismic conditions (relay chatter recovery actions) are incorporated into the SPRA.

The approach used for the SPRA HRA is to develop an integrated performance shaping factor (IPSF) for each HEP that is representative of the seismic accident sequence and apply the additional performance shape factor (i.e., IPSF) to the detailed internal events PRA HEPs based on EPRI guidance documents [59]. For HEPs identified to potentially have a high-risk contribution based on SPRA model quantifications, more detailed HEP evaluations (incorporating seismic impact adjustments) are developed using the EPRI HRA Calculator®. The dependent HEP probabilities are then also re-calculated using the seismic-adjusted HEPs. The details are documented in the PBAPS Seismic PRA HRA Notebook [42].

5.2 SPRA Plant Seismic Logic Model Technical Adequacy

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

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

5.3 Seismic Risk Quantification

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

5.3.1 SPRA Quantification Methodology

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

The analytic tools for the development of a quantified model are the EPRI CAFTA code suite augmented by the ACUBE Binary Decision Diagram (BDD) software. The EPRI CAFTA code suite [51] is well tested and widely used in various industries in numerous countries. The ACUBE code [52] is still expanding its capability, and this will increase the number of cutsets that can be precisely calculated using the BDD algorithm and less reliance on the Minimum Cut Upper Bound (MCUB) approximation.

The PBAPS SPRA model has been developed so that it is modular. Event trees (convertible to fault trees), event tree top logic (nodal fault trees), and system-level fault trees all have been developed as distinct files. In addition, the FRANX tool from the EPRI CAFTA suite has been used to develop a relational database for linking individual fragility events to existing modeled basic events. This modular structure allows individual files to remain manageable and reviewable. A single-top model used for quantification is developed by merging the previously described files. Merging the files into a single-top model is a standard CAFTA modeling technique and is performed by the user.

The model is quantified using PRAQuant, which is a code within the total CAFTA software suite. Also, due to the special circumstances within seismic modeling (i.e., over-counting caused by numerous high failure probability events), the ACUBE code, which uses the BDD algorithm, is used in model quantification to obtain a realistic assessment of the total CDF risk metric.

5.3.2 SPRA Model and Quantification Assumptions

This section discusses modeling assumptions made as part of the seismic PRA quantification. In addition, potential conservatisms that remain in the SPRA risk profile calculation include the following:

<u>Seismic Human Reliability Analysis (HRA)</u>

- As expected of a SPRA, the post-initiator FPIE-based human error probabilities (HEPs) in the SPRA are reconsidered and adjusted upward in failure probability to consider various seismic performance shaping factors. The approach used sets most post-initiator HEPs (except for

FLEX actions) directly to 1.0 failure probability for the highest hazard interval (i.e., %G8 for >0.9g). This may be too conservative given anecdotal information from seismic events.

Seismic Correlation

- Assuming 100% fragility correlation for like equipment installed similarly and located on the same elevation of the same building. This applies to non-significant and significant risk contributors (e.g., 125 VDC batteries racks). This approach defeats design redundancies if the redundancies are the same equipment and in the same general location. This modeling is a common SPRA practice; in fact, most SPRAs in the world use this method of applying a binary approach to fragility correlation modeling. This is a non-consensus analysis area but industry studies are currently investigating this topic. Assigning partial correlation factors to SSC fragility groups throughout the SPRA would likely create a model that cannot be quantified at a reasonable truncation limit and will introduce another significant element of modeling uncertainty (i.e., bases for the various partial correlation factors).

Accident Sequence Modeling

- Generally, there are limited success states for a seismic induced SBO because repair/recovery of seismic-induced failures (including seismic-induced loss of offsite power) is typically not credited in the SPRA. This is a typical SPRA approach. If recovery of offsite power were credited using an extreme weather related OSP non-recovery curve (which would be reflective of downed lines and poles) then the calculated SCDF and SLERF may potentially reduce by less than a percentage point (extreme weather related OSP non-recovery curves have very high failure probabilities in the first 24 hours). However, the PBAPS SPRA model explicitly credits FLEX mitigation strategies. If the FLEX equipment can be aligned in a timely manner and successfully operates as designed, then a success state (i.e., no core damage) for a seismic induced SBO can be achieved.

  The following FLEX strategies are incorporated into the SPRA (with system logic, seismic fragilities and human actions for the alignments):
  - Deep DC Load Shed When ELAP Declared
  - FLEX Generators to Unit 2 and Unit 3 Load Centers
  - FLEX Pump Flow Path for RPV Makeup
  - FLEX Pump Flow Path for Suppression Pool Makeup

- The PBAPS at-power SPRA does not incorporate the FLEX strategy to align the FLEX pump for spent fuel pool (SFP) makeup. This strategy has no direct relationship to at-power SCDF and SLERF accident sequences.

- No credit is modeled for isolation of seismic-induced breaks outside containment. The seismic-induced break outside containment sequence is modeled as leading directly to core damage and LERF. This is a small conservatism. If isolation of such a break could be credited with a proper basis in the modeling, the impact on SCDF would be negligible, but the SLERF may reduce by up to a few percent.
- Assignment of LERF to certain Level 2 PRA accident sequences may be conservative. The SPRA already credits recently re-calculated longer battery life in extending the time to core damage of seismic-induced SBO scenarios. However, certain phenomena exist that recent studies show, such as the NRC SOARCA studies [53], may require re-consideration in the PBAPS PRA models. Examples include the timings and magnitudes of severe accidents involving RPV melt-through and subsequent drywell shell melt-through. The degree of potential conservatisms in these types of Level 2 sequences is discussed in sensitivity cases.

Quantification Process

- The SPRA quantification process makes use of the EPRI ACUBE software module (which employs a binary decision diagram, BDD, algorithm) to minimize “overcounting” in the Boolean summation of result cutsets. A very minor level of over-counting in the SCDF and SLERF metrics (essentially the true values are achieved) exists in the base quantification. This level of precision can be challenged by individual risk applications that may set equipment to “failed” or high failure rates, but such challenges will be addressed as they arise in application of the model to risk informed decision making.

5.4 SCDF Results

The seismic PRA performed for PBAPS shows that the point estimate mean seismic CDF is 2.1E-05/yr for Unit 2 and also 2.1E-05/yr for Unit 3. The Unit 2 Seismic CDF of 2.1E-05/yr is calculated with a single top CAFTA model at a truncation that ranges from 1E-06/yr to 1E-10/yr depending on the seismic hazard interval quantified. The single top PRA model could not be quantified at a consistent truncation limit for all seismic hazard intervals due to quantification limitations associated with a typical desktop or laptop computer. Refer to Section 5.7 for a summary of the quantification truncation limits that support convergence of the PBAPS SPRA model for both SCDF and SLERF quantifications.

Given the similarities in the Unit 2 and Unit 3 SCDF values, the remainder of this section focuses on the Unit 2 results, except as noted. In general, PBAPS Unit 2 and Unit 3 are symmetrical.

The single top model accounts for both the accident sequence failure logic as well as the success logic. This calculation is then refined by the use of the ACUBE computer code operating on the cutsets from the single top to reduce any over counting of failures in the cutsets due to high failure probabilities in the cutsets.

Important Seismic Initiating Event Contributors

Table 5.4-1 summarizes the Unit 2 SCDF contributors by seismic initiating event. Figure 5.4-1 displays the results of Table 5.4-1 in graphical pie chart form, i.e., the CDF contributors by initiating event. Figure 5.4-2 shows the initiating event contribution in the form of a bar graph.

As can be seen from the graphical display, the seismic initiators %G5, %G6, and %G8 are the dominant seismic risk contributors. Seismic hazard interval initiator %G7 contributes less to SCDF than does %G8. Both %G7 and %G8 result in nearly a 100% likelihood of core damage but the initiator frequency of %G7 is lower than that of %G8 (i.e., %G7 is a bounded hazard interval and %G8 is the unbounded final hazard interval).

The seismic initiating event interval with the highest importance relative to the CDF risk metric is %G8 (>0.9g). Over this range core damage is essentially guaranteed.

Conditional Core Damage Probability (CCDP) values were also calculated for the initiators. These CCDP values are displayed in Figure 5.4-3. Figure 5.4-3 shows the CCDP for the %G7-%G8 initiators (0.75g->0.9g) as nearly 1.0. These ground motion values are close to or greater than the median capacity values (A_m) for some of the safety related SSCs at Peach Bottom (e.g., 125V DC battery racks). Thus, it is deemed reasonable that the CCDP for these initiators is very high.

The Unit 3 SCDF contributors by seismic initiating event are similar to those shown for Unit 2.

Important Contributors to Core Damage Frequency

Table 5.4-2 provides the Unit 2 SCDF Fussell-Vesely (FV) importance measures for SSC fragilities. The risk importances are calculated using cutset results (as typical in an R&R workstation environment) and using the EPRI ACUBE software to determine the individual basic event risk importance values. The SCDF FV values for SSC fragilities are based on a weighted sum of the individual SSC FV values calculated for the individual hazard intervals. In other words, the total FV of an

SSC fragility is the weighted sum of the associated eight (8) SSC fragility basic events (one per hazard interval). The SSC FV values for each hazard interval are calculated based on the cutset importance measures for each hazard interval as calculated by ACUBE. The weighted sum is a summation of the individual SSC FV values for an individual hazard interval multiplied by the ratio of the associated ACUBE SCDF for an individual hazard interval and the total ACUBE SCDF for all hazard intervals (i.e., total ACUBE SCDF of 2.14E-05/yr).

Note: The term FV is used here but the ACUBE software actually produces the Criticality Importance (CI) risk measure in place of FV. The CI and FV measures are very close numerically such that any minor difference in their values is non-significant for typical decision-making purposes. A discussion of the relationship of CI and FV is contained in the PBAPS SPRA Quantification report.

Consistent with past SPRA models, the top SCDF FV contributors are associated with AC and DC power supply.

The top 6 contributors to the Unit 2 SCDF FV are as follows:

Normal offsite power (FV = 0.981)

Normal offsite power is expected to have a high FV because there is a high probability for the seismic event to fail offsite power. The fragility is based on a “representative” calculation for offsite AC power consistent with industry guidance.

125 VDC Battery Racks (FV = 0.120)

The 125 VDC batteries have a high-risk impact because loss of all 125 VDC is modeled to result in unavailability of all EDGs, HPCI, and RCIC (i.e., short term station blackout event).

Conowingo Hydroelectric Plant (Alternate AC Source) (FV = 4.42E-02)

The Conowingo Dam provides alternate offsite AC power to PBAPS. The fragility is based on a “representative” calculation for offsite AC power consistent with industry guidance.

Main Control Room Relay Panels (FV = 3.81E-02)

Information from the PBAPS Fire PRA model indicate that the EDGs would be unavailable given structural failure of the MCR Relay Panels (e.g., correlated failure of panels 00C29(A-D)). This fragility group also includes correlated failure of panels 20C003, 20C004C, 30C003, and 30C004C. This may be conservative, but is judged to be appropriate based on current industry SPRA practice.

Relay Chatter Event 359A (All EDGs - Unrecoverable) (FV = 1.07E-02)

The relay chatter analysis study indicates that chatter of the 52B-TD5 relays would result in unavailability of all EDGs. This action is assumed to be unrecoverable given the time required to perform the local action.

D/G Room Supply Temp Control Panel 0(A-D)C479 (FV = 6.78E-03)

Seismic correlated failure of D/G room supply temperature control panels 0(A-D)C479 is modeled to result in unavailability of all primary EDG fan cooling. The fragility is based on a site specific "representative" calculation. However, increasing the control panel capacity would reduce the SCDF by less than 1% and is judged to not provide additional significant risk insights. Therefore, the use of the conservative fragility calculation for the D/G room supply temperature control panels is judged to not have an adverse impact on the PBAPS SPRA results.

The quantitative results showed that there were no SSCs with significant non-seismic failure contribution to SCDF (i.e., no random failures to start, run, etc. with FV > 5E-3).

Table 5.4-3 provides the Unit 3 SCDF Fussell-Vesely (FV) importance measures for SSC fragilities. The Unit 3 SCDF FV contributors are similar to the Unit 2 contributors with the exception of the addition of fragility groups S-DCBS4- (seismic correlated failure of DC Panels 20D24 and 30D21) and S-CC342A- (seismic correlated relay chatter of SI-Overcurrent relays) to the Top 10 Unit 3 contributors. Seismic failure of DC Panel 30D21 has a more risk significant impact on Unit 3 SCDF due to resultant unavailability of RCIC for Unit 3. In addition, seismic correlated relay chatter of the SI-Overcurrent relays was calculated to have a FV slightly below 5E-3 for Unit 2, but slightly above 5E-3 for Unit 3.

Table 5.4-4 provides the Unit 2 SCDF FV importance measures for the operator actions. Similar to the total FV for the SSC fragilities, the total FV for the operator actions is the sum of the individual FV values for the entire range of the hazard interval.

The top four (4) operator action contributors to the Unit 2 SCDF FV are all related to manually aligning backup pneumatic supply to the SRVs to support RPV depressurization following a LOOP event:

- OPERATORS FAIL TO VALVE IN N2 BOTTLES AFTER ACCUMULATOR DEPLETION (EARLY) operation (FV = 2.80E-02)
- OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 2 INS 'B' (EARLY) (FV = 2.66E-02)
- OPS FAILS TO VALVE IN N2 BOTTLES AFTER ACCUM DEPLETION (LATE; CONDITIONAL) (FV = 2.63E-02)

- OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 2 INS 'B' - DELAYED; CONDITIONAL (FV = 2.47E-02)

Table 5.4-5 provides the Unit 3 SCDF Fussell-Vesely (FV) importance measures for the operator actions. The Unit 3 SCDF FV contributors are very similar to the Unit 2 contributors.

Top 10 SCDF Cutset Evaluation

Table 5.4-6 provides the Top 10 Unit 3 SCDF cutsets because the Unit 3 results are slightly more limiting for the PBAPS SPRA model. The Top 10 Unit 2 SCDF cutsets are similar to the Unit 3 Top 10 SCDF cutsets so the Unit 2 cutsets are not explicitly provided. The cutset result file combines the cutsets from all seismic hazard intervals (i.e., %G1 through %G8). However, all of the Top 10 cutsets involve a %G8 seismic initiating event (seismic magnitude >0.9g), which is the dominant contributor to SCDF. The SCDF values identified for each of the cutsets is based on the independent calculated cutset frequency. The integrated SCDF when combining the cutsets using the EPRI ACUBE software may result in a much lower total SCDF.

Cutset #1 (SCDF = 4.94E-6/yr): This cutset involves a %G8 seismic initiating event (seismic magnitude >0.9g) with a LOOP, but with EDG power and HPCI or RCIC initially available. The operator fails to align Suppression Pool Cooling (SPC) with a Human Error Probability (HEP) of 1.0. Although an HEP of 1.0 may appear to be conservative given the long time available to align SPC (e.g., >10 hours), assigning a 1.0 is consistent with the HRA methodology for %G8 events (e.g., loss of critical control room instrumentation). Loss of SPC eventually leads to containment heatup and reaching the Heat Capacity Temperature Limit (HCTL) within approximately 6 hours based on plant specific MAAP thermal hydraulic calculations used to support the FPIE PRA model. The EOPs direct the operators to Emergency Depressurize the RPV, thereby, resulting in unavailability of HPCI and RCIC for long term RPV makeup. Then the operators fail to manually align long term backup pneumatic supplies to the ADS SRVs from either the Nitrogen Bottles or the CAD tank (i.e., SGIG tank). Upon depletion of short term pneumatic supply from the ADS SRV accumulators, the ADS SRVs will close and the RPV will re-pressurize. It could be postulated that re-pressurizing the RPV will restore HPCI or RCIC operation. However, this is not credited consistent with typical industry PRA modeling assumptions (i.e., do not credit a beneficial failure). In addition, two (2) CRD pumps could be credited to support successful high pressure RPV makeup, but Instrument Air compressors are not available during a LOOP event to support valve control to maximize CRD flow. These failures lead to a core damage event during a loss of offsite power (Accident Class 1B) with the RPV at high pressure.

Cutset #2 (SCDF = 4.94E-6/yr): This cutset is similar to Cutset #1 in that it involves a %G8 seismic initiating event (seismic magnitude >0.9g) with a LOOP and EDG power and HPCI or RCIC initially available. A dependent operator action fails all of the following actions: Early SPC, Late SPC, Containment Venting, and alternate CRD pump cooling. The dependent HEP has a value of 1.0 because the SPRA HRA methodology evaluates a 1.0 HEP for each of the four (4) individual operator actions. HPCI and RCIC become unavailable after reaching HCTL, but long term RPV depressurization is successful in this cutset. Low pressure RPV makeup is initially provided by LPCI or Core Spray. Given failure of the operator to vent the primary containment, the SRVs reclose on high containment pressure and preclude continued low pressure RPV makeup. The containment pressure continues to rise until the containment fails on over-pressurization. Continued injection from LPCI and Core Spray is assumed unavailable due to various phenomenological issues (e.g., loss of NPSH, steam binding). CRD is unavailable because primary CRD pump cooling from RBCCW is unavailable during a LOOP event and the operator fails to implement alternate CRD pump cooling. These failures lead to a core damage event during a loss of offsite power with Loss of Containment Heat Removal (Accident Class 2A).

Cutsets #3 through #7 (SCDF ranges from 4.94E-6/yr to 4.70E-6/yr): The next five (5) cutsets are similar to Cutset #2 but involve different failure modes to prevent CRD from providing RPV makeup for the 24 hours mission time (e.g., failure to refill the CST from the RWST or TDT, seismic failure of the CST, seismic induced Very Small LOCA). These failures lead to a core damage event during a loss of offsite power with Loss of Containment Heat Removal (Accident Class 2A).

Cutset #8 (SCDF = 3.84E-6/yr): Cutset #8 is similar to Cutset #1 except RCIC is unavailable due to a relay chatter event preventing EDG power to supply the RCIC battery charger and no credit for operator recovery from the relay chatter event. The operator also fails to perform load shed actions to extend 125 VDC battery life to support extended RCIC operation. HPCI is not credited to operate for longer than 12 hours based on modeling assumptions from the PBAPS FPIE PRA model (e.g., room cooling not credited). Two (2) CRD pumps can be credited for long term RPV makeup, but the operators fail to maximize CRD flow. In addition, the operators fail to manually align long term backup pneumatic supplies to the ADS SRVs from either the Nitrogen Bottles or the CAD tank (i.e., SGIG tank). In addition, the operators at the Conowingo Dam fail to align alternate AC power to PBAPS. These failures lead to a core damage event during a loss of offsite power (Accident Class 1B) with the RPV at high pressure.

Cutset #9 (SCDF = 3.84E-6/yr): Cutset #9 is similar to Cutset #8 but involves different operator action to prevent aligning alternate AC power to PBAPS. Cutset #9 involves failure of the operators at PBAPS to align power from the SBO line from Conowingo Dam to the emergency 4KV busses. These failures lead to a core damage event during a loss of offsite power (Accident Class 1B) with the RPV at high pressure.

Cutset #10 (SCDF = 3.84E-6/yr): Cutset #10 is similar to Cutset #8 but involves a different failure mode to prevent CRD from providing RPV makeup for the 24-hour mission time (e.g., failure of the operator to implement SE-11 to support alternate CRD cooling from RBCCW). These failures lead to a core damage event during a loss of offsite power (Accident Class 1B) with the RPV at high pressure.

Although the cutsets may be conservative because of the many HEPs set to 1.0, the cutsets appear valid based on the PBAPS SPRA modeling assumptions.

A review of sample cutsets from each decade of quantification results did not identify any improper cutsets.

<u>SCDF Accident Class Contributors</u>

The dominant Level 1 accident class contributors to the PBAPS SCDF include the following:

- Class 1B (Station Blackout) – 49%
- Class 2A (Loss of Containment Heat Removal with successful Containment Venting) – 25%
- Class 2F (Similar to Class 2A except the vent operates as designed, suppression pool is saturated but intact) – 21%
- Class 1C (ATWS with failure of RPV makeup) – 4%

The Level 1 accident class definitions for the PBAPS SPRA are based on those defined for the PBAPS FPIE PRA model. The Level 1 accident classes are described in Table 3-1 of the PBAPS SPRA Methods Notebook [39].

Class 1B (Station Blackout) has the highest contribution to the PBAPS Level 1 SCDF. Seismic induced LOOP events are generally amongst the highest contributors for typical SPRA models because recovery of offsite AC power is generally not credited. In addition, one of the highest contributors to SCDF is seismic induced correlated failure of all eight (8) 125 VDC batteries, resulting in unavailability of the EDGs, HPCI, and RCIC. Other significant contributors to Station Blackout scenarios include 1) correlated failure the Main Control Room Essential Relay Panels 00C29A(B/C/D), and 2) correlated relay chatter of the 52B-TD5 relays, which are modeled to cause unavailability of all EDGs). Even if HPCI or RCIC are initially available, no credit for offsite AC power recovery results in eventual depletion of the batteries, leading to core damage. In addition, for one of the dominant Class 1B risk contributors (i.e., loss of 125 VDC batteries) the time to core damage is approximately 1 hour. Therefore, insufficient time is available to support alignment of FLEX equipment for mitigation.

Class 2A is Loss of Containment Heat Removal with failure of Containment Venting (e.g., due to operator failure to initiate Containment Venting, failure to align backup pneumatic supply). Following failure of Containment Venting, drywell pressure increases and results in re-closure of the SRVs and RPV re-pressurization. High pressure RPV makeup is credited, but unavailability of CRD for external injection would result in core damage.

Class 2F (Loss of Containment Heat Removal with successful Containment Venting) is similar to Class 2A, but Containment Venting is available. Following successful Containment Venting, RPV makeup systems with suction aligned to the suppression pool are assumed unavailable due to loss of adequate NPSH. RPV makeup is credited post Containment Venting, but unavailability of HPSW or CRD for external injection would result in core damage.

Class 1C (ATWS with failure of RPV makeup) has a smaller contribution to SCDF compared to the other accident classes due to the relatively high seismic capacity modeled for the SCRAM system (Am=1.35g). Following a failure to SCRAM event, failure of operator action or hardware to support adequate RPV level control is modeled to result in core damage.

Table 5.4-1
UNIT 2 CDF CONTRIBUTORS BY SEISMIC HAZARD INTERVAL INITIATING EVENT

Seismic Hazard IntervalBin	Description	Interval Frequency (/yr)	Interval CDF (/yr)	% of Total SCDF	Cumulative SCDF (/yr)
%G1	%G1 - Hazard Curve: PBAPS Hazard Curve - PGA Range: 0.05g to 0.2g	4.7E-04	3.6E-09	0%	3.6E-09
%G2	%G2 - Hazard Curve: PBAPS Hazard Curve - PGA Range: 0.2g to 0.3g	4.2E-05	5.7E-07	3%	5.7E-07
%G3	%G3 - Hazard Curve: PBAPS Hazard Curve - PGA Range: 0.3g to 0.4g	1.8E-05	1.8E-06	8%	2.3E-06
%G4	%G4 - Hazard Curve: PBAPS Hazard Curve - PGA Range: 0.4g to 0.5g	9.5E-06	2.9E-06	14%	5.3E-06
%G5	%G5 - Hazard Curve: PBAPS Hazard Curve - PGA Range: 0.5g to 0.6g	5.6E-06	4.5E-06	21%	9.7E-06
%G6	%G6 - Hazard Curve: PBAPS Hazard Curve - PGA Range: 0.6g to 0.75g	4.7E-06	4.2E-06	20%	1.4E-05
%G7	%G7 - Hazard Curve: PBAPS Hazard Curve - PGA Range: 0.75g to 0.9g	2.7E-06	2.5E-06	11%	1.6E-05
%G8	%G8 - Hazard Curve: PBAPS Hazard Curve - PGA Range: > 0.9g	5.3E-06	4.9E-06	23%	2.1E-05

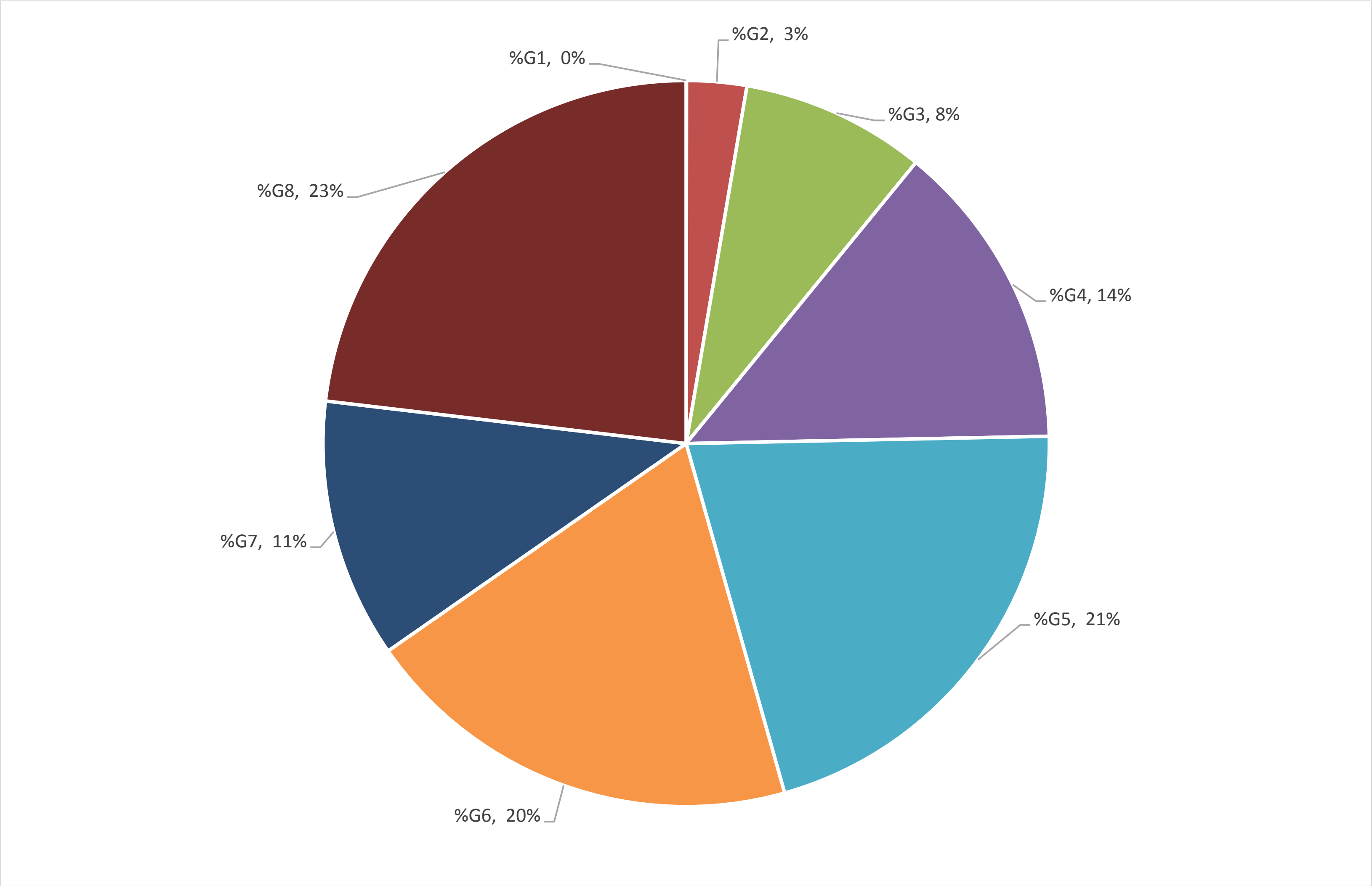

Figure 5.4-1
PBAPS SPRA UNIT 2 SCDF BY INITIATING EVENT

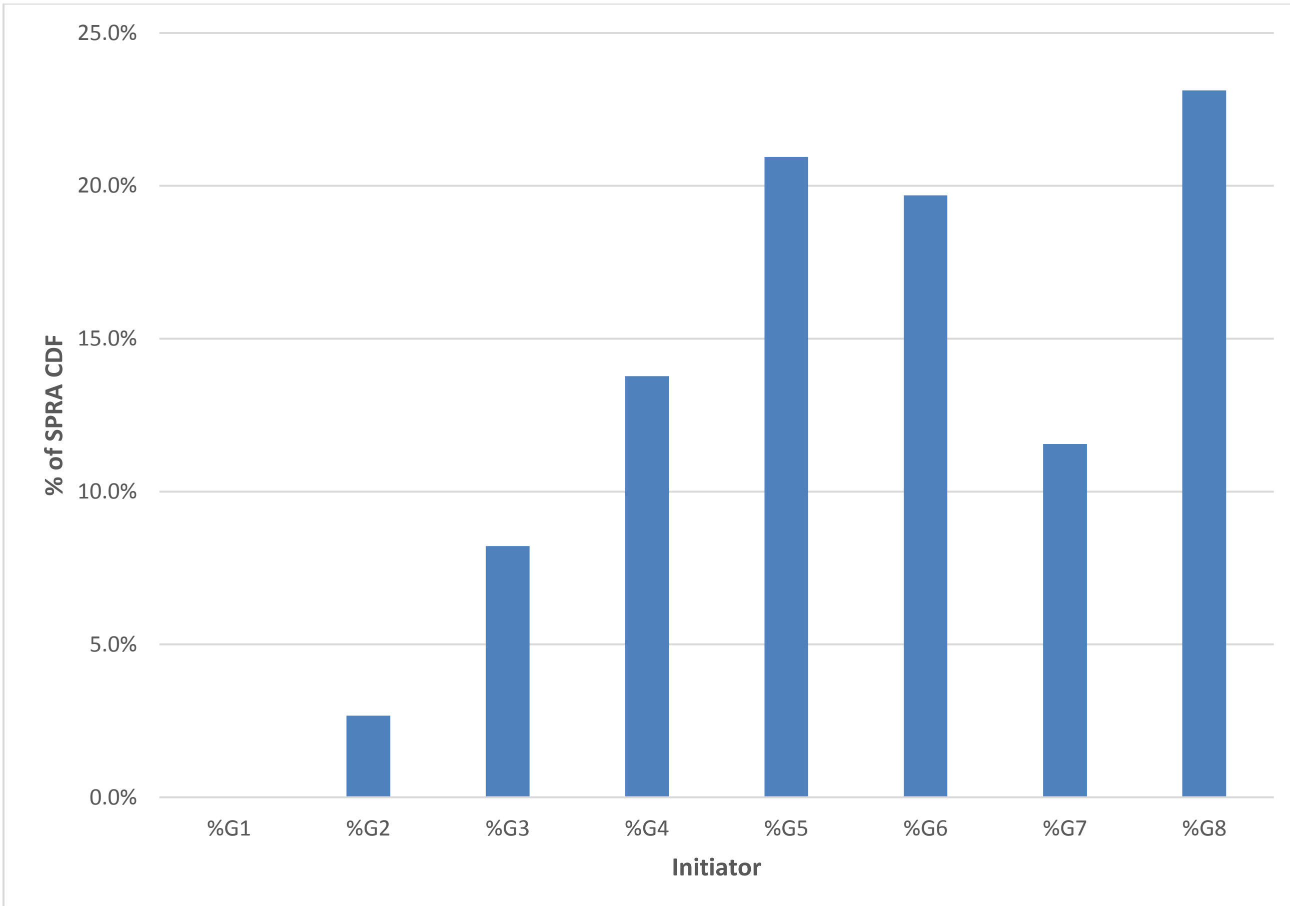

Figure 5.4-2

PBAPS SPRA UNIT 2 SCDF BY INITIATING EVENT

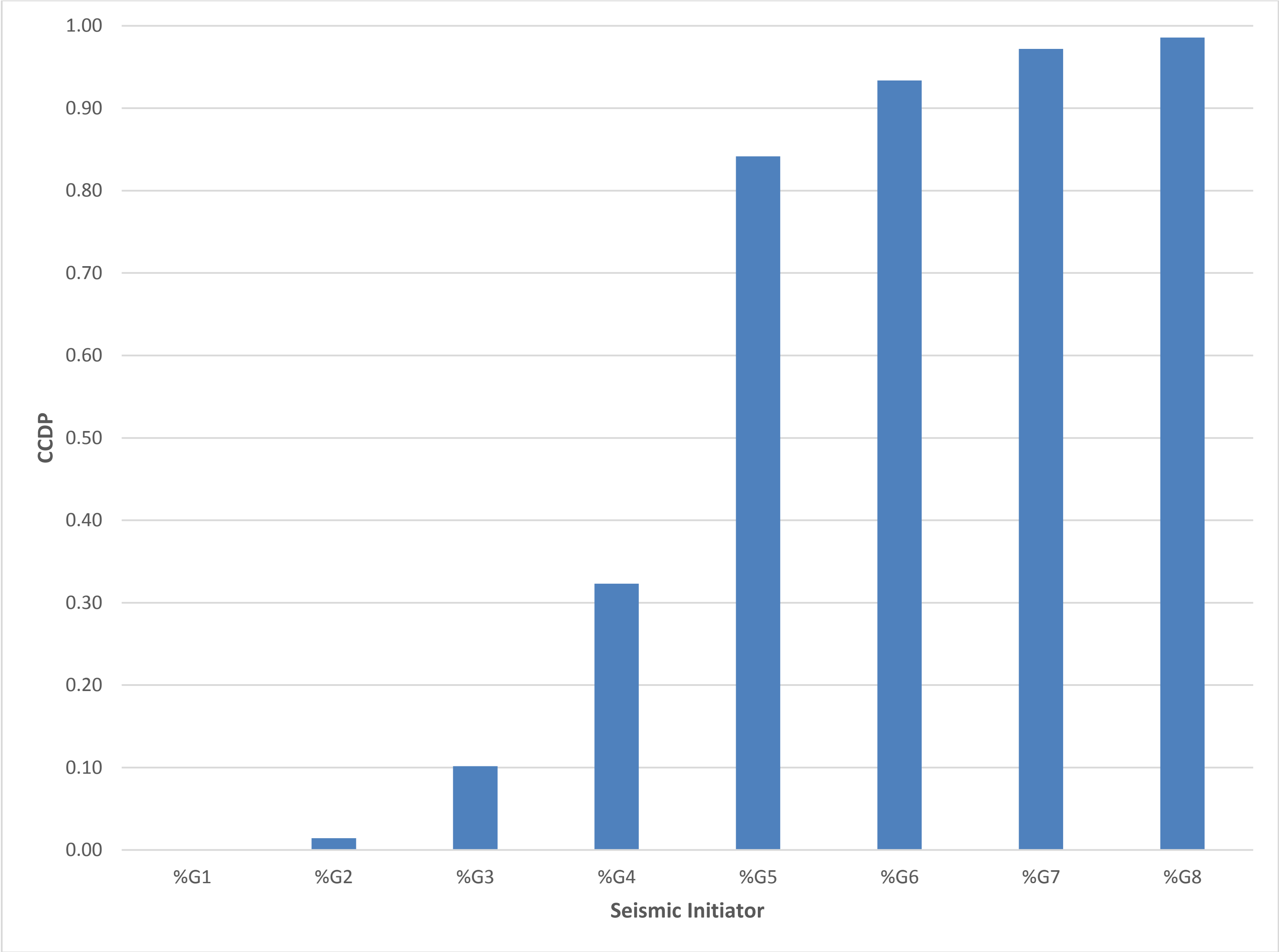

Figure 5.4-3

PBAPS SPRA UNIT 2 CCDP BY INITIATING EVENT

TABLE 5.4-2

PBAPS UNIT 2 SCDF FUSSELL-VESELY IMPORTANCE MEASURES FOR SSC FRAGILITIES

FRAGILITY GROUP ID	FRAGILITY GROUP DESCRIPTION	FV TOTAL	Am (g)	βr	βu	Failure Mode	Fragility Method
OSP	Offsite Power	9.81E-01	0.3	0.30	0.45	Functional	Representative
S-DCBT1-	DC Batteries 2(A-D)D01, 3(A-D)D01	1.20E-01	0.73	0.28	0.52	Anchorage	SOV
S-CNWG2-	Conowingo Hydroelectric Plant (OSP)	4.42E-02	0.3	0.30	0.45	Functional	Representative
S-CEPA1-	Panel 20C003, 20C004C, 30C003, 30C004C, 00C29(A-D)	3.81E-02	0.82	0.28	0.37	Anchorage	SOV
S-CC359A-	Correlated Relay Chatter Group 359A (52B-TD5 relays) (All EDGs - Unrecoverable)	1.07E-02	0.98	0.30	0.43	Functional	SOV
S-DGPA1-	D/G Room Supply Temp Control Panel 0(A-D)C479	6.78E-03	0.86	0.24	0.32	Functional	Representative

TABLE 5.4-3

PBAPS UNIT 3 SCDF FUSSELL-VESELY IMPORTANCE MEASURES FOR SSC FRAGILITIES

FRAGILITY GROUP ID	FRAGILITY GROUP DESCRIPTION	FV TOTAL	Am (g)	βr	βu	Failure Mode	Fragility Method
OSP	Offsite Power	9.81E-01	0.3	0.3	0.45	Functional	Representative
S-DCBT1-	DC Batteries 2(A-D)D01, 3(A-D)D01	1.19E-01	0.73	0.28	0.52	Anchorage	SOV
S-CNWG2-	Conowingo Hydroelectric Plant (OSP)	5.27E-02	0.3	0.3	0.45	Functional	Representative
S-CEPA1-	Panel 20C003, 20C004C, 30C003, 30C004C, 00C29(A-D)	3.77E-02	0.82	0.28	0.37	Anchorage	SOV
S-CC359A-	Correlated Relay Chatter Group 359A (52B-TD5 relays) (All EDGs - Unrecoverable)	1.14E-02	0.98	0.3	0.43	Functional	SOV
S-DCBS4-	DC Panel 20D24, 30D21	1.01E-02	0.86	0.28	0.52	Anchorage	SOV
S-DGPA1-	D/G Room Supply Temp Control Panel 0(A-D)C479	7.26E-03	0.86	0.24	0.32	Functional	Representative
S-CC342A-	Correlated Relay Chatter Group 342A (SI-Overcurrent relays) (All 4KV Busses - Unrecoverable)	5.02E-03	1.29	0.3	0.44	Functional	SOV

TABLE 5.4-4

PBAPS UNIT 2 SCDF FUSSEL-VESELY IMPORTANCE MEASURES FOR OPERATOR ACTIONS

OPERATOR ACTION ID	OPERATOR ACTION DESCRIPTION	FV TOTAL
AHUBTL-RDXI2	OPERATORS FAIL TO VALVE IN N2 BOTTLES AFTER ACCUMULATOR DEPLETION (EARLY)	2.80E-02
AHU--CADDXI2	OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 2 INS 'B'	2.66E-02
AHUBTL-RDXD2	OPS FAILS TO VALVE IN N2 BOTTLES AFTER ACCUM DEPLETION (LATE; CONDITIONAL)	2.63E-02
AHU--CADDXD2	OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 2 INS 'B' - DELAYED; CONDITIONAL	2.47E-02
QHUFXL13DXI2	OPERATOR FAILS TO ALIGN FLEX GENERATOR TO LC E124 OR E324	1.86E-02
EHURLY4KDXI2	OPERATOR FAILS TO MITIGATE RELAY CHATTER FOR 4KV BUSES (SEISMIC)	1.61E-02
QHULS-ACDXI2	OPERATOR FAILS TO PERFORM DEEP DC LOAD SHED	1.34E-02
EHU-SE11DXI0	OPERATOR FAILS TO CROSS TIE 4KV EMERGENCY BUSES	6.93E-03
KHUDGFANDXI0	OPERATOR FAILS TO MANUALLY INITIATE SUPPLEMENTAL FAN	6.68E-03

Notes to Table:

(1) This table covers independent and dependent post-initiator HEPs and their risk contribution; however, if dependent HEPs do not show up in this table that is because their FV value is below 5E-3.

(2) The independent post-initiator HEP FV values presented in this table do not include the risk contribution from the independent HEPs appearing in dependent HEPs.

TABLE 5.4-5

PBAPS UNIT 3 SCDF FUSSELL-VESELY IMPORTANCE MEASURES FOR POST-INITIATOR OPERATOR ACTIONS

Operator Action ID	Description	FV Total
AHUBTL-RDXI3	OPERATORS FAIL TO VALVE IN N2 BOTTLES AFTER ACCUMULATOR DEPLETION (EARLY)	2.29E-02
AHU--CADDXI3	OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 3 INS 'B'	2.23E-02
AHUBTL-RDXD3	OPS FAILS TO VALVE IN N2 BOTTLES AFTER ACCUM DEPLETION (LATE; CONDITIONAL)	2.04E-02
AHU--CADDXD3	OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 3 INS 'B' - DELAYED; CONDITIONAL	1.89E-02
QHUFXL13DXI3	OPERATOR FAILS TO ALIGN FLEX GENERATOR TO LC E134 AND LC E334	1.61E-02
EHURLY4KDXI3	OPERATOR FAILS TO MITIGATE RELAY CHATTER FOR 4KV BUSES (SEISMIC)	1.43E-02
QHULS-ACDXI3	DEEP DC LOAD SHED WHEN ELAP DECLARED (STEPS FOR RCIC)	1.15E-2
EHU-SE11DXI0	OPERATOR FAILS TO CROSS TIE 4KV EMERGENCY BUSES	9.91E-03
KHUDGFANDXI0	OPERATOR FAILS TO MANUALLY INITIATE SUPPLEMENTAL FAN	7.19E-03

Notes to Table:

(1) This table covers independent and dependent post-initiator HEPs and their risk contribution; however, if dependent HEPs do not show up in this table that is because their FV value is below 5E-3.

(2) The independent post-initiator HEP FV values presented in this table do not include the risk contribution from the independent HEPs appearing in dependent HEPs.

TABLE 5.4-6

PBAPS UNIT 3 TOP 10 SEISMIC CDF CUTSETS

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
1	4.94E-06	5.25E-06	%G8	Seismic Initiating Event (>0.9g)
		1.00E+00	1-CL-1B	CLASS 1B
		1.00E+00	1-SEQ-LP1-022	SEQUENCE LP1-022
		9.86E-01	OSP-C-%G8	SEISMIC FRAGILITY FOR %G8: Offsite Power
		1.00E+00	SRX08_AHUBTL-RDXD3	S-HEP G8: OPS FAILS TO VALVE IN N2 BOTTLES AFTER ACCUM DEPLETION (LATE; CONDITIO
		1.00E+00	SRX08_AHUBTL-RDXI3	S-HEP G8: OPERATORS FAIL TO VALVE IN N2 BOTTLES AFTER ACCUMULATOR DEPLETION (EAR
		1.00E+00	SRX08_AHU--CADDXD3	S-HEP G8: OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 2 INS 'B' - DELAYED; CONDITIO
		1.00E+00	SRX08_AHU--CADDXI3	S-HEP G8: OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 2 INS 'B'
		1.00E+00	SRX08_DHU--SPCDXI3	S-HEP G8: OPERATORS FAIL TO INITIATE RHR IN SPC MODE (NON-ATWS) AVOID HCTL
		9.54E-01	ZZAVFACTOR	PLANT AVAILABILITY FACTOR
2	4.94E-06	5.25E-06	%G8	Seismic Initiating Event (>0.9g)
		1.00E+00	1-CL-2A	CLASS 2A
		1.00E+00	1-SEQ-LP1-011	SEQUENCE LP1-011
		9.86E-01	OSP-C-%G8	SEISMIC FRAGILITY FOR %G8: Offsite Power

TABLE 5.4-6

PBAPS UNIT 3 TOP 10 SEISMIC CDF CUTSETS

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
		1.00E+00	SDP08_ZHU--159DXI3-DDMV	S-DHEP G8: Joint HEP for DHU--SPCDXD3, DHU--SPCDXI3, MHUSE11WDXI3, VHU-VENTDXI3
		9.54E-01	ZZAVFACTOR	PLANT AVAILABILITY FACTOR
3	4.94E-06	5.25E-06	%G8	Seismic Initiating Event (>0.9g)
		1.00E+00	1-CL-2A	CLASS 2A
		1.00E+00	1-SEQ-LP1-011	SEQUENCE LP1-011
		9.86E-01	OSP-C-%G8	SEISMIC FRAGILITY FOR %G8: Offsite Power
		1.00E+00	SDP08_ZHU--163DXI3-DDV	S-DHEP G8: Joint HEP for DHU--SPCDXD3, DHU--SPCDXI3, VHU-VENTDXI3
		1.00E+00	SRX08_YHU--GRFDXI3	S-HEP G8: OPERATORS FAIL TO REFILL CST FROM RWST USING THE GRAVITY FEED
		9.54E-01	ZZAVFACTOR	PLANT AVAILABILITY FACTOR
4	4.94E-06	5.25E-06	%G8	Seismic Initiating Event (>0.9g)
		1.00E+00	1-CL-2A	CLASS 2A
		1.00E+00	1-SEQ-LP1-011	SEQUENCE LP1-011
		9.86E-01	OSP-C-%G8	SEISMIC FRAGILITY FOR %G8: Offsite Power

TABLE 5.4-6

PBAPS UNIT 3 TOP 10 SEISMIC CDF CUTSETS

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
		1.00E+00	SDP08_ZHU--163DXI3-DDV	S-DHEP G8: Joint HEP for DHU--SPCDXD3, DHU--SPCDXI3, VHU-VENTDXI3
		1.00E+00	SRX08_YHUGRTDTDXI3	S-HEP G8: OPERATORS FAIL TO REFILL UNIT 3 CST FROM THE TDT (GRAVITY FEED)
		9.54E-01	ZZAVFACTOR	PLANT AVAILABILITY FACTOR
5	4.94E-06	5.25E-06	%G8	Seismic Initiating Event (>0.9g)
		1.00E+00	1-CL-2A	CLASS 2A
		1.00E+00	1-SEQ-LP1-011	SEQUENCE LP1-011
		9.86E-01	OSP-C-%G8	SEISMIC FRAGILITY FOR %G8: Offsite Power
		1.00E+00	SDP08_ZHU--166DXI3-DDVZ	S-DHEP G8: Joint HEP for DHU--SPCDXD3, DHU--SPCDXI3, VHU-VENTDXI3, ZHU--CSTDXI3
		9.54E-01	ZZAVFACTOR	PLANT AVAILABILITY FACTOR
6	4.72E-06	5.25E-06	%G8	Seismic Initiating Event (>0.9g)
		1.00E+00	1-CL-2A	CLASS 2A
		1.00E+00	1-SEQ-LP1-011	SEQUENCE LP1-011
		9.86E-01	OSP-C-%G8	SEISMIC FRAGILITY FOR %G8: Offsite Power
		9.56E-01	S-CNCT1--C-%G8	S-FRAG %G8: Condensate Storage Tank 20T010, 30T010

TABLE 5.4-6

PBAPS UNIT 3 TOP 10 SEISMIC CDF CUTSETS

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
		1.00E+00	SDP08_ZHU--163DXI3-DDV	S-DHEP G8: Joint HEP for DHU--SPCDXD3, DHU--SPCDXI3, VHU-VENTDXI3
		9.54E-01	ZZAVFACTOR	PLANT AVAILABILITY FACTOR
7	4.70E-06	5.25E-06	%G8	Seismic Initiating Event (>0.9g)
		1.00E+00	1-CL-2A	CLASS 2A
		1.00E+00	1-SEQ-LP1-011	SEQUENCE LP1-011
		1.00E+00	BPHVSLOCDXI3	VERY SMALL LOCA OCCURS DUE TO SEISMIC INITIATOR
		9.86E-01	OSP-C-%G8	SEISMIC FRAGILITY FOR %G8: Offsite Power
		1.00E+00	SDP08_ZHU--163DXI3-DDV	S-DHEP G8: Joint HEP for DHU--SPCDXD3, DHU--SPCDXI3, VHU-VENTDXI3
		9.51E-01	SVSL-C-%G8	SEISMIC FRAGILITY FOR %G8: Seismic Induced Very Small LOCA
		9.54E-01	ZZAVFACTOR	PLANT AVAILABILITY FACTOR
8	3.84E-06	5.25E-06	%G8	Seismic Initiating Event (>0.9g)
		1.00E+00	1-CL-1B	CLASS 1B
		1.00E+00	1-SEQ-LP1-006	SEQUENCE LP1-006
		1.00E+00	HPH--SDCDXI3	HPCI LOST AFTER ~ 12 HOURS

TABLE 5.4-6

PBAPS UNIT 3 TOP 10 SEISMIC CDF CUTSETS

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
		9.86E-01	OSP-C-%G8	SEISMIC FRAGILITY FOR %G8: Offsite Power
		7.77E-01	S-CC390--C-%G8	S-FRAG %G8: Relay Chatter Event 390 (33-102 relay) (EDG C - Unrecoverable)
		1.00E+00	SRX08_AHUBTL-RDXD3	S-HEP G8: OPS FAILS TO VALVE IN N2 BOTTLES AFTER ACCUM DEPLETION (LATE; CONDITIO
		1.00E+00	SRX08_AHUBTL-RDXI3	S-HEP G8: OPERATORS FAIL TO VALVE IN N2 BOTTLES AFTER ACCUMULATOR DEPLETION (EAR
		1.00E+00	SRX08_AHU--CADDXD3	S-HEP G8: OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 2 INS 'B' - DELAYED; CONDITIO
		1.00E+00	SRX08_AHU--CADDXI3	S-HEP G8: OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 2 INS 'B'
		1.00E+00	SRX08_BHU--MAXDXI3	S-HEP G8: OPERATORS FAIL TO MAXIMIZE CRD FLOW FOR RPV INJECTION PER T-246 (AT 4
		1.00E+00	SRX08_EHUCWGCNDXI0	S-HEP G8: CONOWINGO OPERATOR FAILS TO ENERGIZE SBO LINE TO PEACH BOTTOM (EXECUTI
		1.00E+00	SRX08_EHULS-ACDXI3	S-HEP G8: INITIAL LOAD SHED PER SE-11, ATT. T (STEPS FOR RCIC)
		1.00E+00	SRX08_EHU-SE11DXI0	S-HEP G8: OPERATOR CROSS TIES 4KV EMERGENCY BUSES
		9.54E-01	ZZAVFACTOR	PLANT AVAILABILITY FACTOR
9	3.84E-06	5.25E-06	%G8	Seismic Initiating Event (>0.9g)
		1.00E+00	1-CL-1B	CLASS 1B
		1.00E+00	1-SEQ-LP1-006	SEQUENCE LP1-006

TABLE 5.4-6

PBAPS UNIT 3 TOP 10 SEISMIC CDF CUTSETS

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
		1.00E+00	HPH--SDCDXI3	HPCI LOST AFTER ~ 12 HOURS
		9.86E-01	OSP-C-%G8	SEISMIC FRAGILITY FOR %G8: Offsite Power
		7.77E-01	S-CC390--C-%G8	S-FRAG %G8: Relay Chatter Event 390 (33-102 relay) (EDG C - Unrecoverable)
		1.00E+00	SRX08_AHUBTL-RDXD3	S-HEP G8: OPS FAILS TO VALVE IN N2 BOTTLES AFTER ACCUM DEPLETION (LATE; CONDITIO
		1.00E+00	SRX08_AHUBTL-RDXI3	S-HEP G8: OPERATORS FAIL TO VALVE IN N2 BOTTLES AFTER ACCUMULATOR DEPLETION (EAR
		1.00E+00	SRX08_AHU--CADDXD3	S-HEP G8: OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 2 INS 'B' - DELAYED; CONDITIO
		1.00E+00	SRX08_AHU--CADDXI3	S-HEP G8: OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 2 INS 'B'
		1.00E+00	SRX08_BHU--MAXDXI3	S-HEP G8: OPERATORS FAIL TO MAXIMIZE CRD FLOW FOR RPV INJECTION PER T-246 (AT 4
		1.00E+00	SRX08_EHUCWGPBDXI0	S-HEP G8: PEACH BOTTOM OPERATOR FAILS TO ALIGN CONOWINGO SBO LINE TO EMERGENCY 4
		1.00E+00	SRX08_EHULS-ACDXI3	S-HEP G8: INITIAL LOAD SHED PER SE-11, ATT. T (STEPS FOR RCIC)
		1.00E+00	SRX08_EHU-SE11DXI0	S-HEP G8: OPERATOR CROSS TIES 4KV EMERGENCY BUSES
		9.54E-01	ZZAVFACTOR	PLANT AVAILABILITY FACTOR
10	3.84E-06	5.25E-06	%G8	Seismic Initiating Event (>0.9g)
		1.00E+00	1-CL-1B	CLASS 1B

TABLE 5.4-6

PBAPS UNIT 3 TOP 10 SEISMIC CDF CUTSETS

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
		1.00E+00	1-SEQ-LP1-006	SEQUENCE LP1-006
		1.00E+00	HPH--SDCDXI3	HPCI LOST AFTER ~ 12 HOURS
		9.86E-01	OSP-C-%G8	SEISMIC FRAGILITY FOR %G8: Offsite Power
		7.77E-01	S-CC390--C-%G8	S-FRAG %G8: Relay Chatter Event 390 (33-102 relay) (EDG C - Unrecoverable)
		1.00E+00	SRX08_AHUBTL-RDXD3	S-HEP G8: OPS FAILS TO VALVE IN N2 BOTTLES AFTER ACCUM DEPLETION (LATE; CONDITIO
		1.00E+00	SRX08_AHUBTL-RDXI3	S-HEP G8: OPERATORS FAIL TO VALVE IN N2 BOTTLES AFTER ACCUMULATOR DEPLETION (EAR
		1.00E+00	SRX08_AHU--CADDXD3	S-HEP G8: OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 2 INS 'B' - DELAYED; CONDITIO
		1.00E+00	SRX08_AHU--CADDXI3	S-HEP G8: OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 2 INS 'B'
		1.00E+00	SRX08_EHUCWGCNDXI0	S-HEP G8: CONOWINGO OPERATOR FAILS TO ENERGIZE SBO LINE TO PEACH BOTTOM (EXECUTI
		1.00E+00	SRX08_EHULS-ACDXI3	S-HEP G8: INITIAL LOAD SHED PER SE-11, ATT. T (STEPS FOR RCIC)
		1.00E+00	SRX08_EHU-SE11DXI0	S-HEP G8: OPERATOR CROSS TIES 4KV EMERGENCY BUSES
		1.00E+00	SRX08_MHUSE11WDXI3	S-HEP G8: OPERATOR FAILS TO IMPLEMENT SE-11 ATTACHMENT W FOR CRD COOLING (LOOP C
		9.54E-01	ZZAVFACTOR	PLANT AVAILABILITY FACTOR

5.5 SLERF Results

The seismic PRA performed for PBAPS shows that the point estimate mean seismic LERF is 4.0E-06/yr for Unit 2 and 4.1E-06/yr for Unit 3. The Unit 2 Seismic LERF of 4.0E-06/yr is calculated with a single top CAFTA model at a truncation that ranges from 5E-08/yr to 1E-12/yr. The seismic LERF of 4.1E-06/yr represents approximately 20% of the seismic CDF of 2.1E-05/yr.

The Unit 3 SLERF is 4.1E-06/yr, which is within 4% of the Unit 2 SLERF. Given the similarities in the Unit 2 and Unit 3 SCDF and SLERF values, the remainder of this section focuses on the Unit 2 results, except as noted. In general, PBAPS Unit 2 and Unit 3 are symmetrical. In addition, the fragility analysis supports that the dominant risk contributors represent correlated failures that impact both units (e.g., failure to SCRAM, 125 VDC battery racks).

Important Seismic Initiating Event Contributors

Table 5.5-1 summarizes the LERF contributors by seismic initiating event. Figure 5.5-1 displays the results of Table 5.5-1 in graphical pie chart form, i.e., the LERF contributors by initiating event. Figure 5.5-2 shows the initiating event contribution in the form of a bar graph.

As can be seen from the graphical display, the seismic initiators %G5, %G6, %G7, and %G8 are the dominant seismic risk contributors. These initiators span the range from 0.50g to >0.9g. Their combined contribution is approximately 90% of the seismic LERF. Seismic hazard interval initiator %G7 contributes less to SLERF than %G8 because the initiator frequency of %G7 is lower than that of %G8 (i.e., %G7 is a bounded hazard interval and %G8 is the unbounded final hazard interval).

Conditional Large Early Release Probability (CLERP) values were also calculated for the initiators. These CLERP values are displayed in Figure 5.5-3. Figure 5.5-3 shows the CLERP for the %G8 initiators (>0.9g) as approximately 0.34 based on eliminating some of the conservatisms associated with the Level 2 (LERF) accident sequence progression modeling.

The Unit 3 SLERF contributors by seismic initiating event are similar to those shown for Unit 2.

Important Contributors to Large Early Release Frequency

Table 5.5-2 provides the Unit 2 SLERF Fussell-Vesely (FV) importance measures for SSC fragilities. The SLERF FV risk importance values are calculated in the same manner as that discussed in Section 5.4 for SCDF FV values, except that the SLERF cutset results are used.

Failure to scram (ATWS) scenarios are also a significant contributor to SLERF because of the relatively low Am value for seismic induced failure to scram and the modeling of ATWS scenarios in the Level 2 SPRA. The Level 2 SPRA is based on the Level 2 FPIE PRA model, which incorporates potentially conservative assumptions for ATWS mitigation.

The top 5 contributors to Unit 2 SLERF FV are as follows:

Normal offsite power (FV = 0.902)

Normal offsite power is expected to have a high FV because there is a high probability for the seismic event to fail offsite power.

SCRAM (RPV Internals) (FV = 0.210)

The SCRAM (RPV Internals) has a high-risk impact because unmitigated failure to SCRAM events result in significant hydrodynamic loads on the containment. Failure to SCRAM events are modeled in the base PBAPS FPIE PRA model to have a high likelihood of leading to early containment failure and a Large Early Release.

125 VDC Batteries (FV = 0.126)

The 125 VDC batteries have a high-risk impact because loss of all 125 VDC is modeled to result in unavailability of all EDGs, HPCI, and RCIC (i.e., short term station blackout event). Loss of all RPV makeup leads to early RPV failure, Mark I shell liner failure, and a Large Early Release.

Conowingo Hydroelectric Plant (FV = 5.11E-02)

The Conowingo Hydroelectric Plant has a high-risk impact because it provides alternate offsite AC power to PBAPS.]

Break Outside Containment (FV = 3.87E-02)

Break Outside Containment (BOC) represents a failure of RCS piping outside the primary containment. Failure to isolate the BOC results in a containment bypass event and is modeled to lead directly to core damage and a large, early release. Isolation of the piping is not credited, which may be conservative.

It is noted that four (4) additional Unit 2 SPRA fragility groups are calculated based on "representative" fragility calculations and have SLERF FV > 5E-3, which is defined as "risk significant" per the ASME/ANS PRA Standard [6]. The four (4) fragility groups are as follows:

Primary Containment Isolation (Inboard and Outboard MSIVs) (FV = 2.45E-02)

Failure of both the inboard and outboard isolation valves to close is assumed to represent failure of containment isolation and lead directly to a Large Early Release for all Level 1 core damage scenarios. The current fragility is based on a conservative calculation because previous quantification results identified that the MSIVs were not risk significant. The MSIVs increased above the risk significant threshold for the final quantification based on changes to the final SPRA model (e.g., reduce truncation limit to support LERF model convergence). A review of the current conservative fragility calculation identifies that the valve capacity could be increased by approximately 67%, which would significantly reduce the risk importance of the MSIVs. However, increasing the valve capacity would reduce the SLERF by 2% at most and is judged to not provide additional significant risk insights. Therefore, the use of the conservative fragility calculation for the MSIVs is judged to not have an adverse impact on the PBAPS SPRA results.

Panel 20C32 (U2 Engineering Sub Systems I Relay Cabinet) (FV = 1.42E-02)

Seismic failure of relay panel 20C32 is modeled to result in unavailability of EDG A and C amongst other systemic impacts based on information from the PBAPS Fire PRA model. The fragility is based on a site specific “representative” calculation. However, increasing the control panel capacity would reduce the SCDF by less than 2% and is judged to not provide additional significant risk insights. Therefore, the use of the conservative fragility calculation for relay panel 20C32 is judged to not have an adverse impact on the PBAPS SPRA results.

Panel 20C39 (U2 HPCI Relay Panel) (FV = 1.15E-02)

Seismic failure of relay panel 20C39 is modeled to result in unavailability of Unit 2 HPCI based on information from the PBAPS Fire PRA model. The fragility is based on a site specific “representative” calculation. However, increasing the control panel capacity would reduce the SCDF by up to approximately 1% and is judged to not provide additional significant risk insights. Therefore, the use of the conservative fragility calculation for relay panel 20C39 is judged to not have an adverse impact on the PBAPS SPRA results.

Panel 20C33 (U2 Engineering Sub Systems II Relay Cabinet) (FV = 7.63E-03)

Seismic failure of relay panel 20C33 is modeled to result in unavailability of EDG B and D amongst other systemic impacts based on information from the PBAPS Fire PRA model. The fragility is based on a site specific “representative” calculation. However, increasing the control panel capacity would reduce the SCDF by less than 1% and is judged to not provide additional significant risk insights. Therefore, the use of the conservative fragility calculation for relay panel 20C33 is judged to not have an adverse impact on the PBAPS SPRA results

The last four (4) fragility groups identified above that are based on site specific representative fragilities and were not risk significant based on the quantitative results previously provided to the PBAPS SPRA peer review team.

Table 5.5-3 provides the Unit 3 SLERF Fussell-Vesely (FV) importance measures for SSC fragilities. The Unit 3 SLERF FV contributors are similar to the Unit 2 contributors with the exception of the addition of fragility group S-DCBS10-(seismic correlated failure of DC Panel 30D11) to the Top 10 Unit 3 contributors because seismic failure of DC Panel 30D11 has a more risk significant impact on Unit 3 SLERF due to resultant unavailability of HPCI for Unit 3.

The current fragility for DC Panel 30D11 is based on a conservative site specific representative calculation because previous quantification results identified that the DC Panel 30D11 was not risk significant. DC Panel 30D11 increased above the risk significant threshold for the final quantification based on changes to the final SPRA model (e.g., reduce truncation limit to support LERF model convergence). Given that the FV is slightly above 1%, increasing the panel capacity would reduce the SLERF by less than 1% and is judged to not provide additional significant risk insights. For example, other seismic induced failures of Unit 3 HPCI (e.g., relay chatter failures) have comparable fragility values and would result in similar cutsets as seismic failure of DC Panel 30D11. Therefore, the use of the conservative fragility calculation for DC Panel 30D11 is judged to not have an adverse impact on the PBAPS SPRA results.

Table 5.5-4 provides the Unit 2 SLERF FV importance measures for the operator actions. Similar to the total FV for the SSC fragilities, the total FV for the operator actions is the sum of the individual FV values for the entire range of the hazard intervals.

The top five (5) operator action contributors to the Unit 2 SLERF FV are examined below:

- OPERATOR FAILS TO MANUALLY START RCIC (BLACK START) (FV = 5.46E-02)
- OPERATOR FAILS TO MITIGATE RELAY CHATTER FOR 4KV BUSES (FV = 3.04E-02)
- OPERATOR CROSS TIES 4KV EMERGENCY BUSES (FV = 2.71E-02)
- OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 2 INS 'B' (FV = 2.36E-02)
- OPERATOR FAILS TO ALIGN FLEX GENERATOR TO LC E124 OR E324 (FV = 2.35E-02)

The above operator actions are slightly different than the top operator actions contributing to Unit 2 SCDF FV due to differences in the types of dominant accident scenarios contributing to either SCDF or SLERF. The significant operator actions contributing to SLERF involve short term Station Blackout scenarios and failure of the operators to align early RPV makeup.

The quantitative results showed that there were no SSCs with significant non-seismic failure contribution to SLERF (i.e., no random failures to start, run, etc. with FV > 5E-3).

However, the Level 2 SPRA model includes a number of non-seismic failures related to phenomenological issues that are based on information from the FPIE PBAPS PRA model. Based on the SLERF FV importance measures, significant contributors from non-seismic failure events include the following:

- LERF Not Precluded Due to SORVs / Timing (2OPPH-LERF-NP-F--) (FV = 0.49). This basic event models phenomenological issues associated with the Level 2 accident progression resulting in a LERF end state.
- CLASS II CANDIDATE FOR EARLY RELEASE (B--OPDHR-EAL2F--) (FV = 0.07). This basic event models the probability that a General Emergency is not declared in sufficient time during a long-term Loss of Containment Heat event (>20 hours).

Table 5.5-5 provides the Unit 3 SLERF Fussell-Vesely (FV) importance measures for the operator actions. The Unit 3 SLERF FV contributors are similar to the Unit 2 contributors.

<u>Top 10 SLERF Cutsets</u>

Table 5.5-6 provides the Top 10 Unit 3 SLERF cutsets because the Unit 3 results are slightly more limiting for the PBAPS SPRA model. The Top 10 Unit 2 SLERF cutsets are generally similar to the Unit 3 Top 10 SLERF cutsets so the Unit 2 cutsets are not explicitly provided. Similar to the top SCDF cutsets, the top SLERF cutsets involve a %G8 seismic initiating event (seismic magnitude >0.9g). A discussion of the top 10 cutsets is as follows:

Cutset #1 (4.80E-7/yr): Seismic induced LOOP with EDGs available (i.e., Station Blackout). Seismic correlated relay chatter of the 150/151 relays results in opening 4KV bus breakers supporting ECCS equipment, including all EDG cooling. The operators fail to manually trip the EDGs in the Main Control Room with the jumper cables in sufficient time to prevent assumed loss of the EDGs due to overheating. In addition, operator failure to load shed to extend 125 VDC battery life results in long term unavailability of HPCI and RCIC for RPV makeup and long term unavailability of SRV control for RPV depressurization. In addition, the operators at the Conowingo Dam fail to align alternate AC power to PBAPS. This leads to a core damage event during a loss of offsite power (Accident Class 1B) with the RPV at high pressure. The operator fails to adequately control RPV water level post core damage. The Level 2 accident progression conditions (e.g., no induced SORV) fail to prevent a LERF from occurring. No recovery of RPV makeup results in RPV failure and Mark I shell liner failure, resulting in a Large Early Release.

Cutset #2 (4.80E-7/yr): Cutset #2 is similar to Cutset #1 but involves a different operator action to prevent aligning alternate AC power to PBAPS. Cutset #2 involves failure of the operators at PBAPS to align power from the SBO line from Conowingo Dam to the emergency 4KV busses.

Cutset #3 (4.73E-7/yr): Cutset #3 is similar to Cutset #1 but involves a different failure mode to prevent aligning alternate AC power to PBAPS. Cutset #3 involves seismic induced failure of the Conowingo Dam Hydroelectric plant to provide alternate AC power. The fragility for failure of the Conowingo Dam to provide alternate AC power is assumed to be the same as the industry value for normal offsite AC power (i.e., Am=0.3g).

Cutset #4 (4.65E-7/yr): Cutset #4 is similar to Cutset #1 except the Station Blackout is caused by seismic induced correlated failure of the 125 VDC battery racks. Failure of the 125 VDC batteries results in unavailability of all EDGs for emergency AC power, HPCI and RCIC for early RPV makeup, and SRV control for RPV depressurization. In addition, the operators fail to locally, manually align RCIC (i.e., Black Start RCIC). This leads to a core damage event during a loss of offsite power (Accident Class 1B) with the RPV at high pressure. The operator fails to adequately control RPV water level post core damage. The Level 2 accident progression conditions (e.g., no induced SORV) fail to prevent a LERF from occurring. No recovery of RPV makeup results in RPV failure and Mark I shell liner failure, resulting in a Large Early Release.

Cutset #5 (4.44E-7/yr): Cutset #5 is similar to Cutset #4 but involves a different failure mode for Black Start RCIC. Seismic induced failure of the CSTs results in unavailability of the primary suction source for RCIC. Given seismic induced failure of the 125 VDC batteries, automatic swap-over of the RCIC suction from the CST to the suppression pool is precluded. In addition, operator action to manually re-align RCIC suction from the CST to the suppression pool is conservatively not credited during a Station Blackout event as part of Black Start RCIC. This conservatism is judged not to have a significant impact on the overall SLERF results and the risk importance measures.

Cutset #6 (4.39E-7/yr): Cutset #6 is similar to Cutset #1 but involves a different failure mode for unavailability of the EDGs. For Cutset #6, unavailability of all EDGs is due to seismic correlated failure of Main Control Room Panels 00C29(A-D).

Cutset #7 (4.39E-7/yr): Cutset #7 is similar to Cutset #6 but involves a different operator action for failure to load shed to extend 125 VDC battery life per procedure SE-11.

Cutset #8 (4.39E-7/yr): Cutset #8 is similar to Cutset #6 but involves operator failure to locally, manually align RCIC (i.e., Black Start RCIC).

Cutset #9 (4.33E-7/yr): Seismic induced LOOP with selected EDGs unavailable. Seismic induced correlated chatter of the 52B-151N relays results in unavailability of EDG A and EDG D and the operators do not recover from the relay chatter event. HPCI and RCIC are unavailable long term due to a combination of loss of EDG power to the battery chargers and operator failure to load shed to extend 125 VDC battery life. The operators do not align backup pneumatic supply to the ADS SRVs. In addition, CRD is unavailable due to unavailability of pneumatic supplies to support maximizing CRD flow for adequate RPV makeup. In addition, the operators at the Conowingo Dam fail to align alternate AC power to PBAPS. This leads to a core damage event during a loss of offsite power (Accident Class 1B) with the RPV at high pressure. The operator does not control RPV water level post core damage. The Level 2 accident progression conditions (e.g., no induced SORV) fail to prevent a LERF from occurring. No recovery of RPV makeup results in RPV failure and Mark I shell liner failure, resulting in a Large Early Release.

Cutset #10 (4.33E-7/yr): Cutset #10 is similar to Cutset #9 but involves a different operator action to prevent aligning alternate AC power to PBAPS. Cutset #10 involves failure of the operators at PBAPS to align power from the SBO line from Conowingo Dam to the emergency 4KV busses.

Although the cutsets may be conservative because of the many HEPs set to 1.0, the cutsets appear valid based on the PBAPS SPRA modeling assumptions.

A review of sample cutsets from each decade of quantification results did not identify any improper cutsets.

SLERF Accident Class Contributors

The dominant Level 2 accident class contributors to the PBAPS SLERF include the following:

- Class 1B (Station Blackout) – 56%
- Class 4A (ATWS with failure of reactivity control) – 23%
- Class 2 (Loss of Containment Heat Removal with successful Containment Venting) – 8% [Note: The contribution from Class 2 subclasses A, F, and L are combined and presented as Class 2.]
- Class 1C (ATWS with failure of RPV makeup) – 5%

As discussed in Section 5.4, Class 1B (Station Blackout) has the highest contribution to the PBAPS Level 1 SCDF. In addition, for one of the dominant Class 1B risk contributors (i.e., loss of 125 VDC batteries), the EDGs and HPCI are unavailable without credit for manual recovery actions. Given no 125 VDC power, manual operation of RCIC (i.e., black start of RCIC) is credited. The time to core damage is approximately 1 hour. No recovery of RPV makeup results in RPV failure and Mark I shell liner failure, resulting in a Large Early Release. For the dominant scenarios, insufficient time is available to support alignment of FLEX equipment for mitigation of CDF or LERF.

The degree of potential conservatisms in these types of Level 2 sequences is discussed in sensitivity cases in Section 5.7 of this report.

Class 4A (ATWS with failure of reactivity control) accidents have a high contribution to SLERF due to the relatively low fragility for seismic induced failure to scram (Am = 1.35g) combined with failure of adequate reactivity control, (e.g., seismic induced failure of 125 VDC batteries precludes injection of SLC). An ATWS with failure of reactivity control is modeled to result in containment failure due to overpressure from the high core power generation and assumed loss of all RPV makeup following containment failure. Recovery of RPV makeup is not credited in Level 2 PRA for ATWS scenarios, resulting in a Large Early release.

Contribution from Class 2 comprises of mainly Class 2F and Class 2A accidents. Class 2F (Loss of Containment Heat Removal with successful Containment Venting) is a long term scenario with delayed core damage (e.g., approximately 20 hours). For the PBAPS FPIE and SPRA model, a long term Loss of Containment Heat Removal scenario is postulated to lead to a LERF scenario if a General Emergency (GE) is not declared "early" based on the interpretation of the site specific Emergency Action Levels (EALs). If a GE is not declared sufficiently early, then core damage and radionuclide release could occur within a relatively short time frame such that adequate evacuation cannot be completed. A probability of 5E-2 is estimated for failure to declare a GE sufficiently early during a Loss of CHR scenario based on discussions with Exelon Emergency Planning personnel. The probability of 5E-2 is based on the PBAPS FPIE PRA model and used for the PBAPS SPRA model. The probability of 5E-2 is applied to all Class 2F cutsets in the Level 2 Class 2E (Early) Containment Event Tree (CET) for potentially leading to a LERF endstate. Class 2F accident class cutset have a high potential for leading to a LERF endstate because the containment is unisolated (e.g., successfully vented) or failed on high containment pressure.

Class 2A (Loss of Containment Heat Removal with failure of Containment Venting) is similar to Class 2F, but Containment Venting is unavailable (e.g., due to operator failure to initiate Containment Venting, failure to align backup pneumatic supply). Similar to Class 2F, a 5E-2 probability is applied to all Class 2A cutsets in the Class 2E (Early) CET for potentially leading to a LERF endstate.

Class 1C (ATWS with failure of RPV makeup) may contribute to SLERF due to failures of RPV level control (e.g., operator action) in the Level 1 PRA that lead to early core damage scenarios and potentially impact RPV level control in the Level 2 PRA. Loss of RPV makeup in the Level 2 SPRA leads to RPV failure and likely drywell shell liner failure.

TABLE 5.5-1
UNIT 2 SLERF CONTRIBUTORS BY SEISMIC HAZARD INTERVAL INITIATING EVENT

Seismic Hazard Interval ID	Description	Interval Frequency (/yr)	Interval LERF (/yr)	% of Total SLERF	Cumulative SLERF (/yr)
%G1	%G1 - Hazard Curve: PBAPS Hazard Curve - PGA Range: 0.05g to 0.2g	4.7E-04	2.0E-10	0%	2.0E-10
%G2	%G2 - Hazard Curve: PBAPS Hazard Curve - PGA Range: 0.2g to 0.3g	4.2E-05	3.5E-08	1%	3.6E-08
%G3	%G3 - Hazard Curve: PBAPS Hazard Curve - PGA Range: 0.3g to 0.4g	1.8E-05	1.2E-07	3%	1.6E-07
%G4	%G4 - Hazard Curve: PBAPS Hazard Curve - PGA Range: 0.4g to 0.5g	9.5E-06	2.4E-07	6%	4.0E-07
%G5	%G5 - Hazard Curve: PBAPS Hazard Curve - PGA Range: 0.5g to 0.6g	5.6E-06	4.8E-07	12%	8.8E-07
%G6	%G6 - Hazard Curve: PBAPS Hazard Curve - PGA Range: 0.6g to 0.75g	4.7E-06	7.8E-07	20%	1.7E-06
%G7	%G7 - Hazard Curve: PBAPS Hazard Curve - PGA Range: 0.75g to 0.9g	2.7E-06	6.5E-07	16%	2.3E-06
%G8	%G8 - Hazard Curve: PBAPS Hazard Curve - PGA Range: > 0.9g	5.3E-06	1.7E-06	42%	4.0E-06

Notes to Table 5.5-1:

A sensitivity study has been performed to evaluate the risk impact of subdividing the %G8 seismic hazard interval into six (6) intervals with the highest bin at > 2.3g. The CLERP for the bin at > 2.3g is assumed to be 1.0 and the Unit 3 SLERF is calculated to increase to 4.83E-06/yr. Refer to sensitivity case 1d in Section 5.7.

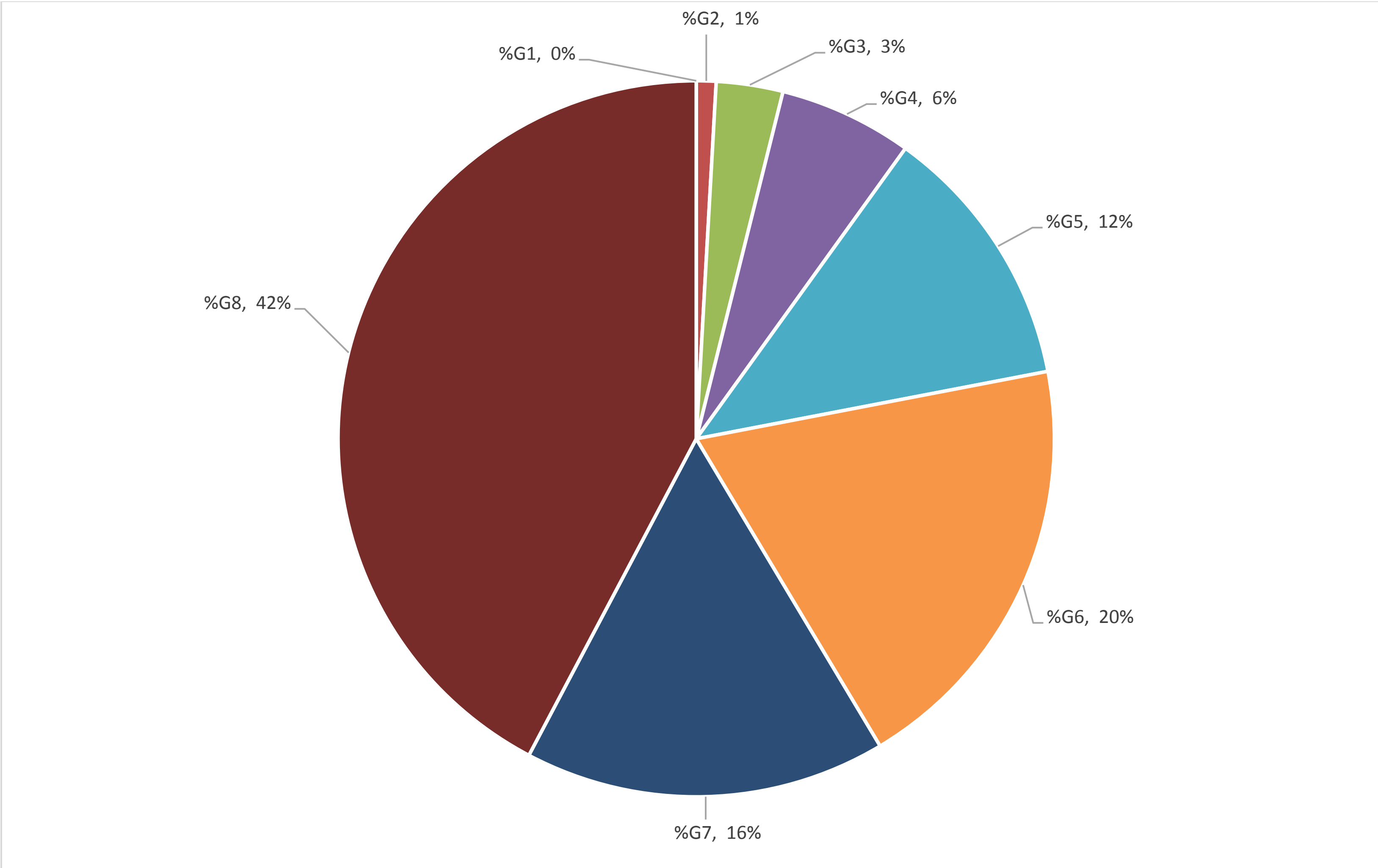

Figure 5.5-1

PBAPS SPRA UNIT 2 SLERF BY HAZARD INTERVAL INITIATING EVENT

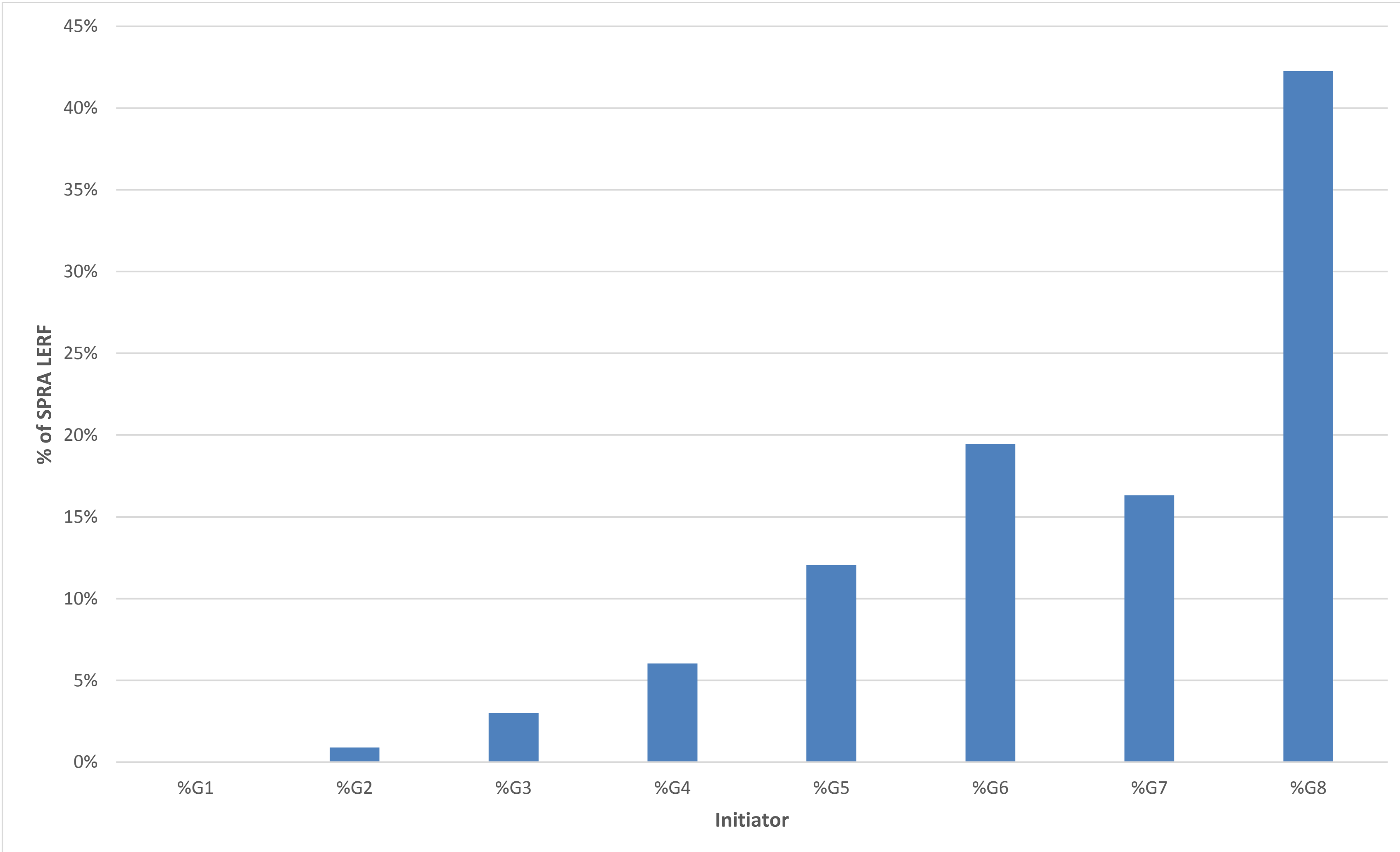

Figure 5.5-2

PBAPS SPRA UNIT 2 SLERF BY HAZARD INTERVAL INITIATING EVENT

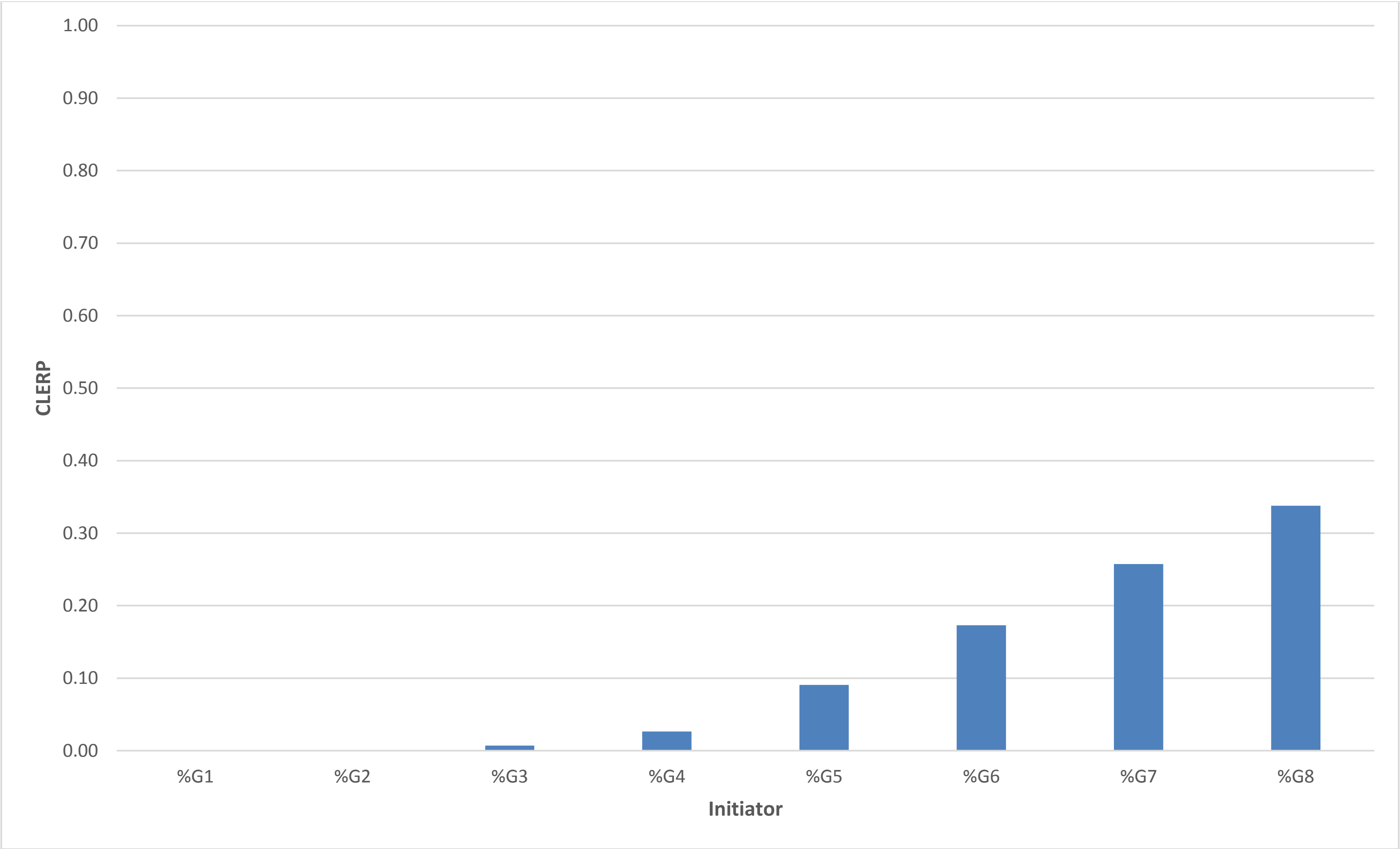

Figure 5.5-3

PBAPS SPRA UNIT 2 CLERP BY HAZARD INTERVAL INITIATING EVENT

TABLE 5.5-2

UNIT 2 SLERF FUSSELL-VESELY IMPORTANCE MEASURES FOR SSC FRAGILITIES

FRAGILITY GROUP ID	FRAGILITY GROUP DESCRIPTION	FV TOTAL	Am (g)	βr	βu	Failure Mode	Fragility Method
OSP	Offsite Power	9.02E-01	0.3	0.3	0.45	Functional	Representative
SCRAM	RPV Internals (Scram)	2.10E-01	1.35	0.28	0.32	Anchorage	CDFM
S-DCBT1-	DC Batteries 2(A-D)D01, 3(A-D)D01	1.26E-01	0.73	0.28	0.52	Anchorage	SOV
S-CNWG2-	Conowingo Hydroelectric Plant (OSP)	5.11E-02	0.3	0.3	0.45	Functional	Representative
BOC	Break Outside Containment	3.87E-02	2.69	0.35	0.4	Anchorage	CDFM
SML	Seismic Induced Medium LOCA	3.12E-02	2.69	0.35	0.4	Anchorage	CDFM
S-CEPA1-	Panel 20C003, 20C004C, 30C003, 30C004C, 00C29(A-D)	2.65E-02	0.82	0.28	0.37	Anchorage	SOV
S-PCI2	Primary Containment Isolation (Inboard and Outboard MSIVs)	2.45E-02	2.18	0.24	0.32	Functional	Representative
S-CEPA7-	Panel 20C32 (U2 Engineering Sub Systems I Relay Cabinet)	1.42E-02	0.83	0.24	0.32	Functional	Representative
S-CNCT1-	Condensate Storage Tank 20T010, 30T010	1.38E-02	0.5	0.24	0.32	Anchorage	CDFM
S-SGTK1-	SGIG Nitrogen Tank	1.16E-02	0.78	0.24	0.26	Anchorage	CDFM
S-CEPA6-	Panel 20C39 (U2 HPCI Relay Panel)	1.15E-02	0.83	0.24	0.32	Functional	Representative
S-CC190A-	Correlated Relay Chatter Group 190A (52B-151N relays) (EDGs A and D - Recoverable)	9.18E-03	0.82	0.3	0.39	Functional	SOV
S-CEPA8-	Panel 20C33 (U2 Engineering Sub Systems II Relay Cabinet)	7.63E-03	0.83	0.24	0.32	Functional	Representative
S-CC138-	Relay Chatter Group 138 (150G relay) (4KV Bus 20A15 - Recoverable)	7.21E-03	0.78	0.3	0.43	Functional	SOV
S-DCBS6-	DC Panel 2(A-D)D17, 3AD17, 3CD17, 3DD17	6.20E-03	1.46	0.28	0.44	Functional	SOV

TABLE 5.5-3

UNIT 3 SLERF FUSSELL-VESELY IMPORTANCE MEASURES FOR SSC FRAGILITIES

FRAGILITY GROUP ID	FRAGILITY GROUP DESCRIPTION	FV TOTAL	Am (g)	βr	βu	Failure Mode	Fragility Method
OSP	Offsite Power	9.04E-01	0.3	0.3	0.45	Functional	Representative
SCRAM	RPV Internals (Scram)	2.02E-01	1.35	0.28	0.32	Anchorage	CDFM
S-DCBT1-	DC Batteries 2(A-D)D01, 3(A-D)D01	1.02E-01	0.73	0.28	0.52	Anchorage	SOV
S-CEPA1-	Panel 20C003, 20C004C, 30C003, 30C004C, 00C29(A-D)	5.23E-02	0.82	0.28	0.37	Anchorage	SOV
S-CNWG2-	Conowingo Hydroelectric Plant (OSP)	4.92E-02	0.3	0.3	0.45	Functional	Representative
BOC	Break Outside Containment	3.74E-02	2.69	0.35	0.4	Anchorage	CDFM
SML	Seismic Induced Medium LOCA	3.01E-02	2.69	0.35	0.4	Anchorage	CDFM
S-DCBS4-	DC Panel 20D24, 30D21	2.50E-02	0.86	0.28	0.52	Anchorage	SOV
S-PCI2	Primary Containment Isolation (Inboard and Outboard MSIVs)	2.37E-02	2.18	0.24	0.32	Functional	Representative
S-CNCT1-	Condensate Storage Tank 20T010, 30T010	1.49E-02	0.5	0.24	0.32	Anchorage	CDFM
S-DCBS10-	250 VDC Bus 30D11	1.38E-02	0.51	0.24	0.32	Anchorage	Representative
S-SGTK1-	SGIG Nitrogen Tank	1.07E-02	0.78	0.24	0.26	Anchorage	CDFM
S-CC190A-	Correlated Relay Chatter Group 190A (52B-151N relays) (EDGs A and D - Recoverable)	8.21E-03	0.82	0.3	0.39	Functional	SOV

TABLE 5.5-4

PBAPS UNIT 2 SLERF FUSSELL-VESELY IMPORTANCE MEASURES FOR SSC OPERATOR ACTIONS

OPERATOR ACTION ID	OPERATOR ACTION DESCRIPTION	FV TOTAL
RHUBLKSTDXI2	OPERATOR FAILS TO MANUALLY START RCIC (BLACK START) - SEISMIC PRA VERSION	5.46E-02
EHURLY4KDXI2	OPERATOR FAILS TO MITIGATE RELAY CHATTER FOR 4KV BUSES (SEISMIC)	3.04E-02
EHU-SE11DXI0	OPERATOR CROSS TIES 4KV EMERGENCY BUSES	2.71E-02
AHU--CADDXI2	OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 2 INS 'B'	2.36E-02
QHUFXL13DXI2	OPERATOR FAILS TO ALIGN FLEX GENERATOR TO LC E124 OR E324	2.35E-02
AHU--CADDXD2	OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 2 INS 'B' - DELAYED; CONDITIONAL	2.16E-02
AHUBTL-RDXI2	OPERATORS FAIL TO VALVE IN N2 BOTTLES AFTER ACCUMULATOR DEPLETION (EARLY)	2.15E-02
AHUBTL-RDXD2	OPS FAILS TO VALVE IN N2 BOTTLES AFTER ACCUM DEPLETION (LATE; CONDITIONAL)	1.97E-02
QHULS-ACDXI2	OPERATOR FAILS TO PERFORM DEEP DC LOAD SHED	1.57E-02
2CZOP-SLCLWL-H--	OPERATORS FAIL TO INJECT SLC WITH BORON ON LOW WATER LEVEL	1.38E-02
RHUCSTSPDXI2	OPERATOR FAILS TO SWAP RCIC SUCTION FROM CST TO SUPPRESSION POOL	1.36E-02
EHULS-ACDXI2	OPS FAIL TO PERFORM SE-11 LOAD SHED FOR FLEX (single unit- RCIC only)	1.12E-02
EHUATT-TDXI0	OPS FAIL TO PERFORM SE-11 LOAD SHED FOR FLEX (single unit, both divisions)	9.64E-03
EHURLYDGDXI2	OPERATOR FAILS TO MITIGATE RELAY CHATTER for EDGs (SEISMIC)	9.18E-03

Notes to Table:

(1) This table covers independent and dependent post-initiator HEPs and their risk contribution; however, if dependent HEPs do not show up in this table that is because their FV value is below 5E-3.
(2) The independent post-initiator HEP FV values presented in this table do not include the risk contribution from the independent HEPs appearing in dependent HEPs.

TABLE 5.5-5

PBAPS UNIT 3 SLERF FUSSELL-VESELY IMPORTANCE MEASURES FOR SSC OPERATOR ACTIONS

OPERATOR ACTION ID	OPERATOR ACTION DESCRIPTION	FV TOTAL
RHUBLKSTDXI3	OPERATOR FAILS TO MANUALLY START RCIC (BLACK START) - SEISMIC PRA VERSION	9.57E-02
AHUBTL-RDXI3	OPERATORS FAIL TO VALVE IN N2 BOTTLES AFTER ACCUMULATOR DEPLETION (EARLY)	6.41E-02
AHUBTL-RDXD3	OPS FAILS TO VALVE IN N2 BOTTLES AFTER ACCUM DEPLETION (LATE; CONDITIONAL)	5.70E-02
AHU--CADDXI3	OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 3 INS 'B'	3.34E-02
AHU--CADDXD3	OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 3 INS 'B' - DELAYED; CONDITIONAL	3.15E-02
EHU-SE11DXI0	OPERATOR CROSS TIES 4KV EMERGENCY BUSES	2.13E-02
EHURLY4KDXI3	OPERATOR FAILS TO MITIGATE RELAY CHATTER FOR 4KV BUSES (SEISMIC)	1.83E-02
3CZOP-SLCLWL-H--	OPERATORS FAIL TO INJECT SLC WITH BORON ON LOW WATER LEVEL	1.57E-02
RHUCSTSPDXI3	OPERATOR FAILS TO SWAP RCIC SUCTION FROM CST TO SUPPRESSION POOL	1.47E-02
QHUFXL13DXI3	OPERATOR FAILS TO ALIGN FLEX GENERATOR TO LC E134 AND LC E334	1.34E-02
EHULS-ACDXI3	INITIAL LOAD SHED PER SE-11, ATT. T (STEPS FOR RCIC)	1.10E-02
EHURLYDGDXI3	OPERATOR FAILS TO MITIGATE RELAY CHATTER for EDGs (SEISMIC)	9.60E-03
QHULS-ACDXI3	DEEP DC LOAD SHED WHEN ELAP DECLARED (STEPS FOR RCIC)	9.22E-03

Notes to Table:

(1) This table covers independent and dependent post-initiator HEPs and their risk contribution; however, if dependent HEPs do not show up in this table that is because their FV value is below 5E-3.

(2) The independent post-initiator HEP FV values presented in this table do not include the risk contribution from the independent HEPs appearing in dependent HEPs.

TABLE 5.5-6

PBAPS UNIT 3 TOP 10 SEISMIC LERF CUTSETS

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
1	4.80E-07	5.25E-06	%G8	Seismic Initiating Event (>0.9g)
		1.00E+00	1-CL-1B	CLASS 1B
		1.00E+00	1-SEQ-LP2-018	SEQUENCE LP2-018
		1.00E+00	2-ES-H/E	CLASS H/E
		1.00E+00	2-SEQ-1BE-081	SEQUENCE 1BE-081
		1.50E-01	3OPPH-LERF-NP-F--	LERF Not Precluded Due to SORVs / Timing
		9.00E-01	3RXRX-FRECINJH--	OPERATOR FAILS TO RECOVER INJECTION BEFORE RPV MELT
		9.86E-01	OSP-C-%G8	SEISMIC FRAGILITY FOR %G8: Offsite Power
		7.19E-01	S-CC136A--C-%G8	S-FRAG %G8: Relay Chatter Group 136A (150/151 relays) (All EDGs - Recoverable)
		1.00E+00	SRX08_EHUCWGCNDXI0	S-HEP G8: CONOWINGO OPERATOR FAILS TO ENERGIZE SBO LINE TO PEACH BOTTOM (EXECUTI
		1.00E+00	SRX08_EHULS-ACDXI3	S-HEP G8: INITIAL LOAD SHED PER SE-11, ATT. T (STEPS FOR RCIC)
		9.54E-01	ZZAVFACTOR	PLANT AVAILABILITY FACTOR
		1.00E+00	SRX08_EHURLDGTDXI3	S-HEP G8: OP FAILS TO TRIP EDGs TO MITIGATE CHATTER OF 150/151 RELAYS - SEISMIC
2	4.80E-07	5.25E-06	%G8	Seismic Initiating Event (>0.9g)

TABLE 5.5-6

PBAPS UNIT 3 TOP 10 SEISMIC LERF CUTSETS

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
		1.00E+00	1-CL-1B	CLASS 1B
		1.00E+00	1-SEQ-LP2-018	SEQUENCE LP2-018
		1.00E+00	2-ES-H/E	CLASS H/E
		1.00E+00	2-SEQ-1BE-081	SEQUENCE 1BE-081
		1.50E-01	3OPPH-LERF-NP-F--	LERF Not Precluded Due to SORVs / Timing
		9.00E-01	3RXRX-FRECINJH--	OPERATOR FAILS TO RECOVER INJECTION BEFORE RPV MELT
		9.86E-01	OSP-C-%G8	SEISMIC FRAGILITY FOR %G8: Offsite Power
		7.19E-01	S-CC136A--C-%G8	S-FRAG %G8: Relay Chatter Group 136A (150/151 relays) (All EDGs - Recoverable)
		1.00E+00	SRX08_EHUCWGPBDXI0	S-HEP G8: PEACH BOTTOM OPERATOR FAILS TO ALIGN CONOWINGO SBO LINE TO EMERGENCY 4
		1.00E+00	SRX08_EHULS-ACDXI3	S-HEP G8: INITIAL LOAD SHED PER SE-11, ATT. T (STEPS FOR RCIC)
		9.54E-01	ZZAVFACTOR	PLANT AVAILABILITY FACTOR
		1.00E+00	SRX08_EHURLDGTDXI3	S-HEP G8: OP FAILS TO TRIP EDGs TO MITIGATE CHATTER OF 150/151 RELAYS - SEISMIC
3	4.73E-07	5.25E-06	%G8	Seismic Initiating Event (>0.9g)
		1.00E+00	1-CL-1B	CLASS 1B

TABLE 5.5-6

PBAPS UNIT 3 TOP 10 SEISMIC LERF CUTSETS

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
		1.00E+00	1-SEQ-LP2-018	SEQUENCE LP2-018
		1.00E+00	2-ES-H/E	CLASS H/E
		1.00E+00	2-SEQ-1BE-081	SEQUENCE 1BE-081
		1.50E-01	3OPPH-LERF-NP-F--	LERF Not Precluded Due to SORVs / Timing
		9.00E-01	3RXRX-FRECINJH--	OPERATOR FAILS TO RECOVER INJECTION BEFORE RPV MELT
		9.86E-01	OSP-C-%G8	SEISMIC FRAGILITY FOR %G8: Offsite Power
		7.19E-01	S-CC136A--C-%G8	S-FRAG %G8: Relay Chatter Group 136A (150/151 relays) (All EDGs - Recoverable)
		9.86E-01	S-CNWG2--C-%G8	S-FRAG %G8: Conowingo Hydroelectric Plant (OSP)
		1.00E+00	SRX08_EHULS-ACDXI3	S-HEP G8: INITIAL LOAD SHED PER SE-11, ATT. T (STEPS FOR RCIC)
		9.54E-01	ZZAVFACTOR	PLANT AVAILABILITY FACTOR
		1.00E+00	SRX08_EHURLDGTDXI3	S-HEP G8: OP FAILS TO TRIP EDGs TO MITIGATE CHATTER OF 150/151 RELAYS - SEISMIC
4	4.65E-07	5.25E-06	%G8	Seismic Initiating Event (>0.9g)
		1.00E+00	1-CL-1B	CLASS 1B
		1.00E+00	1-SEQ-LP2-030	SEQUENCE LP2-030

TABLE 5.5-6

PBAPS UNIT 3 TOP 10 SEISMIC LERF CUTSETS

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
		1.00E+00	2-ES-H/E	CLASS H/E
		1.00E+00	2-SEQ-1BE-081	SEQUENCE 1BE-081
		1.50E-01	3OPPH-LERF-NP-F--	LERF Not Precluded Due to SORVs / Timing
		9.00E-01	3RXRX-FRECINJH--	OPERATOR FAILS TO RECOVER INJECTION BEFORE RPV MELT
		9.86E-01	OSP-C-%G8	SEISMIC FRAGILITY FOR %G8: Offsite Power
		6.97E-01	S-DCBT1--C-%G8	S-FRAG %G8: DC Batteries 2(A-D)D01, 3(A-D)D01
		1.00E+00	SRX08_RHUBLKSTDXI3	S-HEP G8: OPERATOR FAILS TO MANUALLY START RCIC (BLACK START) - SEISMIC PRA VERS
		9.54E-01	ZZAVFACTOR	PLANT AVAILABILITY FACTOR
5	4.44E-07	5.25E-06	%G8	Seismic Initiating Event (>0.9g)
		1.00E+00	1-CL-1B	CLASS 1B
		1.00E+00	1-SEQ-LP2-030	SEQUENCE LP2-030
		1.00E+00	2-ES-H/E	CLASS H/E
		1.00E+00	2-SEQ-1BE-081	SEQUENCE 1BE-081
		1.50E-01	3OPPH-LERF-NP-F--	LERF Not Precluded Due to SORVs / Timing

TABLE 5.5-6

PBAPS UNIT 3 TOP 10 SEISMIC LERF CUTSETS

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
		9.00E-01	3RXRX-FRECINJH--	OPERATOR FAILS TO RECOVER INJECTION BEFORE RPV MELT
		9.86E-01	OSP-C-%G8	SEISMIC FRAGILITY FOR %G8: Offsite Power
		9.56E-01	S-CNCT1--C-%G8	S-FRAG %G8: Condensate Storage Tank 20T010, 30T010
		6.97E-01	S-DCBT1--C-%G8	S-FRAG %G8: DC Batteries 2(A-D)D01, 3(A-D)D01
		1.00E+00	SRX08_RHUCSTSPDXI3	S-HEP G8: OPERATOR FAILS TO SWAP RCIC SUCTION FROM CST TO SUPPRESSION POOL
		9.54E-01	ZZAVFACTOR	PLANT AVAILABILITY FACTOR
6	4.39E-07	5.25E-06	%G8	Seismic Initiating Event (>0.9g)
		1.00E+00	1-CL-1B	CLASS 1B
		1.00E+00	1-SEQ-LP2-018	SEQUENCE LP2-018
		1.00E+00	2-ES-H/E	CLASS H/E
		1.00E+00	2-SEQ-1BE-081	SEQUENCE 1BE-081
		1.50E-01	3OPPH-LERF-NP-F--	LERF Not Precluded Due to SORVs / Timing
		9.00E-01	3RXRX-FRECINJH--	OPERATOR FAILS TO RECOVER INJECTION BEFORE RPV MELT
		9.86E-01	OSP-C-%G8	SEISMIC FRAGILITY FOR %G8: Offsite Power

TABLE 5.5-6

PBAPS UNIT 3 TOP 10 SEISMIC LERF CUTSETS

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
		6.58E-01	S-CEPA1--C-%G8	S-FRAG %G8: Panel 20C003, 20C004C, 30C003, 30C004C, 00C29(A-D)
		1.00E+00	SRX08_EHUATT-TDXI0	S-HEP G8: OPS FAIL TO PERFORM SE-11 LOAD SHED FOR FLEX (single unit, both divisi
		9.54E-01	ZZAVFACTOR	PLANT AVAILABILITY FACTOR
7	4.39E-07	5.25E-06	%G8	Seismic Initiating Event (>0.9g)
		1.00E+00	1-CL-1B	CLASS 1B
		1.00E+00	1-SEQ-LP2-018	SEQUENCE LP2-018
		1.00E+00	2-ES-H/E	CLASS H/E
		1.00E+00	2-SEQ-1BE-081	SEQUENCE 1BE-081
		1.50E-01	3OPPH-LERF-NP-F--	LERF Not Precluded Due to SORVs / Timing
		9.00E-01	3RXRX-FRECINJH--	OPERATOR FAILS TO RECOVER INJECTION BEFORE RPV MELT
		9.86E-01	OSP-C-%G8	SEISMIC FRAGILITY FOR %G8: Offsite Power
		6.58E-01	S-CEPA1--C-%G8	S-FRAG %G8: Panel 20C003, 20C004C, 30C003, 30C004C, 00C29(A-D)
		1.00E+00	SRX08_EHULS-ACDXI3	S-HEP G8: INITIAL LOAD SHED PER SE-11, ATT. T (STEPS FOR RCIC)
		9.54E-01	ZZAVFACTOR	PLANT AVAILABILITY FACTOR

TABLE 5.5-6

PBAPS UNIT 3 TOP 10 SEISMIC LERF CUTSETS

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
8	4.39E-07	5.25E-06	%G8	Seismic Initiating Event (>0.9g)
		1.00E+00	1-CL-1B	CLASS 1B
		1.00E+00	1-SEQ-LP2-018	SEQUENCE LP2-018
		1.00E+00	2-ES-H/E	CLASS H/E
		1.00E+00	2-SEQ-1BE-081	SEQUENCE 1BE-081
		1.50E-01	3OPPH-LERF-NP-F--	LERF Not Precluded Due to SORVs / Timing
		9.00E-01	3RXRX-FRECINJH--	OPERATOR FAILS TO RECOVER INJECTION BEFORE RPV MELT
		9.86E-01	OSP-C-%G8	SEISMIC FRAGILITY FOR %G8: Offsite Power
		6.58E-01	S-CEPA1--C-%G8	S-FRAG %G8: Panel 20C003, 20C004C, 30C003, 30C004C, 00C29(A-D)
		1.00E+00	SRX08_RHUBLKSTDXI3	S-HEP G8: OPERATOR FAILS TO MANUALLY START RCIC (BLACK START) - SEISMIC PRA VERS
		9.54E-01	ZZAVFACTOR	PLANT AVAILABILITY FACTOR
9	4.33E-07	5.25E-06	%G8	Seismic Initiating Event (>0.9g)
		1.00E+00	1-CL-1B	CLASS 1B
		1.00E+00	1-SEQ-LP1-028	SEQUENCE LP1-028

TABLE 5.5-6

PBAPS UNIT 3 TOP 10 SEISMIC LERF CUTSETS

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
		1.00E+00	2-ES-H/E	CLASS H/E
		1.00E+00	2-SEQ-1BE-081	SEQUENCE 1BE-081
		1.50E-01	3OPPH-LERF-NP-F--	LERF Not Precluded Due to SORVs / Timing
		9.00E-01	3RXRX-FRECINJH--	OPERATOR FAILS TO RECOVER INJECTION BEFORE RPV MELT
		9.86E-01	OSP-C-%G8	SEISMIC FRAGILITY FOR %G8: Offsite Power
		6.49E-01	S-CC190A--C-%G8	S-FRAG %G8: Relay Chatter Group 190A (52B-151N relays) (EDGs A and D - Recov.)
		1.00E+00	SRX08_AHUBTL-RDXD3	S-HEP G8: OPS FAILS TO VALVE IN N2 BOTTLES AFTER ACCUM DEPLETION (LATE; CONDITIO
		1.00E+00	SRX08_AHUBTL-RDXI3	S-HEP G8: OPERATORS FAIL TO VALVE IN N2 BOTTLES AFTER ACCUMULATOR DEPLETION (EAR
		1.00E+00	SRX08_AHU--CADDXD3	S-HEP G8: OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 2 INS 'B' - DELAYED; CONDITIO
		1.00E+00	SRX08_AHU--CADDXI3	S-HEP G8: OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 2 INS 'B'
		1.00E+00	SRX08_EHUCWGCNDXI0	S-HEP G8: CONOWINGO OPERATOR FAILS TO ENERGIZE SBO LINE TO PEACH BOTTOM (EXECUTI
		1.00E+00	SRX08_EHULS-ACDXI3	S-HEP G8: INITIAL LOAD SHED PER SE-11, ATT. T (STEPS FOR RCIC)
		1.00E+00	SRX08_EHU-SE11DXI0	S-HEP G8: OPERATOR CROSS TIES 4KV EMERGENCY BUSES
		1.00E+00	SRX08_RHUBLKSTDXI3	S-HEP G8: OPERATOR FAILS TO MANUALLY START RCIC (BLACK START) - SEISMIC PRA VERS

TABLE 5.5-6

PBAPS UNIT 3 TOP 10 SEISMIC LERF CUTSETS

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
		9.54E-01	ZZAVFACTOR	PLANT AVAILABILITY FACTOR
		1.00E+00	SRX08_EHURLYDGDXI3	S-HEP G8: OPERATOR FAILS TO MITIGATE RELAY CHATTER for EDGs (SEISMIC)
10	4.33E-07	5.25E-06	%G8	Seismic Initiating Event (>0.9g)
		1.00E+00	1-CL-1B	CLASS 1B
		1.00E+00	1-SEQ-LP1-028	SEQUENCE LP1-028
		1.00E+00	2-ES-H/E	CLASS H/E
		1.00E+00	2-SEQ-1BE-081	SEQUENCE 1BE-081
		1.50E-01	3OPPH-LERF-NP-F--	LERF Not Precluded Due to SORVs / Timing
		9.00E-01	3RXRX-FRECINJH--	OPERATOR FAILS TO RECOVER INJECTION BEFORE RPV MELT
		9.86E-01	OSP-C-%G8	SEISMIC FRAGILITY FOR %G8: Offsite Power
		6.49E-01	S-CC190A--C-%G8	S-FRAG %G8: Relay Chatter Group 190A (52B-151N relays) (EDGs A and D - Recov.)
		1.00E+00	SRX08_AHUBTL-RDXD3	S-HEP G8: OPS FAILS TO VALVE IN N2 BOTTLES AFTER ACCUM DEPLETION (LATE; CONDITIO
		1.00E+00	SRX08_AHUBTL-RDXI3	S-HEP G8: OPERATORS FAIL TO VALVE IN N2 BOTTLES AFTER ACCUMULATOR DEPLETION (EAR
		1.00E+00	SRX08_AHU--CADDXD3	S-HEP G8: OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 2 INS 'B' - DELAYED; CONDITIO

TABLE 5.5-6

PBAPS UNIT 3 TOP 10 SEISMIC LERF CUTSETS

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
		1.00E+00	SRX08_AHU--CADDXI3	S-HEP G8: OPERATOR FAILS TO ALIGN CAD TANK TO UNIT 2 INS 'B'
		1.00E+00	SRX08_EHUCWGPBDXI0	S-HEP G8: PEACH BOTTOM OPERATOR FAILS TO ALIGN CONOWINGO SBO LINE TO EMERGENCY 4
		1.00E+00	SRX08_EHULS-ACDXI3	S-HEP G8: INITIAL LOAD SHED PER SE-11, ATT. T (STEPS FOR RCIC)
		1.00E+00	SRX08_EHU-SE11DXI0	S-HEP G8: OPERATOR CROSS TIES 4KV EMERGENCY BUSES
		1.00E+00	SRX08_RHUBLKSTDXI3	S-HEP G8: OPERATOR FAILS TO MANUALLY START RCIC (BLACK START) - SEISMIC PRA VERS
		9.54E-01	ZZAVFACTOR	PLANT AVAILABILITY FACTOR
		1.00E+00	SRX08_EHURLYDGDXI3	S-HEP G8: OPERATOR FAILS TO MITIGATE RELAY CHATTER for EDGs (SEISMIC)

5.6 SPRA Quantification Uncertainty Analysis

A parametric uncertainty assessment of the PBAPS SCDF and SLERF is performed using the EPRI UNCERT (Ver. 4.0) software in combination with the EPRI ACUBE (Ver. 2.0) software. UNCERT is a Windows based program that uses CAFTA generated cutsets and PRA databases as inputs to quantify the parametric uncertainty distribution of a group of cutsets.

Probability distribution types and associated distribution statistics are assigned to each of the basic events. These distributions are entered into the CAFTA database for the SPRA. In addition, Type Code information (stored in the TC Table within the CAFTA "rr" database) is used to account for the state of knowledge dependence among correlated input distributions. UNCERT randomly samples from each of the input distributions, in conjunction with the Type Code database, calls the ACUBE algorithm at each sample to compute the best estimate result of the CAFTA cutset file. The results are stored and the input distributions are sampled many additional times. When all the trials are completed, the stored results are processed to form a probability distribution of the sampled SCDF and SLERF result.

A Monte Carlo (or Latin Hypercube sampling algorithm) evaluation can be performed using correlated or uncorrelated probability distributions to represent the inputs for the basic events. The probability density distribution describing the uncertainty in a component failure probability is characterized as a state of knowledge about an assumed fixed value, the same state of knowledge (i.e., the same distribution) may in fact underlie many distinct basic events. For example, the knowledge of the failure rate of one particular motor operated valve is typically based on experience with MOVs in various plant systems. Therefore, the various basic events that involve the failure of a motor operated valve are all in fact estimated from a single "state of knowledge" distribution. Therefore, basic events based on common data are mapped to a single data variable to ensure proper state of knowledge correlation in the parametric sampling process. This is performed by assigning an appropriate Type Code to each unique basic event.

Distribution information is assigned to all basic events in the cutset files, except for those that are intended to be modeled as constants. The sampling covers both non-seismic variables in the cutsets as well as seismic variables. The distribution sampling of the seismic hazard intervals and fragilities are summarized below.

The seismic hazard interval initiating events are sampled using the sampling equations provided by the FRANX software; for example, the equation for the %G1 seismic interval is as follows:

IF(@POINTCALC=1,4.65E-04,INVLOGN(3.60E-04,3.2406,W))

This equation uses a logical IF statement as a switch to determine whether the point estimate mean frequency should be returned or whether a sampling capability equation should be returned for use in parametric uncertainty sampling.

If the @POINTCALC variable (added to the SPRA Type Code database) is set by the analyst to a value of one (1) (the base value used in SPRA quantification runs) the initiator frequency equation returns the point estimate mean value (4.65E-04 in the example above). If the @POINTCALC Type Code is set by the analyst to 0 (or any value other than 1) the initiator equation invokes the CAFTA INVLOGN (inverse lognormal) function that will be used in parametric uncertainty sampling of cutsets using the UNCERT software.

The INVLOGN function takes three arguments, the median frequency, the frequency error factor, and the sampled lognormal percentile. The third argument of the INVLOGN function, the W Type Code variable, is used to ensure the state of knowledge correlation sampling of the seismic hazard intervals. This variable makes sure that during sampling that each of the hazard interval initiators is being sampled from the same hazard percentile. During UNCERT parametric uncertainty sampling the W Type Code variable (a Uniform distribution variable) is randomly sampled between 0.01 and 0.99. A W Type Code sample instructs the INVLOGN function to return the associated lognormal percentile; for example, a W Type Code sample value of 0.23 during UNCERT analysis will instruct the INVLOGN function to calculate the 23% percentile of the seismic interval initiators.

The seismic fragility basic events are sampled using the fragility sampling equation provided by the FRANX software:

$$f = \Phi\left[\frac{\lambda n\left(\frac{a}{A_m}\right)}{\beta_R}\right]$$

Where:

Φ is the standard Gaussian cumulative distribution.

a is the peak ground acceleration level.

A_m is the median seismic capacity of the component.

β_u is assigned as a Lognormal error factor distribution statistic to the A_m (the EF version of β_u is auto-calculated by the EPRI FRANX software using the relationship EF=exp(1.645* β_u)).

β_R is the parameter that accounts for random variability in the ground acceleration capacity.

Like the hazard interval initiators, the fragility basic events use CAFTA equations to implement the fragility model concept. For example, the format of a SLC pump fragility basic event equation for the %G6 hazard interval is as follows:

> IF(@POINTCALC=1, cummlogn_med('S-SLPM1--AM', EXP(1.645* 'S-SLPM1--BC'), '@%G6'),cummlogn_med('S-SLPM1--AM', EXP(1.645*'S-SLPM1--BR'),'@%G6'))

This equation uses a logical IF statement as a switch to determine whether the point estimate mean fragility should be returned or whether a sampling capability equation should be returned. If the @POINTCALC Type Code is set by the analyst to a value of one (1) (the default value) the fragility equation returns the point estimate mean fragility using the typical β_C version of the fragility mathematical model. If the @POINTCALC Type Code is set to 0 (or any value other than 1) the fragility equation invokes the CAFTA CUMMLOGN_MED function that will be used in parametric uncertainty sampling and will employ the β_R version of the fragility mathematical model with the β_U defining the distribution of the A_m during the sampling.

SCDF Uncertainty

Parametric sampling of the PBAPS SPRA SCDF was performed on the base SCDF cutset file using the UNCERT Latin Hypercube sampling option, ACUBE BDD value of /c=3000 cutsets (which produces 100% BDD at each pass), and 20,000 samples. The resulting spread of the SCDF is often characterized by the Range Factor of the resulting sampling distribution (calculated as the SQRT(95%/5%) or 95%/5%). For the PBAPS SPRA SCDF, the range factor is approximately 6.5:

- SCDF 95%: 9.03E-05/yr
- SCDF 50%: 1.42E-05/yr
- SCDF 5%: 2.14E-06/yr

This uncertainty range factor on SCDF is reasonable and generally reflective of the uncertainty of the hazard curve (the dominant hazard intervals are %G5 thru %G8 and each of these has an error factor in the 6 to 7 range).

SLERF Uncertainty

Parametric sampling of the PBAPS SPRA SLERF was performed on the base SLERF cutset file using the UNCERT Latin Hypercube sampling option, ACUBE BDD value of /c=7000 cutsets (which produces ~99% BDD at each pass), and 15,000 samples. The resulting spread of the SLERF is often characterized by the Range Factor of the resulting sampling distribution (calculated as the SQRT(95%/5%) or 95%/5%). For the PBAPS SPRA SLERF, the range factor is approximately 7.0:

- SLERF 95%: 2.54E-05/yr

- SLERF 50%: 3.70E-06/yr
- SLERF 5%: 5.24E-07/yr

This uncertainty range factor on SLERF is reasonable and generally reflective of the uncertainty of the hazard curve (the dominant hazard intervals are %G5 thru %G8 and each of these has an error factor in the 6 to 7 range). The uncertainty spread in SLERF is similar to that of SCDF because many of the dominant accident scenarios comprising SCDF proceed directly to SLERF or with very few additional failures.

Completeness Uncertainty

The SPRA should be of sufficient scope and level of detail to support the risk-informed decision under consideration.

Overall Scope: The overall scope of the SPRA is reasonably defined in terms of the following:

- Metrics used to evaluate risk
- Plant Operating States (POSs) for which the risk is to be evaluated
- Types of hazard groups and initiating events that can potentially challenge and disrupt the normal operation of the plant and, if not prevented or mitigated, would eventually result in core damage, a release, and/or health effects.

The following discussions are implemented for the PBAPS SPRA.

- The risk metrics used are SCDF and SLERF. This is typical of SPRAs and consistent with industry PRA standards and the SPID.
- The Plant Operating State (POS) is limited to at-power; this is consistent with the SPID requirements. The PBAPS SPRA does not model postulated seismic-induced accidents during shutdown or during power transition states.
- The SPRA addresses the entire (i.e., well beyond design basis) seismic hazard curve (PGA-based). Separate SPRA models are not explicitly built to model different spectral hazard curves. This is a typical SPRA modeling approach (i.e., PGA hazard curve used).
- The SPRA covers the typical spectrum of seismic-induced initiating event states (e.g., seismic-induced LOOP, seismic-induced LOOP-LOCA, seismic-induced LOOP-ATWS, seismic-induced key building failures, etc.) as well as seismic-induced secondary hazards.

Level of Detail: A number of decisions made by the analyst determine the level of details included in an SPRA. These decisions include, for example, the structure of the event trees, the mitigating systems that should be included as providing potential success for critical safety functions, the structure of the fault trees, and the screening criteria used to determine which failure modes for which SSCs are to be included.

The level of details needed is that detail required to capture the effect of an application (i.e., the SPRA model needs to be of sufficient detail to ensure the impact of the application can be assessed).

The level of details in the system fault tree models, accident sequence models, human reliability analysis, and data of the SPRA models is effectively the same as the detailed at-power PRA models used as input to development of the SPRA. The one set of accident sequences not explicitly included in the PBAPS SPRA are "seismic transient" sequences because they are assessed as non-significant contributors or already addressed by at-power internal events accident sequence models. Given the low capacity of offsite power, the plant likely would remain at power (not trip) if offsite power was not failed by the seismic event or if a trip did occur the likelihood of seismic-induced failure of significant mitigation equipment is very low. This modeling approach is typical of SPRAs.

The completeness of the PBAPS SPRA is sufficient for most risk applications, typical of full-scope SPRAs and consistent with the SPID.

5.7 SPRA Quantification Sensitivity Analysis

Candidate sensitivity cases for the PBAPS SPRA model were identified consistent with the methodology provided in NUREG-1855 [28] and performed for the PBAPS FPIE PRA model. The selection process for the sensitivity cases is documented in Appendix I of the PBAPS SPRA Quantification Notebook [45]. 14 sensitivity cases have been identified in the following five (5) PRA element categories.

PRA Element	Description
IE	Probabilistic Seismic Hazard Analysis (PSHA) (Cases 1a, 1b, 1c, 1d)
AS	Seismic LERF evaluation (Cases 2a, 2b)
SC	Core cooling success following containment failure or venting through non-hard pipe vent paths (Very Small LOCA) (Case 3a)
SY	Operability of equipment for seismic induced accident sequences (SSC Fragilities) (Cases 4a, 4b, 4c, 4d, 4e)
HR	HRA Evaluation under seismic event (Cases 5a, 5b)

Table 5.7-1 provides a summary of the sensitivity cases performed. The seismic PRA model has been used to provide insights and feedback on the degree of seismic safety enhancement (seismic risk reduction) that can be achieved by potential SPRA model enhancements.

In addition, Tables 5.7-2 and 5.7-3 provide the SCDF and SLERF truncation sensitivity cases, respectively, to support the selection of truncation limits for the base SPRA model quantification. Quantification truncation sensitivities to establish adequate model results convergence were performed and evaluated as part of the peer review.

Sensitivity Case 1a: Assume the 84% Upper Bound of Seismic Hazard Curve

This sensitivity was performed by replacing the seismic initiating events with values consistent with the 84% Upper Bound of the PBAPS Seismic Hazard Curve. The SCDF and SLERF increases by 104% and 115%, respectively. This sensitivity demonstrates that changes to the initiator frequency (Hazard) can have a significant impact on results.

Sensitivity Case 1b: Assume the 16% Lower Bound of Seismic Hazard Curve

This sensitivity was performed by replacing the seismic initiating events with values consistent with the 16% Lower Bound of the PBAPS Seismic Hazard Curve. The SCDF and SLERF decreases by 82% and 81%, respectively. This sensitivity demonstrates that changes to the initiator frequency (Hazard) can have a significant impact on results.

Sensitivity Case 1c: Assume the EPRI 1989 Seismic Hazard Curve

This sensitivity was performed by replacing the seismic initiating events with values consistent with the PBAPS EPRI 1989 Seismic Hazard Curve. These values were used in the PBAPS IPEEE document. It is understood that the plant specific fragility calculations developed for the PBAPS 2017 SPRA model and used for this sensitivity case are not based on the same seismic hazard input used to develop the EPRI 1989 Seismic Hazard Curve. Therefore, there is a potential disconnect between the seismic hazard frequencies and the seismic fragilities in this sensitivity case. However, for the purposes of evaluating the potential impact of using different mean hazard frequencies from a hazard curve, this potential disconnect between the hazard curve and the fragilities is not explicitly evaluated. The SCDF and SLERF decreases by 56% and 69%, respectively. This sensitivity demonstrates that changes to the initiator frequency (Hazard) can have a significant impact on results.

Sensitivity Case 1d: Subdivide G8 Seismic Hazard Interval to increase CLERP

Seismic hazard interval G8 is the widest seismic interval which captures all seismic events of magnitude greater than 0.9g. SCDF already reaches a CCDP of effectively 1.0 at magnitudes of around 0.9g, however SLERF only reaches a CLERP of about 0.32 for the %G8 seismic hazard interval. This sensitivity study subdivides the %G8 hazard interval from 0.9g to 2.3g using six hazard intervals to represent slices of the hazard curve instead of a single hazard interval for > 0.9g. The aim of this sensitivity case is to measure the increase in CLERP when the higher magnitude portions of the hazard curve are accounted for in greater detail (e.g., %G8 hazard interval CLERP ~ 1.0). The six extended %G8 hazard intervals along with their frequencies and representative ground motions are shown below.

%G8 Hazard Interval (g)	Representative Ground Motion (g)	Frequency (/yr)	LERF (/yr)	CLERP
0.9->1.1	0.995	1.96E-06	5.61E-07	2.86E-01
1.1->1.3	1.1958	1.12E-06	4.52E-07	4.04E-01
1.3->1.6	1.44222	9.26E-07	5.10E-07	5.51E-01
1.6->1.9	1.7436	4.76E-07	2.76E-07	5.79E-01
1.9->2.3	2.0905	3.31E-07	1.45E-07	4.37E-01

>2.3	2.53	4.41E-07	4.41E-07(1)	1.00E+00

Note 1: Due to computer memory limitations, this case could only be quantified at a truncation level below 2.50E-07/yr where the LERF is calculated at 2.71E-07/yr (CLERP = 0.62). Due to the high representative ground motion for this hazard interval, it is assumed that quantification at lower truncation would yield more cutsets summing to a LERF of 4.41E-07/yr (i.e., CLERP = 1.0).

The base contribution of hazard intervals %G1-%G7 to SLERF is 2.45E-06/yr. Adding this to the LERFs calculated above for the extended %G8 hazard interval, the total SLERF for this sensitivity is 4.83E-06/yr, approximately a 17% increase from the base SLERF.

Sensitivity Case 2a: Credit 0.1 Conditional Probability for SLERF Reduction

This sensitivity case is based on a separate, more detailed investigation to determine if there are potential conservatisms in the treatment of assigning LERF end states for both the FPIE PRA, Fire PRA, and SPRA models. This sensitivity case supports potential options to reduce the relatively high calculated SLERF value 4.1E-06/yr for the PBAPS baseline SPRA model. The discussion below is based on a review of NUREG/CR-7110, U.S. Nuclear Regulatory Commission, "*State-of-the-Art Reactor Consequence Analyses Project, Volume 1: Peach Bottom Integrated Analysis"*, Rev. 1, May 2013 [53] and supplemental MAAP runs performed for a separate evaluation [62]. The conclusions from this separate evaluation [62] are as follows:

"An investigation into the assumptions related to the likelihood of unmitigated short term SBO scenarios with no RPV makeup at time=0 leading to LERF resulted in the following insights.

- *The likelihood of experiencing a SORV during the core melt progression process is assessed as being quite high (i.e., 95% likelihood). The presence of a SORV has a dramatic influence on the potential source terms as much of the fission products are swept to the suppression pool prior to vessel failure and subsequent liner melt-through.*
- *MELCOR and recent MAAP5 runs indicate that the time to vessel failure may be longer than previously anticipated, and the time that the fission product releases exceed the threshold value for being characterized as large in SORV scenarios could be extended for a significant amount of time, and may not occur at all (at least within the first 48 hours).*
- *The recent evacuation time estimates for PBAPS indicate that the time to evacuate 100% of the population out to the EPZ is shorter*

than in previous analysis (i.e., 6.5 hours at most, compared to more than 8 hours previously).

- *If [Steam Line Rupture] SLR occurs, then the likelihood of a large and early release increases dramatically since the fission products are released directly to the drywell in this scenario and do not get the benefit of being transported to the suppression pool from the SORV. High Pressure scenarios are also assessed as being more likely to lead a large and early release.*
- *The conditions required for a SLR in [Short Term Station Blackout] STSBO scenarios was examined in the SOARCA study. Conditions for SLR would only occur if the SORV seized partially open such that enough depressurization would occur to preclude other SRV openings, but at the same time keep the RPV pressure high enough to enable a SLR. This is assessed as fairly unlikely, and a 10% likelihood value is assigned.*

A Monte Carlo analysis was used to estimate the overall likelihood of LERF combining the inputs above. The results show that the likelihood is about 11.6%. This likelihood is dominated by the assumptions related to a SLR occurring."

Given the discussion above, however, a bounding value can also be derived. It is assessed that the likelihood of an SORV leading to conditions that would not be LERF is very high and that the condition that would be LERF is very low. A 5% bounding value for SORV scenarios leading to LERF can be applied. For the SLR and High-Pressure scenarios, LERF cannot be precluded so these scenarios can be conservatively assumed to be LERF. The results show that the bounding analysis is about 19.3%, or approximately 20%.

The current base SPRA model uses a potentially conservative conditional value of 15% for the conditional probability for SLERF reduction. Sensitivity Case 2a reduces the conditional value to a likely best estimate value of 10%. The SLERF decreases by 17% for this sensitivity case.

<u>Sensitivity Case 2b: Assume Seismic Events >0.5g Result in SLERF</u>

This sensitivity was performed by estimating the impact on SLERF when assuming that all seismic events with magnitude >0.5g result in sufficient delay in the evacuation time such that they are modeled as leading directly to the SLERF end state. This sensitivity case is performed by assuming that all SCDF contributors >0.5g (i.e., %G5, %G6, %G7, %G8) are assumed to be equal to SLERF. The SLERF increases significantly by nearly a factor of three (3) for this sensitivity case.

Sensitivity Case 3a: Very Small LOCA Impact on CRD Success Criteria

The existence of a Very Small LOCA is assumed to preclude credit for CRD for RPV makeup due to inadequate flow capacity from CRD to overcome both the decay heat boiloff and the approximate 50 gpm flow from the Very Small LOCA. Given the potential uncertainty in the Very Small LOCA flow rate, this sensitivity evaluates the risk impact if CRD could be credited even if a Very Small LOCA existed. The estimated risk reduction is based on the baseline SCDF FV value of 7.3E-04 and SLERF FV value of 2.2E-04 for the Very Small LOCA. The sensitivity results show a negligible change in SCDF and SLERF.

Sensitivity Case 4a: Fragility sensitivity study for Reference Earthquake

It was determined that for some high-risk contributors, an earthquake level higher than the GMRS would be appropriate. For these components, fragilities were improved based on a more appropriate reference earthquake. These improved fragilities were input to the PRA model and a sensitivity study was performed to determine the impact. At a higher reference earthquake level, the structures experience significant cracking which decreases the frequency and increases damping. This results in a decrease in overall building response. This in-turn increases the fragilities for some components and decreases the overall plant seismic risk. The results of the quantitative sensitivity study supported a very minor decrease in SCDF and SLERF (i.e., <1%). In addition, there were minimal changes to the SCDF and SLERF risk importance measures. The sensitivity study was performed to ensure that considering the higher reference earthquake did not result in identifying additional plant vulnerabilities and risks and did not identify any new risk insights.

Sensitivity Case 4b: Improve Fragility for 125 VDC Battery Racks

Correlated failure of the 125 VDC Battery Racks is modeled with an A_m value of 0.73g. Correlated failure of the 125 VDC Battery Racks is a significant contributor to the PBAPS SPRA results because it results in an early Station Blackout scenario with no RPV makeup from HPCI and limited credit for operator action for Black Start RCIC. Case 4b evaluates the risk impact if the A_m for 125 VDC Battery Racks is increased from 0.73g (anchorage failure) to 1.74g (estimate of functional failure). The SCDF and SLERF decreases by 11% and 10%, respectively.

Sensitivity Case 4c: Eliminate fragility modeling uncertainty (β_u) from SSC fragility calculations

This sensitivity case assesses the effect of assuming perfect knowledge of the SSC fragility characterization. Fragility modeling uncertainty is a critical impact on the calculated risk metric. There currently is not an approach available to reduce this uncertainty to zero. For this sensitivity case, the SCDF and SLERF significantly decreases by 60% and 67%, respectively.

Sensitivity Case 4d: Improve A_m for normal offsite AC power

The fragility for normal offsite AC power is based on an industry generic value of A_m=0.3g for the PBAPS SPRA model. Given the high-risk contribution from seismic induced loss of offsite power events, any enhancements to the offsite AC power fragility would likely be very beneficial. However, the ceramic insulators are often a limiting failure mode for offsite AC power. In addition, significant work has been performed at some locations in an attempt to demonstrate significant improvement in the generic value used, without success. For this sensitivity case, the A_m for offsite AC power is assumed to be increased to 0.5g. The SCDF and SLERF decreases by 29% and 20%, respectively.

Sensitivity Case 4e: Identify risk impact for correlation of RHR pumps

The fragility analyses [25] identified that Pumps B and C are not oriented the same as each other or the same as pumps A and D. Pumps A and D are not correlated because the response spectra peaks are slightly different, and they are far enough apart that the motions will not occur at the exact same time for the two pumps. Therefore, these pumps are not correlated This sensitivity case evaluates the impact of assuming correlated seismic failure of all four (4) RHR pumps within the same unit. With the relatively high Am value for the RHR pumps, this sensitivity case resulted in no change in SCDF or SLERF.

Sensitivity Case 5a: Do not credit operator recovery from relay chatter events

This sensitivity case eliminates all credit for operator recovery from relay chatter events. The SCDF and SLERF increases by 8% and 2%, respectively. The risk increase is relatively small because the Base Case SPRA model provides relatively minimal credit for operator recovery from relay chatter events due to the potentially limited amount of time available for the operators to perform the necessary actions.

Sensitivity Case 5b: Improve credit for operator recovery from relay chatter events

This sensitivity case enhances credit for operator recovery from relay chatter events by reducing the base case relay chatter HEPs by a factor of two (2). The SCDF and SLERF increases by 1% and 2%, respectively. The risk decrease is relatively small because the Base Case SPRA model still has high SCDF and SLERF contributors from other failure modes.

TABLE 5.7-1
SUMMARY OF PBAPS SPRA SENSITIVITY CASES

Sensitivity Case #(1)	Description	CDF (/yr)	Delta CDF (/yr)	% Delta CDF	LERF (/yr)	Delta LERF (/yr)	% Delta LERF
Base Case	Base Case (Unit 3)	2.14E-05	N/A	N/A	4.14E-06	N/A	N/A
Case 1a	Use 84% upper bound limit from PB Seismic Hazard curve.	4.37E-05	2.23E-05	104.2%	8.88E-06	4.74E-06	114.5%
Case 1b	Use 16% lower bound limit from PB Seismic Hazard curve.	3.93E-06	-1.75E-05	-81.6%	7.81E-07	-3.36E-06	-81.1%
Case 1c	Use 1989 EPRI Hazard Curve	9.47E-06	-1.19E-05	-55.7%	1.30E-06	-2.84E-06	-68.6%
Case 1d	Subdivide %G8 Seismic Hazard Interval	2.14E-05	N/A	N/A	4.83E-06	6.90E-07	16.7%
Case 2a	Apply conditional SLERF probability of 0.1 instead of 0.15 to address potential more realistic evaluation of Level 2 phenomena.	2.14E-05	N/A	N/A	3.45E-06	-6.90E-07	-16.7%
Case 2b	Assume that seismic events >0.5g have a significant impact on evacuation and any core damage events result in LERF.	2.14E-05	N/A	N/A	1.65E-05	1.24E-05	298.6%

TABLE 5.7-1
SUMMARY OF PBAPS SPRA SENSITIVITY CASES

Sensitivity Case #(1)	Description	CDF (/yr)	Delta CDF (/yr)	% Delta CDF	LERF (/yr)	Delta LERF (/yr)	% Delta LERF
Case 3a	Revise impact of Very Small LOCA on CRD success criteria for RPV inventory makeup	2.14E-05	-2.00E-08	-0.1%	4.14E-06	-9.00E-10	0.0%
Case 4a	Fragility sensitivity study for Reference Earthquake sensitivity study	2.13E-05	-1.00E-07	-0.5%	4.10E-06	-4.00E-08	-1.0%
Case 4b	Improve fragility for 125 VDC Battery Racks	1.90E-05	-2.40E-06	-11.2%	3.74E-06	-4.00E-07	-9.7%
Case 4c	Eliminate fragility modeling uncertainty (Bu) from SSC fragility calculations	8.50E-06	-1.29E-05	-60.3%	1.36E-06	-2.79E-06	-67.3%
Case 4d	Improve Am for normal offsite AC power	1.53E-05	-6.10E-06	-28.5%	3.30E-06	-8.40E-07	-20.3%
Case 4e	Identify risk impact for correlation of RHR pumps	2.14E-05	0.00E+00	0.0%	4.14E-06	0.00E+00	0.0%
Case 5a	Do not credit operator recovery from relay chatter events.	2.31E-05	1.70E-06	7.9%	4.21E-06	6.70E-08	1.6%

TABLE 5.7-1

SUMMARY OF PBAPS SPRA SENSITIVITY CASES

Sensitivity Case #[(1)]	Description	CDF (/yr)	Delta CDF (/yr)	% Delta CDF	LERF (/yr)	Delta LERF (/yr)	% Delta LERF
Case 5b	Improve credit for operator recovery from relay chatter events.	2.12E-05	-2.00E-07	-0.9%	4.06E-06	-8.00E-08	-1.9%

Notes to Table 5.7-1:

(1) The sensitivity study SCDF and SLERF quantifications use the same truncation levels (per hazard interval) as the Base Case. This is reasonable for the purposes of sensitivity studies and is typical practice given that truncation levels are typically set at a level that already challenges computer memory and computational speed. The truncation level convergence test (i.e., < +~5% per decade decrease in truncation level) used in the Base Case quantifications if specifically re-confirmed for each of the sensitivity cases is expected to produce the same truncation levels for most of the sensitivity studies.

TABLE 5.7-2

SCDF TRUNCATION SENSITIVITY CASES

Seismic Hazard Interval	Hazard Interval Frequency	SCDF	Truncation	SCDF	Truncation	% Change	Hazard Interval Truncation Selected
%G1	4.65E-04	3.58E-09	**1.00E-10**	3.62E-09	1.00E-11	1.2%	**1.00E-10**
%G2	4.21E-05	5.70E-07	**1.00E-10**	5.75E-07	1.00E-11	0.9%	**1.00E-10**
%G3	1.81E-05	1.78E-06	**1.00E-10**	1.81E-06	1.00E-11	1.5%	**1.00E-10**
%G4	9.54E-06	3.03E-06	**1.00E-10**	3.14E-06	1.00E-11	3.5%	**1.00E-10**
%G5	5.57E-06	4.44E-06	**1.00E-09**	4.45E-06	1.00E-10	0.1%	**1.00E-09**
%G6	4.72E-06	4.20E-06	**1.00E-08**	4.20E-06	1.00E-09	0.1%	**1.00E-08**
%G7	2.66E-06	2.47E-06	**1.00E-07**	2.47E-06	1.00E-08	0.1%	**1.00E-07**
%G8	5.25E-06	4.94E-06	**1.00E-06**	4.96E-06	1.00E-07	0.5%	**1.00E-06**
	Total SCDF	**2.14E-05**					

TABLE 5.7-3

SLERF TRUNCATION SENSITIVITY CASES

Seismic Hazard Interval	Hazard Interval Frequency	SLERF	Truncation	SLERF	Truncation	% Change	Hazard Interval Truncation Selected
%G1	4.65E-04	2.02E-10	**1.00E-12**	2.15E-10	1.00E-13	6.5%(1)	**1.00E-12**
%G2	4.21E-05	3.85E-08	**1.00E-11**	3.92E-08	1.00E-12	1.8%	**1.00E-11**
%G3	1.81E-05	1.34E-07	**1.00E-11**	1.36E-07	1.00E-12	1.6%	**1.00E-11**
%G4	9.54E-06	2.67E-07	**1.00E-11**	2.88E-07	1.00E-12	7.7%(2)	**1.00E-11**
%G5	5.57E-06	5.41E-07	**1.00E-09**	6.03E-07	1.00E-10	11.5%(3)	**1.00E-09**
%G6	4.72E-06	8.07E-07	**1.00E-09**	1.02E-06	1.00E-10	26.4%(4)	**1.00E-09**
%G7	2.66E-06	6.57E-07	**1.00E-09**	2.28E-06	1.00E-10	246.9%(5)	**1.00E-09**
%G8	5.25E-06	1.69E-06	**5.00E-08**	1.79E-06	5.00E-09	5.8%(6)	**5.00E-08**
	Total SLERF	**4.14E-06**					

Notes to Table 5.7-3:

(1) An increase of 6.5% is deemed acceptable due to the low risk contribution of the G1 seismic hazard interval (< 1%).

(2) The cutset file for this case included 7% Group 2 Contribution (i.e., high BDD). The 7.7% increase is conservative and is deemed acceptable.

(3) The cutset file for this case included 14% Group 2 Contribution. The 11.5% increase is conservative and is deemed acceptable.

(4) The cutset file for this case included 22% Group 2 Contribution. The 26.4% increase is conservative and is deemed acceptable.

(5) The cutset file for this case included 94.4% Group 2 Contribution (i.e., low BDD). The 246.9% increase is judged to be conservative and is deemed acceptable.

(6) Due to computational limits, the model cannot be quantified at a lower truncation level. The 5.8% increase is deemed acceptable.

5.8 SPRA Logic Model and Quantification Technical Adequacy

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

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

6.0 Conclusions

A seismic PRA has been performed for PBAPS in accordance with the guidance in the SPID [2]. The PBAPS Unit 2 Seismic PRA shows that the seismic CDF is 2.1E-5/yr and the seismic LERF is 4.0E-6/yr. The PBAPS Unit 3 Seismic PRA shows that the seismic CDF is 2.1E-5/yr and the seismic LERF is 4.1E-6/yr. Uncertainty, importance, and sensitivity analyses were performed. Sensitivity studies were performed to identify critical assumptions, evaluate the risk impact to variations in the critical assumptions, 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 the assumptions incorporated into the SPRA model.

The Seismic PRA as described in this submittal reflects the as-built/as-operated Seismic PRA freeze date of February 28, 2018 [61]. Appendix A provides a discussion of the peer review assessment performed for the SPRA. It also contains a list and subsequent disposition of peer review findings. There are no significant plant changes that are not included in the model which would have an adverse impact on the results. Reference section A.9 and Table A-5 for additional information. Further, no seismic hazard vulnerabilities were identified, and no plant actions have been taken or are planned given the insights (including final SCDF and SLERF values) from this study.

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8.0 Acronyms

ADS	Automatic Depressurization System
ANS	American Nuclear Society
ARI	Alternate Rod Insertion
ASCE	American Society of Civil Engineers
ASM	Application Specific Model
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram (also ATWT, Anticipated Transient Without Trip)
BDD	Binary Decision Diagram
BE	Best Estimate
BOC	Beginning of Cycle
CCDP	Conditional Core Damage Probability
CCF	Common Cause Failure
CDF	Core Damage Frequency
CET	Containment Event Tree
CEUS	Central and Eastern United States
CDFM	Conservative Deterministic Failure Margin
CLERP	Conditional Large Early Release Probability
COV	Coefficient of Variation
CRD	Control Rod Drive
CST	Condensate Storage Tank
DGB	Diesel Generator Building
DHR	Decay Heat Removal
DWS	Drywell Spray
ECT	Emergency Cooling Tower
ECW	Emergency Cooling Water
EDG	Emergency Diesel Generator
EPRI	Electric Power Research Institute
EOPs	Emergency Operating Procedures
EPZ	Emergency Planning Zone
ESEL	Expedited Seismic Equipment List
ESEP	Expedited Seismic Evaluation Program
ESW	Emergency Service Water
FEM	Finite Element Model
FIRS	Foundation Input Response Spectra
FLEX	diverse and FLEXible coping

F&O	Finding and Observation
FPIE	Full Power Internal Events
FV	Fussell-Vesely
GERS	Generic Ruggedness Response Spectra
GIP	Generic Implementation Procedure
GMC	Ground Motion Characterization
GMRS	Ground Motion Response Spectra
HCLPF	High-Confidence-of-Low-Probability of Failure
HCTL	Heat Capacity Temperature Limit
HEP	Human Error Probability
HF	High Frequency
HI	Human Interaction
HLR	High Level Requirement
HPCI	High Pressure Coolant Injection
HPSW	High Pressure Service Water
HRA	Human Reliability Analysis
HVAC	Heating, Ventilation and Air Conditioning
Hz	Hertz (unit)
IPEEE	Individual Plant Examination for External Events
IPSF	Integrated Performance Shaping Factor
ISLOCA	Inter-System LOCA
ISRS	In-Structure Response Spectrum
LAR	Limited Analytical Review
LB	Lower Bound
LERF	Large Early Release Frequency
LF	Low Frequency
LMSM	Lumped Mass Stick Model
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
MAFE	Mean Annual Frequency of Exceedance
MCR	Main Control Room
MCUB	Minimum Cut Upper Bound
MOV	Motor Operated Valve
N2	Nitrogen
NEI	Nuclear Energy Institute
NHS	Normal Heat Sink (i.e. Ultimate Heat Sink)

NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NTTF	Near Term Task Force
OPS	Operations
PBAPS	Peach Bottom Atomic Power Station
PGA	Peak Ground Acceleration
POS	Plant Operating State
PRA	Probabilistic Risk Assessment
PRT	Peer Review Team
PS	Pump Structure
PSHA	Probabilistic Seismic Hazard Analysis
RASP	Risk Assessment Standardization Project
RB	Reactor Building
RBCCW	Reactor Building Closed Cooling Water
RCIC	Reactor Core Isolation Cooling
RCICS	RCIC System
RG	Regulatory Guide
RHR	Residual Heat Removal
RLME	Repeated Large Magnitude Earthquake
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RW	Radwaste
RWST	Refueling Water Storage Tank
SBO	Station Blackout
SCDF	Seismic Core Damage Frequency
SEL	Seismic Equipment List
SFP	Spent Fuel Pool
SFR	Seismic Fragility Element Within ASME/ANS PRA Standard
SGIG	Safety Grade Instrument Gas
SHA	Seismic Hazard Analysis Element Within ASME/ANS PRA Standard
SHS	Seismic Hazard Submittal
SIET	Seismic Initiating Event Tree
SLC	Standby Liquid Control
SLERF	Seismic Large Early Release Frequency
SLR	Steam Line Rupture
SMA	Seismic Margin Assessment

SORV	Stuck-Open Relief Valve
SOV	Separation of Variables
SPC	Suppression Pool Cooling
SPID	Screening, Prioritization and Implementation Details
SPR	Seismic PRA Modeling Element Within ASME/ANS PRA Standard
SPRA	Seismic Probabilistic Risk Assessment
SQUG	Seismic Qualification Utility Group
SR	Supporting Requirements
SRT	Seismic Review Team
SRV	Safety Relief Valve
SSC	Structure, System or Component; Seismic Source Characterization
SSEL	Safe Shutdown Equipment List
SSHAC	Senior Seismic Hazard Analysis Committee
SSI	Soil Structure Interaction
STSBO	Short Term Station Blackout
SWEL	Seismic Walkdown Equipment List
TB	Turbine Building
TDT	Torus Dewatering Tank
UB	Upper Bound
UHS	Ultimate Heat Sink (i.e. Normal Heat Sink)
USI	Unresolved Safety Issue
V/H	Vertical/Horizontal acceleration ratio
Vs	shear wave velocity
WW-DW	Wet Well – Dry Well
ZPA	Zero Period Acceleration
βc	Composite logarithmic standard deviation
βr	Randomness logarithmic standard deviation
βu	Uncertainty logarithmic standard deviation

Appendix A

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

A.1. Overview of Peer Review

The PBAPS Seismic PRA was subjected to an independent peer review against the pertinent requirements in Part 5 of the ASME/ANS PRA Standard [4]. The peer review assessment [27], and subsequent disposition of peer review findings, is summarized here. The scope of the review encompassed the set of technical elements and supporting requirements (SR) for the SHA (seismic hazard), SFR (seismic fragilities), and SPR (seismic PRA modeling) elements for seismic CDF and LERF. The peer review therefore addressed the set of SRs identified in Tables 6-4 through 6-6 of the SPID [2].

The information presented here establishes that the SPRA has been peer reviewed by a team with adequate credentials to perform the assessment, establishes that the peer review process followed meets the intent of the peer review characteristics and attributes in Table 16 of RG1.200 R2 [15] and the requirements in Section 1-6 of the ASME/ANS PRA Standard [4], and presents the significant results of the peer review.

The PBAPS SPRA peer review was conducted during the week of March 20, 2017 at the Exelon offices in Kennett Square, PA. As part of the peer review, a walk-down of portions of PBAPS Units 2 & 3 was performed on Tuesday March 21, 2017 by 2 members of the peer review team who have the appropriate SQUG training and an additional member with expertise in the SPR technical elements and supporting requirements.

A.2. Summary of the Peer Review Process

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

The SPRA peer review process defined in [5] involves an examination by each reviewer of their assigned PRA technical elements against the requirements in the Standard [4] to ensure the robustness of the model relative to all of the requirements.

Implementing the review involves a combination of a broad scope examination of the PRA elements within the scope of the review and a deeper examination of portions of the PRA elements based on findings during the initial review. The SRs provide a structure which, in combination with the peer reviewers' PRA experience, provides the basis for examining the various PRA technical elements. If a reviewer identifies a question or discrepancy, that leads to additional investigation until the issue is resolved or a Fact and Observation (F&O) is written describing the issue and its potential impacts, and suggesting possible resolution.

For the review of the SHA, a team of two peer reviewers was assigned, one having lead responsibility. For the review of the SFR and SPR, a team of three peer reviewers was assigned to each with one having lead responsibility for that area. One of the SPR reviewers also served as the team lead, meaning that the total peer review team consisted of eight reviewers. In addition, there were a number of observers for each area as well as observers from the USNRC. In addition to those asked by the peer review team, the observers also submitted questions to and held discussions with the SPRA team.

For each SR reviewed, the responsible reviewers reached consensus regarding which of the Capability Categories defined in the Standard [4] that the PRA meets for that SR, and the assignment of the Capability Category for each SR was ultimately based on the consensus of the full review team. The Standard [4] also specifies high level requirements (HLR). Consistent with the guidance in the Standard, capability Categories were not assigned to the HLRs, but a qualitative assessment of the applicable HLRs in the context of the PRA technical element summary was made based on the associated SR Capability Categories.

As part of the review team's assessment of capability categories, F&Os were prepared. There are three types of F&Os defined in [5]: Findings, which identify issues that must be addressed in order for an SR (or multiple SRs) to meet Capability Category II; Suggestions, which identify issues that the reviewers have noted as potentially important but not requiring resolution to meet the SRs; and Best Practices, which reflect the reviewers' opinion that a particular aspect of the review exceeds normal industry practice. The focus in this Appendix is on Findings and their disposition relative to this submittal.

A.3. Peer Review Team Qualifications

The members of the peer review team were:

Team Lead

The Team Lead was Mr. Paul Amico of Jensen Hughes. Mr. Amico also served as one of the reviewers of the technical elements associated with SPR. Mr. Amico has 40 years of experience in the performance and management of domestic and international programs related to risk assessments and their application in nuclear power plants. He has been involved with seismic PRA for more than 35 years and is active in development of seismic PRA standards and in performance of seismic PRAs.

SHA

The SHA Lead was Dr. Glenn Rix of Geosyntec. Dr. Rix has over 30 years of experience in geotechnical earthquake engineering and engineering seismology, particularly in the central and eastern US (CEUS), and in seismic hazard assessment and risk mitigation. Dr. Rix was assisted in the hazard review by Dr. Annie Kammerer of Annie Kammerer Consulting. Dr. Kammerer has more than 17 years of experience in integrated seismic hazard and risk evaluations and performance-based risk-informed engineering. She is the lead of the seismic hazard working group for the ASME/ANS external event PRA and a member of the working group for the ANS SHA Standard ANSI/ANS 2.29-2008. She is also the author of the current NRC guidance for performing PSHA.

SFR

The SFR Lead was Mr. Gregory Hardy of Simpson, Gumpertz and Heger (SGH). Mr. Hardy has 35 years of experience in structural mechanics engineering with emphasis on probabilistic risk assessments, earthquake experience data based studies, finite element analysis, seismic margin studies and vibration testing for equipment qualification. Mr. Hardy was assisted by Mr. Eddie Guerra and Dr. Se-Kwon Jung. Mr. Guerra has over 7 years of experience in seismic engineering and seismic risk assessments including developing fragility calculations, performing building analysis and conducting seismic walkdowns. Dr. Jung has 15 years of experience in civil/structural engineering, specializing in finite element analysis of building structures and structural fragilities.

SPR

The SPR Lead was Mr. Lawrence Mangan of First Energy Nuclear Operating Company. Mr. Mangan has 8 years of experience in developing and maintaining PRA models for the Perry Nuclear Power Plant. He participated in two previous internal events PRA peer reviews. He also co-authored NUREGs related to reliability modeling of digital control systems for nuclear power plants. Mr. Mangan was assisted by Mr. Habib Shtiah as well as by Mr. Paul Amico. Mr Shtiah has 10 years of experience in development seismic PRAs as well as in related activities. He is currently the lead for the seismic PRA at Columbia Station.

In addition to the reviewers listed, the team was assisted by several working and non-working observers. Working observers included Mr. Jerry Doughty of Ameren, 30 years of experience including 2 years PRA experience, Dr. Ram Srinivasan, an independent consultant currently assisting TVA with the SPRA at Watts Bar, Sequoyah and Browns Ferry with 45 years of experience, including 9 years of fragility experience and Mr. Daniel Vazquez of Dominion with 17 years of experience including 9 years of fragility experience. Non-working observers included Mr. Eyad Ali and Mr. Mrinal Bose of Exelon and Mr. C. J. Fong, Mr. Todd Hilsmeier, Mr. Bob Pettis, Mr. Shilp Vasavada and Mr. Nathan Sanfilippo from the NRC. Some of the observers from the NRC only attended portions of the peer review.

The peer review team members met the peer reviewer independence criteria in NEI 12-13 [5]. None of the peer review team members had any involvement with the PBAPS elements under review as documented in the peer review report

A.4. Summary of the Peer Review Conclusions

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

SHA

- As required by the 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 (SSC) inputs to the PSHA are based on the Central and Eastern U. S. (CEUS) regional SSC model published in NUREG-2115 (i. e., the CEUS-SSC model). The ground motion characterization (GMC) inputs to the PSHA are based on an updated CEUS ground motion model published by EPRI [34]. The seismic hazard analysis for the PBAPS site also accounts for the effects of local site response for those structures, systems and components that are not founded on hard rock.

For PBAPS, both the SSC and GMC portions of the PSHA were developed as a result of a Senior Seismic Hazard Analysis Committee, Level 3 methodology (SSHAC, Level 3). In the case on the GMC, a SSHAC level 2 analysis was performed to update a prior Level 3 study. These studies satisfy the requirements of the PRA Standard related to the method of conduct of the PSHA, 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.

In the implementation of the CEUS-SSC model for the PBAPS site, all distributed seismic sources in the CEUS model were included in the PSHA calculations. By including all seismic sources in the analysis, the contributions of "near" and "far-field" earthquake sources to ground motions at PBAPS were considered. In addition, an effort was made to identify any local sources that may not have been included in the regional model, but none were identified. Additional information pertinent to the site response analysis was collected and assessed in developing the site response.

The CEUS-SSC described only includes earthquakes through 2008. For developing the PSHA at PBAPS, 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 rate of seismicity of sources in the CEUS-SSC model important to the seismic hazard at the PBAPS site. In addition, a separate seismicity catalog of non-tectonic (human-induced) earthquake was compiled and evaluated. It was concluded that an additional hazard analysis was not required for these sources.

The PSHA results are provided over an appropriately wide range of spectral frequencies and annual frequencies of exceedances. Uncertainties on the rock hazard are quantified, analyzed and reported as required in the PRA Standard [4]. The lower-bound magnitude chosen for the analysis is consistent with standard practice. The results include fractile and mean hazard curves, and median and mean uniform hazard response spectra.

The seismic hazard analysis for the PBAPS site included a site response analysis for structures, systems and components not founded on hard rock. Site-specific shear-wave velocity measurements based on historical information were used to inform the site response analysis. The analysis includes the effects of site topography, surficial geologic deposits and site geotechnical properties on ground motions at the site.

Both the aleatory and epistemic uncertainties have been addressed in characterizing the seismic sources, ground motion models and site response analyses. Epistemic uncertainty is represented by three shear wave velocity profiles and two sets of modulus reduction and damping curves. Aleatory variability is represented by 60 random realization 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 the SPID [2]. Correlation between properties is modeled when appropriate.

The later sections of this Appendix provide a summary of the Facts and Observations (F&O's) identified by the Peer Review Team that were classified as Findings. The Appendix also provides a resolution for each of these "findings".

SFR

- As required by the PRA Standard, all structures, systems and components (SSCs) that play a role in the seismic PRA were identified as candidates for subsequent seismic fragility evaluation. This was performed through the development of a Seismic Equipment List (SEL). As permitted by the Standard, inherently rugged components such as manual valves, check valves, cables and reset pushbuttons were screened out from further fragility evaluations.

  As required by the PRA Standard, seismic fragility evaluations were based on realistic seismic responses that the SSCs experience at their failure levels. To this end, new structural models were developed and used in the development of structural responses. These new models included either new finite element models or a combination of finite element and enhanced/refined lumped mass stick models. Soil-structure interaction (SSI) analysis was performed using median centered (best estimate) properties and considering variability in soil properties (best estimate, upper bound and lower bound). The input motion corresponded to the GMRS.

  For rock-founded structures including the Reactor Building, Turbine Building and the Radwaste Building, SSI analysis was performed to account for incoherency of the ground motion. Structural response analyses were performed for the best estimate (BE), upper bound (UB) and lower bound (LB) soil and structural properties. Both median centered and ~84th percentile structure response and in-structure response spectra were developed. Two sets of seismic response analyses were performed; an initial baseline analysis assuming fully cracked concrete and a supplementary analysis based on the PBAPS structures being essentially un-cracked at the GMRS level earthquake.

  A plant seismic walkdown and/or walk-by was performed for all SSCs credited in the PRA model as documented in the walkdown notebook. The overall walkdown effort was divided into four separate walkdowns: familiarization, outage, balance of plant and relays. Walkdowns focused on anchorage, lateral seismic support, functional characteristics and potential systems interactions for the SSCs in Unit 3. The walkdown observations are generally documented appropriately in support of the fragility analysis. The walkdowns also identified the potential for seismic-induced fires and floods. Subsequently, walkdown reviews of Unit 2 SSCs were performed to either confirm their similarity to the Unit 3 SSCs or to document their differences.

Consistent with the three quantifications performed to obtain SCDF and SLERF values, fragility values were calculated in three phases. For the first quantification, representative fragilities were calculated by performing a simplified scaling of the existing design basis calculations and accounting for available margin the designs of the SSCs. For the second quantification, refined fragilities were calculated for the top contributors determined from the initial quantification results and additional sensitivity studies. The top risk contributors were selected on the basis of ranking of FV values. The fragilities for these top contributors were calculated using the CDFM approach described in EPRI NP-6041-SL [14].

Based on the results of the second quantification and additional sensitivity analyses, a set of dominant contributors to seismic risk were identified based on FV importance measures for individual SSCs. For these dominant contributors (approximately 60 SSCs in 11 equipment classes), a more refined fragility analysis was generally performed. In general, these more refined analyses were performed using SOV approach though in some cases, more refined CDFM calculations were used to develop these refined fragilities.

The Standard [4] requires that the seismic fragility parameters be based on plant-specific data supplemented as needed by earthquake experience data, fragility test data and generic qualification test data. The peer review team found that this requirement was generally satisfied. The later sections of this Appendix provide a summary of the Facts and Observations (F&O's) identified by the Peer Review Team that were classified as Findings. The Appendix also provides a resolution for each of these "findings".

SPR

- As required by the PRA Standard, the logic model appropriately includes seismic initiating events and other failures including seismic-induced unreliability and unavailability failure modes, based on the Full Power Internal Events (FPIE) model, and human errors. The seismic PRA model was developed by modifying the FPIE PRA model to incorporate specific aspects of seismic analysis that are different from the FPIE. The seismic PRA model integrates the seismic hazard, the seismic fragilities, and the systems-analysis aspects appropriately to quantify core damage frequency and large early release frequency.

The quantification of the SPRA model was performed in three steps, consistent with the development of fragilities. Each quantification was used to identify the top contributors to overall seismic risk and the fragilities for those top contributors were refined and input to the subsequent quantification. In addition, a number of sensitivity studies were performed to provide an understanding of the impact of the various modeling and screening assumptions

The later sections of this Appendix provide a summary of the Facts and Observations (F&O's) identified by the Peer Review Team that were classified as Findings. The Appendix also provides a resolution for each of these "findings".

The review team concluded that the PBAPS seismic PRA model is of good quality and integrates the seismic hazard, the seismic fragilities, and the systems-analysis aspects appropriately to quantify core damage frequency and large early release frequency. The seismic PRA analysis was documented in a manner that facilitates applying and updating the SPRA model. Facts and observations identified as findings and SRs graded as Not Met are discussed in the following section along with a resolution for each.

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

Table A-1 presents a summary of the SRs graded as Not Met or less than Capability Category II, and the disposition for each. Table A-2 presents summary of the Finding F&Os that have not been closed through an NRC accepted process, and the disposition for each. As indicated in Table A-2, all Finding F&Os have been addressed or dispositioned, along with all SRs graded as Not Met.

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

SR	Assessed Capability Category	Associated Finding F&Os	Disposition to Achieve Met or Capability Category II
SHA			
SHA-I1	Not Met	5-7	Associated F&O has been resolved. SR is judged to be Met.
SFR			
[None]	N/A	N/A	N/A
SPR			
SPR-C1	Not Met	1-1, 1-2, 1-3, 1-8, 3-1	Associated F&Os have been resolved. SR is judged to be Met.

A.6. Summary of Technical Adequacy of the SPRA for the 50.54(f) Response

The set of supporting requirements from the ASME/ANS PRA Standard [4] that are identified in Tables 6-4 through 6-6 of the SPID [2] define the technical attributes of a PRA model required for a SPRA used to respond to implement the 50.54(f) letter. The conclusions of the peer review discussed above and summarized in this submittal demonstrates that the PBAPS SPRA model meets the expectations for PRA scope and technical adequacy as presented in RG 1.200, Rev. 2 [15] as clarified in the SPID [2].

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

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

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

The PBAPS SPRA reflects the as-built and as-operated plant as of the cutoff date for the SPRA, February 28, 2018 [61]. There are no permanent plant changes that have not been reflected in the SPRA model. See section A.9 for additional discussion.

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

The Owners Group performed a full scope peer review of the PBAPS internal events PRA and internal flooding PRA that forms the basis for the SPRA to determine compliance with ASME PRA Standard, RA-S-2008, including the 2009 Addenda A [55] and RG 1.200 [15]. This internal events PRA review was performed in 11/08/2010 – 11/12/2010 [56]. This review documented findings for all supporting requirements (SRs) which failed to meet at least Capability Category II. All of the internal events and internal flooding PRA peer review findings that may affect the SPRA model have been addressed.

The Owners Group performed a full peer review of the PBAPS SPRA the week of March 20, 2017. The results of this peer review are discussed above, including resolution of SRs not assessed by the peer review as meeting Capability Category II, and resolution of peer review findings pertinent to this submittal. The peer review team expressed the opinion that the PBAPS seismic PRA model is of good quality and integrates the seismic hazard, the seismic fragilities, and the systems-analysis aspects appropriately to quantify core damage frequency and large early release frequency. The general conclusion of the peer review was that the PBAPS SPRA is judged to be suitable for use for risk-informed applications.

- Table A-1 provides a summary of the disposition of SRs judged by the peer review to be not met, or not meeting Capability Category II.
- Table A-2 provides a summary of the disposition of the open SPRA peer review findings.
- Table A-3 provides an assessment of the expected impact on the results of the PBAPS SPRA of ‘Not Met’ SRs.

Table A-3 Summary of Impact of Not Met SRs		
SR #	**Summary of Issue**	**Impact on SPRA Results**
SHA-I1	A screening level analysis was used to assess soil-related failures related to seismic events including liquefaction, bearing capacity and slope stability. The analysis used to assess potential failures of the rock slope behind the plant lacks the appropriate rigor and depth to screen this hazard from further evaluation.	The analysis used to assess potential failures of the rock slope behind the plant was revised to respond to the Peer Review assessment of this SR. The approach used to initially assess the rock slope was extremely conservative. Based on the comment from the peer review team, the slope was evaluated using a less-conservative approach consistent with industry recommended methodologies. This revised analysis shows that there is significant margin with respect to failure even at earthquakes much higher than the GMRS. Thus, there is no potential for seismically induced failure of this slope that would have any impact on results of the SPRA.
SPR-C1	Conservative assumptions used in the development of the SPRA model contain enough conservatism that they bias insights such as relative risk significance of SSCs and operator actions. In addition, the conditional core damage probabilities and conditional large early release probabilities are likely too high.	The SPRA model has been revised to respond to the Peer Review assessment of this SR. Changes include: developing more detailed fragilities for certain items, incorporating additional operator actions, incorporating FLEX into the model, and enhancing the Level 2 accident sequence progression. As a result of these changes, SCDF reduced slightly and SLERF reduced significantly.

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

The PRA Standard [4] includes a number of requirements related to identification and evaluation of the impact of assumptions and sources of uncertainty on the PRA results. NUREG-1855 [28] and EPRI 1016737 [29] provide guidance on assessment of uncertainty for applications of a PRA. As described in NUREG-1855 [28], 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 PBAPS SPRA model quantification (see Section 5 of this submittal).
- Modeling uncertainties are considered in both the base internal events PRA and the SPRA. Assumptions are made during the PRA development as a way to address a particular modeling uncertainty because there is not a single definitive approach. Plant-specific assumptions made for each of the PBAPS SPRA technical elements are noted in the SPRA documentation that was subject to peer review, and a summary of important modeling assumptions is included in Section 5.
- Completeness uncertainty addresses scope and level of detail. Uncertainties associated with scope and level of detail are documented in the PRA but are only considered for their impact on a specific application. No specific issues of PRA completeness were identified in the SPRA peer review.

A summary of potentially important sources of uncertainty in the PBAPS SPRA is provided in Table A-4.

Table A-4 Summary of Potentially Important Sources of Uncertainty

PRA Element	Summary of Treatment of Sources of Uncertainty per Peer Review	Potential Impact on SPRA Results
Seismic Hazard	The PBAPS SPRA Peer Review Team stated that the equations used to calculate mean amplification factor and associated variability do not maintain separation between aleatory variability and epistemic uncertainty in the PSHA calculations.	PBAPS is a hard rock site. Any variation in uncertainty that may result from a different approach to combining the aleatory variability and epistemic uncertainty is negligible and more than offset by the variation in soil properties used in the analysis of the various structures. Reference Table A.2 Finding 5-1 for more information.
Seismic Fragilities	The PBAPS SPRA peer review team stated that the understanding of the appropriate reference earthquake should be confirmed. The reference earthquake should reflect the earthquake where most of the seismic risk originates. The selection of the	A sensitivity study was performed to determine the potential impact of using a larger reference earthquake as input to the SPRA. The impact of the sensitivity study is discussed in Section 5.7 and Table A.2 Finding 5-15.

Table A-4 Summary of Potentially Important Sources of Uncertainty

PRA Element	Summary of Treatment of Sources of Uncertainty per Peer Review	Potential Impact on SPRA Results
	reference earthquake can affect the realism in the seismic response due to non-linearities in the structures and soil/rock properties.	
Seismic PRA Model	The PBAPS SPRA peer review team had no issues with SPRA sources of uncertainty treatment and noted that the sources of uncertainty are discussed in Appendix I of the SPRA Quantification report. Appendix I of the SPRA Quantification report considers the various technical aspects of the SPRA development to identify key modeling uncertainties to investigate with sensitivity studies. The following are key areas of uncertainties identified: - Seismic hazard curve - Equipment functionality after battery depletion - Continued core cooling following venting or primary containment failure - SSC fragilities - Seismic human reliability analysis - Seismic-induced piping failure scenarios	As discussed in Section 5.6, candidate sources of SPRA model uncertainty are identified for the following: • PSHA • Accident Sequence analysis (e.g., Level 2) • Core Cooling success following Containment Venting (e.g., Very Small LOCA) • SSC Fragilities • Seismic HRA Section 5.7 discusses the sensitivity case results for the above sources of modeling uncertainty. The sensitivity cases evaluating potential variations in the PSHA supported that the CDF and LERF could range by a factor of 2 higher or lower.

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

The PBAPS SPRA reflects the plant as of the cutoff date for the SPRA, which was February 28, 2018 [61]. All modifications to the plant prior to the cutoff date that have an impact on the seismic PRA model have been included in the model. This includes implementation of FLEX. Note that the hardened containment vent system (HCVS) for NRC Order EA-13-109 [64] is not yet implemented at PBAPS. Table A-5 lists significant plant changes subsequent to the cutoff date and provides a qualitative assessment of the likely impact of those changes on the SPRA results and insights.

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

Description of Plant Change	Impact on SPRA Results
As part of ESEP, PBAPS committed to relocating two (2) RCIC Steam Leak Relays to a lower location in their host cabinet [36]. The Unit 3 relay was relocated in Fall of 2017. The Unit 2 relay is scheduled to be relocated in Fall of 2018.	The relocation of the RCIC Steam Leak relays are not considered in the SPRA model. The Unit 2 relay was not relocated prior to the cutoff date. In order to maintain symmetry for the Unit 2 and Unit 3 SPRA models, the Unit 3 relay relocation was conservatively not credited. Relocating the relays to a lower location in their host cabinet will increase the relay fragility. Therefore, the existing fragility value in the SPRA model is conservative. Any impact on the model results is expected to be minor due to the low risk significance of the relays.

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
SPR-B2 SPR-C1	1-1	The analysis of the FLEX human failure events (HFEs) was overly conservative. This was associated with the Not Met for SPR-C1	The EPRI screening method was used to adjust all HEPs, including those for FLEX. This results in overly conservative HEPs, especially for FLEX. As a result, FLEX is not credited in the model leading to significant cutsets that do not realistically represent the as-operated plant	Perform the human reliability assessment (HRA) for the FLEX actions using detailed analysis and incorporate FLEX into the model.	The HRA for FLEX was performed using detailed analysis and FLEX system fault tree logic was incorporated into the model (e.g., operator action to align FLEX generators to Unit 2 and Unit 3 load centers, operator action to align FLEX pumps for RPV makeup) with the more detailed HEPs, along with more refined HEPs for other operator actions. SPRA Quantification Notebook [45] is updated.
SPR-C1	1-2	FLEX is not credited in the model and the justification provided does not support retaining this conservatism. This was associated with the Not Met for SPR-C1	By not crediting FLEX in the model, important cutsets are generated that result in core damage where crediting FLEX would decrease the importance of these cutsets. Thus, crediting FLEX could result in changing the risk profile and providing additional risk insights.	Credit FLEX in the model	FLEX was incorporated into the SPRA model (e.g., credit aligning FLEX generators to Unit 2 and Unit 3 load centers, credit aligning FLEX pumps for RPV makeup), refined HEPs associated with the FLEX actions were included and refined fragilities were developed for FLEX components for input to the model. Fragility Report [25], Walkdown Report [33] and

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
					SPRA Quantification Notebook [45] are updated.
SPR-B2 SPR-C1	1-3	The analysis of human failure events (HFEs) with long time frames combined with the structure of the HRA bins was overly conservative. This was associated with the Not Met for SPR-C1	There are a number of HEPs that become guaranteed failures at high pga levels due to the use of the EPRI screening approach. This is overly conservative given time period available to complete the operator action. Furthermore, the structure of the bins themselves is conservative and there may be a disconnect between the bins used and the bins defined by EPRI.	Remove excess conservatism from the long timeframe HFEs. Revisit the bin structure and modify to more closely match plant failure modes. Determine a time frame at which the FPIE HEPs should not be adjusted, regardless of the magnitude of the earthquake. Perform detailed HRA on long term events that are risk-significant to get more realistic HEPs.	The Seismic HRA approach was revised in response to this F&O to reduce conservatisms. This included revising the SHRA bin scheme and performing detailed HEP calculations for risk-significant long-term HEPs. The SPRA HRA Notebook [42] and SPRA Quantification Notebook [45] have been updated.
SPR-A1	1-4	There are two aspects for identifying potential sources of seismically induced fires – seismic unique sources that would be identified in a walkdown and sources from a fire PRA that could be caused by a seismic event. The first was	Table G-1 of the EPRI SPRA Implementation Guide was used as the criteria to identify those FPRA fire ignition sources for consideration in the SPRA. While some of the sources deemed not significant can be considered obvious (e.g., hot work, low oil pumps), others are not (e.g., switchgear). Further, there are about 35 ignition sources identified in	Define an organized and logical approach to identify and screen potential fire sources. Include consideration of sources identified in the Fire PRA. Screen those that are obviously not risk significant. For others	The PBAPS SPRA approach to identification and assessment of postulated seismic-fire interactions follows the SPRA Implementation Guide 3002000709 and ASME/ANS RA-Sb–2013 Supporting Requirements SFR-E4, SFR-E-5 and SPR-B9. This includes use

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
		adequately addressed and the walkdown was sufficient to identify such sources. However, the basis for the pre-screening of generic sources considered in a fire PRA was not adequate given the current knowledge of the potential for such fires.	NUREG/CR-6850, Sup. 1, and there is no indication they were all considered either during the screening process or during the walkdown.	determine the potential for seismically induced fire and include those sources in the model.	of PBAPS fire PRA (FPRA) information as well as plant walkdowns and drawing reviews to identify sources for consideration. A formal question by the review team acknowledged the extent of the seismic-induced fire analysis, "*Overall, the discussion in the PBAPS documents contain a significant amount of information regarding the consideration of seismically-induced internal fires.*" Based on discussions with peer reviewers onsite and subsequently after receipt of the peer review report, the peer reviewers indicated their key issue was the need to explicitly disposition non-safety electrical cabinets that were not included in the SPRA or walked down.

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
					A list of PBAPS U-2 and U-3 non-safety electrical equipment ≥440VAC (consistent with input from utility personnel piloting EPRI 3002005289 [47]) not already assessed and walked down was developed (using PBAPS FPRA and plant information sources). This equipment was walked down in October 2017 by a seismic review team (including two SQUG-qualified senior experts) to identify conditions that would result in a seismically induced fire. Failure modes investigated included anchorage (sliding, overturning), II/I, inadequate cable slack, and cabinet internals. All the items on the October 2017 walkdown were screened from further consideration with the exception of two non-safety

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
					load centers in the U-3 turbine building and one non-safety MCC in the radwaste building. Postulated seismic-induced fire consequences were investigated based on information in the PBAPS Fire PRA model and Fire cable database and then sensitivity quantifications were performed using the SPRA model. Postulated seismic-induced fires from these three electrical cabinets have a limited impact on SSCs credited in the SPRA; conservatively assuming a 1.0 probability that seismic-induced fire would occur from each of these cabinets has a negligible impact on calculated SCDF and SLERF results. The PBAPS Walkdown Report [33] and the PBAPS Seismic PRA Methods Notebook [39] have been updated to capture these

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
					additional investigations and to incorporate additional discussions on the screening of various sources.
SPR-A5	1-5	There are cases where multiple fragility values are calculated for the same SSC but only the minimum value was input to the model, even if other values are close.	When multiple independent failure modes exist for an SSC and the fragilities are close together, the failure probability contributions for the separate events are additive. Eliminating failure modes for this case underestimates the contribution due to failure of the SSC.	Incorporate additional uncorrelated SSC failure modes that are close together. Justify the criterion used to define 'close' such that significant additional contributions to failure are considered. Add the missing failure modes or combine the failure modes into a single fragility curve.	The fragility calculations that used the SOV approach were reviewed. It has been shown in various studies that considering closely spaced modes can reduce seismic fragilities by about 15% or 20%. Calculations performed using the CDFM approach are not considered to be within this level of accuracy. For all components that were evaluated using the SOV approach, the fragilities for different failure modes were either not closely spaced or were correlated. Therefore, explicit consideration of closely spaced modes is not required. Fragility Report [25] is updated.
SPR-B4b	1-7	Post-earthquake actions to recover from relay chatter are included in the model but a single undeveloped	Relay chatter recovery actions vary as to location and difficulty. The use of a single HFE for relay recovery does	Evaluate the variation in relay chatter recovery actions and assess whether it is	All operator actions included in the model, including those associated with recovery from relay chatter were revisited.

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
		HEP is used to model recovery from all scenarios in which recovery from relay chatter is possible.	not provide a realistic representation of the variation in actions.	necessary to provide separate HFEs to cover the variation, based on the extent of variation in the HEPs	The PBAPS SPRA models separate operator actions to credit recovery from the following relay chatter induced failure modes: • EDG unavailability • 4kV Bus unavailability • HPCI or RCIC unavailability • ECW unavailability Operator action walkdowns were preformed to determine the difficulty of performing these actions. More refined HEPs were generated based on results of the walkdowns as well as interviews with operations and more detailed evaluations. These refined HEPs were input to the model and the model was re-quantified considering these revised HEPs along with other changes made as a result of the peer review team review. SPRA Human Reliability Analysis HRA Notebook [42] and the Walkdown Report [33] are updated.

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
SPR-C1	1-8	SR is judged to be Not Met and this F&O is written based on conservative assumptions that bias insights such as relative risk significance of SSCs and operator actions. In addition, the conditional core damage probabilities and conditional large early release probabilities are believed to be too high. This is the result of too much focus on F-V and sensitivity analysis of individual failures. This SR was considered Not Met	The results of the quantification, especially insights and mitigation capability, are biased toward conservatism. Conservatism include certain HEPs, operator actions related to certain failures, not including FLEX in the model, definition of LERF, actions related to loss of DC power and actions related to loss of injection. In addition, there exists an opportunity to refine fragility values for some components which has the potential to remove conservatism from the model. The V/H ratio (PSHA) may also be adding conservatism. Reduction in conservatism in the model will produce results that are more realistic and will provide greater insight into overall plant risk and vulnerabilities.	Implement refinements and reduce conservatism such that results and insights are consistent with plant capabilities.	All aspects of the model associated with operator actions were revisited to reduce conservatism and reflect actual plant capabilities. FLEX was incorporated into the model (e.g., credit aligning FLEX generators to Unit 2 and Unit 3 load centers, credit aligning FLEX pumps for RPV makeup) along with actions to recover from loss of injection and loss of DC power. Fragilities for top risk contributors were also revisited and conservatism was reduced from some fragility calculations (including the Conowingo Dam) to provide more realistic values. The model was re-quantified to incorporate these changes. The final results are considered to accurately reflect the plant capability and its as-operated condition. SPRA Quantification Notebook [45], SPRA HRA Notebook [42]

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
					and Fragility Report [25] are updated.
SFR-A2 SPR-C1	3-1	The approach used to determine which seismic failures are potentially significant to the results of the SPRA primarily focused on the Fussell-Vesely (FV) importance of individual SSCs. As a result, there is a lack of realism in a number of fragility calculations that potentially yields an overall unrealistic result. This was associated with the Not Met for SPR-C1.	Many of the cutsets and associated results have similar fragility values such that when the fragility of a single item is improved, there is essentially no change in CDF or LERF. Past SPRAs have shown that focusing on individual FV numbers and improving values for items one at a time does not necessarily change the dominant risk contributors. Fragilities for pairs or groups of components should be improved and input to the model and the combined impact be assessed to identify the dominant contributors. This should be combined with other improvements such as refined HEPs	Use an approach to determine which seismic failures are potentially significant that considers the combined impact of sets of failures.	Numerous improvements were made to the SPRA model to reduce conservatism and provide more realistic results. These included refining seismic HRA approaches for all post-initiator operator actions, adding additional operator actions to address loss of DC power and loss of injection, including FLEX in the model and refining SSC fragilities. All these improvements were added to the model and the model was re-quantified to account for all the improvements as a whole to obtain the combined effect of these reductions in conservatism. The final results are considered to accurately reflect the plant capability and its as-operated condition. SPRA Quantification Notebook [45] and Fragility Report [25] are updated.

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SR	F&O	Description	Basis	Suggested Resolution	Disposition
SHA-E2	5-1	The equations used to calculate mean amplification factor and associated variability do not maintain separation between aleatory variability and epistemic uncertainty in the PSHA calculations.	Epistemic uncertainty in the site response parameter should be incorporated into the PSHA in a manner that is consistent with the way it is treated in the evaluation of SSCs. That is, the epistemic uncertainty and aleatory variability should not be combined prior to the last step in the PSHA. While there will be no effect on the mean, the uncertainty in the soil hazard reflected by the fractile curves will be underestimated.	The treatment of aleatory variability and epistemic uncertainty in the site response component of the PSHA should be consistent with the SSC components	PBAPS is a hard rock site. All the major structures are founded on the hard reference rock except the DGB. The DGB is founded on a series of shear walls and piles that extend to hard rock. The 30 feet of soil between the bottom of the DGB and the rock surface has some slight impact on the horizontal stiffness of the piles. However, any variation in uncertainty that may result from a different approach to combining the aleatory variability and epistemic uncertainty is negligible and more than offset by the variation in soil properties used in the analysis of the various structures. Maintaining separation of aleatory variability and epistemic uncertainty through the determination of mean amplification factor and associated variability is not a requirement of the SPID and is based on a recent interpretation of the ASME

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SR	F&O	Description	Basis	Suggested Resolution	Disposition
					Standard. Since incorporation of this finding would have negligible impact on the results and since it is not a requirement of the SPID, no changes were made to the PSHA.
SHA-F2	5-2	No sensitivity analyses have been performed to evaluate the relative importance of site response parameters for FIRS 2 through FIRS 4	The key site response parameters for FIRS 2 through FIRS 4 include the shear wave velocity profile and modulus/damping curves. No sensitivity analyses were performed to evaluate the relative importance of the site response parameters for the soil hazard curves calculated for FIRS 2 through FIRS 4.	Perform sensitivity analyses to evaluate the relative importance of site response parameters of FIRS 2 through FIRS 4 and document the results in the PSHA	Comparisons of the amplification functions for FIRS 2 thru 4 at 1E-4 and 1E-5 for various branches of the logic tree adopted for the site response analyses were added to the PSHA report [6] to resolve this comment. Incorporation of this finding had no impact on the results. PSHA report [6] is revised.
SHA-G1	5-3	An idealized V/H scaling relation that envelops the available CEUS rock V/H scaling relationship is used to develop vertical spectra.	There is large uncertainty regarding V/H scaling factors for the CEUS. This was addressed by using an idealized V/H ratio that envelopes the available data. This adds conservatism to the results.	The epistemic uncertainty in V/H scaling factors should be addressed using a logic tree where the relative confidence in the available scaling relationships is reflected by the weights assigned to each branch.	The PSHA was revised to incorporate a logic tree as recommended by the peer review team. Since PBAPS is a hard rock site, full weight was given to the CEUS V/H relation in the logic tree. The result was essentially no change to the site response or to the FIRS. PSHA report [6] is revised.

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SR	F&O	Description	Basis	Suggested Resolution	Disposition
SHA-J2	5-5	Improvements are needed in the documentation to sufficiently document the process used in the PSHA.	Eight separate recommendations were made by the peer review team to improve the documentation of the process used in the PSHA. These recommendations are summarized below: 1.The specific references used to confirm the lack of local seismic sources should be listed in the PSHA report 2. The comparison made between the GMPEs from EPRI [34] and the GMPEs from the draft NGA-East report to establish that the EPRI [34] GMPEs remain valid in light of new available data should be documented in the PSHA. 3.More detailed information on the properties of the compacted backfill soils should be provided in the PSHA. In addition, the comparison made between the empirical curves from EPRI and Darendeli to account for uncertainty in the soil properties should be documented in the PSHA to justify the fact that only the EPRI curves were used. 4. NUREG/CR-6728 describes three variations of Approach 3 used for	Revise the PSHA to address the documentation issues identified in this finding.	The PSHA was revised to address various findings from the peer review team, including F&O 5-3 and the documentation findings identified in F&O 5-5. With regard to the findings in F&O 5-5, the following specific changes were made to the PSHA. 1. The specific references used were added to the PSHA. 2.The PSHA was revised to provide the comparison between the two GMPEs and show that the EPRI [34] curves remain valid. 3. Available data on the backfill soils was added and the comparison between the EPRI and Darendeli curves was provided. The comparison shows that the use of only EPRI curves is justified. 4. Approach 3 was used in all cases. No changes were made to the PSHA as a result of this recommendation.

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SR	F&O	Description	Basis	Suggested Resolution	Disposition
			combining amplification factors with the hard-rock hazard to develop soil hazard curves. The PSHA documentation should be more precise in describing which variation (3, 3A, or 3B) was used. The documentation notes that Approach 3 is used, but it appears that actually approach 3B was used. 5. The PSHA should be revised to be clear that guidance provided in Reg Guides, NUREGs, ISGs and similar documents are, in fact, guidance and not requirements. 6. Figures should be added to compare the mean seismic hazard curves for the seven spectral frequencies for GMRS/FIRS1 to FIRS 4 to simplify review of the PSHA. 7. The documentation in the PSHA should be revised to more accurately describe the purpose and scope of NGA-East and the discussion of NRCs NTTF Recommendation 2.1, seismic. 8. The discussion of the SSHAC process used in the EPRI Ground Motion Model Review Project should be revised to remove the word		5. The suggested changes were made to the wording in the PSHA. 6. All the information requested by the peer review team was already included in the PSHA report reviewed by the peer review team in a slightly different format. Including additional figures to provide the same information in a different format does not have any impact on the end results or any conclusions of the PSHA. Thus, no changes were made to the PSHA as a result of this recommendation. 7. The wording in the PSHA was revised as recommended by the peer review team. 8. The wording in the PSHA was revised as recommended by the peer review team. PSHA report [6] is updated.

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SR	F&O	Description	Basis	Suggested Resolution	Disposition
			"improved" in describing the SSHAC process used.		
SHA-I1	5-7	A screening level analysis was used to assess potential failures of the rock slope adjacent to the plant. The process used lacked the appropriate depth and rigor to justify screening out failure of this slope. This SR was considered Not Met.	An overly simplistic analysis was performed to screen out failure of this slope. One issue is that the analysis relied heavily on shear strength values of uncertain origin reported in the UFSAR. Significant developments in rock mass characterization have occurred in the past 50 years that should be recognized and used to develop updated shear strength parameters. Furthermore, the updated shear strength parameters should be based on the original laboratory test data, rather than subsequent interpretations, to the extent possible. A second issue is that the methodology used to perform the pseudo-static stability calculation is over simplistic and does not reflect current state of the practice (SOP) in this area.	The screening-level analyses used to evaluate the stability of the rock slope should be updated using (i) modern procedures for estimating the shear strength of the rock mass and (ii) modern procedures for performing the stability/deformation calculations. The screening should take into account that the evaluation is being performed in the context of a SPRA and must, therefore, clearly demonstrate that there is no potential contribution to CDF or LERF.	The analysis of the rock slope was revised to use a more rigorous approach consistent with the state of the practice. The revised analysis demonstrates that screening out potential failure of the rock slope was appropriate. A detailed review of the properties and the basis for the properties used in the analysis was conducted and it was determined that the values were appropriate for the type of analysis that was performed in the revised calculation. In addition, due to the significant capacity and margin against slope failure, reasonable variation in the properties used would have no impact on the results or conclusion of the analysis. Soil and slope stability evaluation [48] is updated.

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SR	F&O	Description	Basis	Suggested Resolution	Disposition
SFR-G2	5-8	Improvements are needed to the documentation to sufficiently document the process used to develop seismic fragilities and incorporate them into the SPRA model	A number of recommendations were made by the peer review team to improve the documentation of the process used in the PSHA. These recommendations are summarized below: 1. The guidance regarding the frequency range of interest needs to be revised to be consistent throughout the document. In addition, guidance needs to be added related to treatment of items sensitive to high-frequency inputs. 2. Fragility values for small, medium and large break LOCA are based on only pipe failures. These need to be assessed to determine if other failures (supports, anchors, vendor supplied hardware, etc.) would control the fragility values. 3. Documentation needs to be added to support the judgement that the recirc pumps are inherently rugged and have high capacities. 4. Not all the values contained in the quantification model are provided in the Fragility Report.	The various documents should be revised to incorporate the suggested changes related to the documentation provided in the Fragility Report and the FRANX table to facilitate review of the process used to develop input to the SPRA model.	The relevant documents were revised to address the recommendations identified in this finding. The following specific changes were made to the Fragility Report [25] to address the findings. 1. The criteria document was revised to provide clear guidance on the frequency range of interest. In addition, guidance was added to address components with natural frequencies between 20 Hz and about 40 Hz consistent with latest industry consensus. Calculations were reviewed and fragilities were adjusted if required. The new fragilities are used in the final quantification. 2. The design requirements associated with support, anchor and vendor supplied component design were reviewed and it was determined that basing the fragility on pipe failure was appropriate. Additional

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SR	F&O	Description	Basis	Suggested Resolution	Disposition
			5. The relay walkdown report needs to be revised to clarify the process used to screen relays. 6. The relay evaluation in the Fragility Report lacks the information needed to verify relay capacities. 7. An averaging technique was used in the initial stages of the project that biased the results toward the best-estimate structural properties, instead of equally weighting the different assumptions related to the structural properties. This process was changed and an appropriate weighting was used. This revised, more appropriate approach was used to develop inputs for all detailed and refined fragility calculations (CDFM and SOV calculations). However, this is not clear in the documentation and the Fragility Report should be revised to clearly state this. 8. There were some inconsistencies between the criteria document and the approach used for some of the SOV calculations. 9. Some information in the FRANX fragility table for equipment in the		discussion was added to the associated documentation. 3. Documentation was added to show that the in-line Recirc Pumps are rugged and have capacities as high as the piping. 4. A complete review of the FRANX input file compared to the Fragility Report was performed to ensure that all values were available in the Fragility Report and that the values match. 5. The relay walkdown report was revised as suggested by the peer review team (PRT) 6. The relay evaluation was updated to provide the basis for the capacity information used to determine fragilities for each relay. The document and page number for the source was added and an explanation was added to specifically state the basis for each capacity value.

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SR	F&O	Description	Basis	Suggested Resolution	Disposition
			EDG is not provided in the fragility table prepared by the fragility team. 10. The fragility value reported for Distributed Piping in the FRANX fragility table was not consistent with the value provided for distributed piping in the fragility table. The final fragility value used for Distributed Piping is associated with the fragility for fire protection sprinkler piping. 11. The final fragility table does not provide building fragilities. 12. In some cases, it was difficult to find the basis for the fragility value provided in the FRANX table. 13. Values in the FRANX table for some LOCA scenarios was not consistent with the data in the fragility table. 14. Documentation should be added to justify using the same fragility value for Small, Medium and Large LOCA. 15. Fragility associated with slope stability was not listed in the fragility table.		7. The Fragility Report was revised to clearly describe the process used. 8. The SOV calculations were reviewed and were either revised or annotated to describe any deviations from the criteria document. 9. A complete comparison between the FRANX input file and the Fragility Report was performed to ensure consistency and all issues were resolved. 10. Failure of distributed piping results in a flood of SEL components. In general, distributed piping has a higher capacity than the capacity of the neighboring components. Two conditions were identified where the fragility of the distributed piping could have lower fragilities than the components: piping with Victaulic couplings and sprinkler heads that could be damaged and lead to a flood. Bounding conditions for both cases were evaluated and it

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SR	F&O	Description	Basis	Suggested Resolution	Disposition
					was determined that the sprinkler head fragility controls. This fragility was used for the bounding flood case from the SPRA model. Consistency between the fragility value in the fragility report and the value in the FRANX input table was verified. 11. Building fragilities were added to the table 12. This was addressed during the FRANX and Fragility Report comparison 13. This was addressed during the FRANX and Fragility Report comparison 14. This documentation was added to the Fragility Report 15. Slope stability was added to the fragility table and shown as screened out based on a more refined analysis that showed significant seismic capacity.

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SR	F&O	Description	Basis	Suggested Resolution	Disposition
SFR-E1	5-9	No process was identified to verify that significant plant design changes have not impacted the results of previous walkdown information from USI A-46/IPEEE.	Non-intrusive walk-by inspections were conducted for SSCs previously walked down as part of the USI A-46 or IPEEE programs. The potential that plant modifications could have been made that would impact the results of the walkdown was apparently not addressed	Conduct a review of significant design changes since the time of USI A-46 and IPEEE to confirm that these changes had no impact on the previous walkdown results.	This finding was issued because the previous USI A-46 and IPEEE walkdowns were used as input in the decision to perform walk-bys instead of detailed walkdowns and these original walkdown records are not maintained current. The concern was that plant changes could have occurred that would have invalidated the conclusions of the previous walkdowns and would not have been identified. However, prior to performing the walkdowns, the fragility team retrieved current drawings and calculations for the SEL components. This current information was used to develop the fragilities. The modification process at PBAPS requires that drawings and calculations be updated to reflect plant modifications. Since these updated drawings and calculations were used as the basis for the fragilities, any plant changes since the USI A-46 and IPEEE

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SR	F&O	Description	Basis	Suggested Resolution	Disposition
					walkdowns would have been identified and considered in the evaluation of the item. Walkdown Report [33] is updated.
SFR-C5	5-11	The assessment of potential pounding between buildings did not consider the limitation in available "seismic shake space" due to the existence of elastomeric material between the buildings and ground motions higher than the GMRS.	The existence of an elastomeric material in the gaps between buildings will limit the allowable displacement to less than the ½" assumed in the analysis. The potential for impact at ground motions higher than the GMRS needs to be assessed taking into account the actual available gap between buildings.	A building interaction assessment needs to be performed using the actual available gap between buildings to determine the potential for building impact and the resulting effect on component fragilities. Ground motions higher than the GMRS should be considered.	All the structures in the Reactor Building Complex at PBAPS are founded on a common base mat resting on hard rock. This significantly limits the displacement between buildings. Within the Reactor Building Complex, seismic gaps exist at the interfaces between the Turbine Building, the Reactor Building, the turbine pedestals, and the Radwaste / Main Control Room Complex. The other structures are either stand-alone or are structurally interconnected. An analysis was performed to determine at what earthquake level the expected building displacements would exceed the available gap, taking the elastomeric material into account. The actual locations where the

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SR	F&O	Description	Basis	Suggested Resolution	Disposition
					impact could most likely occur were also identified. The impact would be expected to produce high-frequency “shock” accelerations that have minimal potential to damage components and can only result in chatter of relays and similar chatter-sensitive devices. Relay host cabinets in proximity to the impact points were reviewed to determine if the seismic level at which building impact would occur would result in a reduction in relay or relay cabinet fragility. A number of relay cabinets were determined to be located in proximity to the impact points. However, in all cases, the fragility of the cabinet was already lower than the fragility associated with the building impact. Therefore, the potential for building impact was determined to have negligible effect on any component fragilities. Fragility Report [25] is updated.

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SR	F&O	Description	Basis	Suggested Resolution	Disposition
SFR-E2	5-13	There were instances where potential seismic interactions were identified in the walkdown and documented in the walkdown report but were not addressed in the development of fragilities.	Several instances were identified by the peer review team where the walkdown notes identified seismic interactions that had the potential to affect the fragilities for some SEL components but where the potential adverse interactions were not closed in the walkdown report or addressed in the fragility calculations.	Provide an assessment for the items identified in the F&O finding to ensure there is no impact on the fragility for any SEL components.	The specific items listed in the peer review finding were resolved through further evaluation of the potential for adverse interactions. In addition, the entire walkdown report was reviewed to determine if there were other similar cases where potential items were not resolved. All identified issues were resolved and there was no impact on any fragilities for any SSEL components. Fragility Report [25] and Walkdown Report [33] is updated.
SFR-C1	5-15	The reference earthquake should reflect the earthquake where most of the seismic risk originates. The selection of the reference earthquake can affect the realism in the seismic response due to non-linearities in the structures and soil/rock properties.	The GMRS was selected as the reference earthquake for the PBAPS SPRA. The PBAPS structures were originally assessed to be cracked at the GMRS level and the Q1 fragilities that are included in the latest quantification model are developed based on cracked reinforced concrete structures and include a higher damping value. Subsequent to the initial quantification, it was determined that these structures were not sufficiently cracked at the GMRS to result in the structure	Exelon should confirm their understanding of the appropriate reference earthquake. Following that an assessment of the cracking at that reference earthquake level should be performed to define the appropriate structure frequency and damping for seismic response analysis. A seismic response analysis has	All improvements to the seismic fragilities and the PRA model suggested by the peer review team were incorporated into the PRA model. This included improved fragilities for certain components including the Conowingo Dam as well as inclusion of FLEX and additional operator actions. A final quantification of the model was

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SR	F&O	Description	Basis	Suggested Resolution	Disposition
			frequency shifting lower and the damping being increased to reflect significant cracking. For the subsequent quantification fragilities included in the latest model/quantification the linear structure response and lower damping values were used. Review of the latest quantification notebook seismic risk results show that the SCDF contributions are dominated by hazard intervals ranging between 0.4g to 0.75g. The hazard interval ranging from 0.5g-0.6g could be argued to represent the center of the risk contribution and could potentially be used to justify characterization of the appropriate reference earthquake. The SCDF associated with the GMRS level is relatively low. As such, the peer review team concludes that a realistic assessment of what reference earthquake is driving the risks should be conducted by the fragility team. The results could show that the current SCDF estimates are conservative based on the use of the uncracked building response. Some of the very low fragilities (e.g. the batteries) would likely merit this uncracked response. But many others could potentially have	already been performed for both the cracked and uncracked cases. As such, the effects of any potential changes to the refined fragilities associated with those SSCs that can be justified to use the updated reference earthquake could be relatively made (either as a fragility update or a sensitivity study). The reference earthquake should be approximately the point at which the CCDF reaches 50% of the total CDF.	performed, and the results were used to determine the appropriate reference earthquake. It was determined that for some high-risk contributors, significant cracking would occur in the structure prior to failure of the component. Thus, a higher reference earthquake would be appropriate for these items. Since a fully cracked model was available, fragilities were improved for risk-significant items with capacities above those at which significant cracking would occur. In-structure response spectra developed from the fully cracked building models were used as input to develop these improved fragilities. The following process was used to provide updated fragilities for high-risk contributors for the purpose of the sensitivity study.

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SR	F&O	Description	Basis	Suggested Resolution	Disposition
			the cracked response as the reference earthquake. Because a significant part of the risk is stemming from the larger hazard intervals, this could end up lowering the risk.		• If the HCLPF of the component was less than 1.5 times the cracking level, no updates to the fragility were provided. • If the HCLPF of the component was more than 2.0 times the cracking level, the fragility was updated to reflect inputs from the fully -racked building model • If the HCLPF of the component was between 1.5 and 2.0 times the cracking level, updates were made on a case-by-case basis, based on a review of frequency range of interest and other aspects of the analysis In all cases, only functional fragilities were updated. Anchorage fragilities were not

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SR	F&O	Description	Basis	Suggested Resolution	Disposition
					updated because it is judged that as significant cracking occurs, anchorage capacities would be reduced. In all cases, the fragility associated with the controlling failure mode (after implementing the process described above) was provided to the SPRA team. These updated fragilities were input to the PRA model and a sensitivity study was performed to determine the impact. The cracking that would result from a higher reference earthquake would decrease the frequency and increase the damping of the structure. This results in a decrease in overall building response as shown by comparing the responses from the un-cracked and cracked building models. Using the input from the fully cracked building models, using the GMRS to determine the structural response, yields an upper bound on fragility improvements and maximizes the impact of

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SR	F&O	Description	Basis	Suggested Resolution	Disposition
					structural cracking. This is considered to be appropriate for a sensitivity study to determine the potential impact of considering a larger reference earthquake. The results of the quantitative sensitivity study supported a very minor decrease in SCDF. In addition, there were minimal changes to the SCDF and SLERF risk importance measures. The sensitivity study was performed to ensure that considering the maximum improvements in component fragilities did not result in identifying additional plant vulnerabilities and risks and did not identify any new risk insights.
SFR-F2	5-16	There is a lack of justification for the seismic fragility provided for the Conowingo Dam.	The Conowingo Dam provides two functions related to operation of PBAPS. The first is to supply backup power to the plant and the second is to maintain water level at a sufficient height at the plant to allow for the water in the river to be used as the source of cooling water. Both functions	Perform additional research and perform walkdowns if required to obtain more design information related to construction of the dam. Use this information to develop a more refined	After further research, original drawings used to construct the dam were located. This information was used to develop a more detailed and realistic fragility for the dam to be used as input to the final quantification. As a result of

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			are high risk contributors. Based on judgement, both failure modes were assigned the same fragility as loss of offsite power. It is appropriate to assign the same fragility as loss of offsite power to the failure of the dam to provide power to the plant due to the design of the power distribution system from the dam to the plant. However, there is insufficient justification for using this same value for structural failure of the dam, resulting in loss of the normal (i.e. ultimate) heat sink, given the risk importance of this function.	and more realistic fragility for the structural failure mode associated with this dam.	this effort, a significant improvement in the fragility of the structural failure mode of the dam was obtained. Fragility Report [25] is updated.
SFR-F1	5-21	The use of a generic value from the Risk Assessment Standardization Project (RASP) handbook for fragility of distributed piping was not justified.	A very high fragility was assigned to failure of distributed piping, based on generic values from the NRC RASP handbook. No justification was provided for why the values from this handbook were appropriate for the piping in the PBAPS. Because this high fragility caused the distributed piping to have a very low contribution to risk, the fragility was never improved to a more realistic value.	Provide additional justification for using the value from the RASP handbook or develop a more appropriate and realistic plant-specific fragility for distributed piping and use this more realistic value in the quantification of the PRA model	In the initial quantification of the seismic PRA model, a generic value from the RASP handbook was assigned to distributed piping. However, even though distributed piping was not determined to be a significant contributor to seismic risk, a more refined and realistic value was used in subsequent quantifications. This more refined fragility was based on a bounding configuration determined via walkdown. This bounding

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					configuration was analyzed to determine an appropriate site-specific fragility. This value was used in subsequent quantifications. However, this evaluation was based on the calculation discussed in peer review finding 5-22. As a result of this finding, the calculation that developed the fragility for the distributed piping was revised to develop a more realistic fragility. The Fragility Report is updated [25].
SFR-F1	5-22	Inappropriate damping and unrealistic boundary conditions are used for the fragility assessment for fire protection piping.	The fragility calculation used 7% damping instead of 5% and the analysis considered the piping to be simply supported whereas the piping in the plant is all continuous.	Revise the calculation to address the identified issues.	A further review of information from the walkdown (including pictures and discussion of fire protection piping) was performed subsequent to the peer review and additional details were obtained with respect to fire piping dimensions to determine a more appropriate configuration to represent the bounding configuration for both distributed piping and fire protection piping. The relevant calculation was then revised to evaluate the more realistic

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					bounding piping configurations. Appropriate damping and boundary conditions were used in this reassessment. For the continuous piping, the calculation was not updated to assume continuous rather than simply supported. This is because the analysis is conservative and this configuration was determined to not be bounding even using a conservative assumption. For the cantilever piping segments, the length of the cantilever was significantly reduced to reflect actual observed field conditions. This configuration controls the fragility. Fragility Report [25] is updated.
SFR-B1	5-23	Assigning 1.8g peak spectral in-structure HCLPF capacity to cable trays based solely on meeting the SQUG GIP Limited Analytical Review (LAR) guidelines without defining the associated fragility based on the PGA is not appropriate.	Insufficient logic is provided for assigning a HCLPF capacity equal to 1.8g peak spectral in-structure HCLPF capacity to cable trays and thus screening them out of the quantification based solely on meeting the LAR guidelines is not justified.	Develop a fragility for cable trays referenced to the ground motion PGA value and show that the conclusion to screen out cable tray failure from the SPRA model quantification is appropriate based on a comparison of this	The calculation for cable trays was revised to demonstrate that cable trays have higher capacities than the associated equipment and therefore do not control the fragilities for any SEL components. Fragility Report [25] is updated.

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SR	F&O	Description	Basis	Suggested Resolution	Disposition
				fragility to fragilities for components determined the same way.	
SFR-A2	5-24	Sufficient information was not provided to justify the use of GERS relay capacity data for certain relays	The use of generic GERS capacity information requires that certain caveats associated with the generic information be verified for certain relays. There was no documentation that the caveats associated with use of this information had been verified.	Provide documentation to show that the caveats associated with the GERS for relay evaluations have been verified.	As a result of the peer review comments, fragility calculations for all relays have been reviewed to ensure that the associated caveats were met or appropriate justification was provided for the capacity used. Fragility Report [25] is updated.
SFR-D2	5-25	For equipment anchorage evaluations, the approach of using the ZPA for equipment with frequencies above 20 Hz is not always justifiable.	When equipment was judged to be “rigid”, the fragility was evaluated using the ZPA. In many cases, components were judged to be rigid when their natural frequency exceeded about 20 Hz. The issue is that the actual frequency may be in the range between about 20 Hz and about 40 Hz and that there is significant amplification above the ZPA in this frequency range. Using the ZPA may under-predict the seismic demand leading to non-conservative fragilities.	For anchorage evaluations, the seismic input should be based on the lesser of the acceleration at 20 Hz or the acceleration at the natural frequency of the equipment, if the frequency is above 20 Hz. That is, peaks above about 20 Hz do not need to be considered but accelerations above the ZPA at the natural frequency of the equipment do need to be considered if the frequency is between	The fragility calculations were revisited to determine how components with frequencies above about 20 Hz but less than about 40 Hz were handled. In cases where the ZPA was used for components that could not be shown to be rigid (have a natural frequency above about 40 Hz), the fragility was reviewed to determine the impact of considering the estimated frequency of the component taking into account existing conservatism in the calculation. This review was documented in a separate Appendix to the Fragility Report

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SR	F&O	Description	Basis	Suggested Resolution	Disposition
				about 20 Hz and about 40 Hz. Further, for stiff welded anchorages, justification should be provided to support that SSCs judged to be 'rigid' are well above 40 Hz.	[25]. Based on this review and the consideration of spectral accelerations in the range of 20 Hz to 40 Hz, any impact on the fragility values was negligible and thus no fragilities were revised with respect to this F&O.
SPR-D1	6-1	No review of industry SELs was performed, and some SSCs (DFOST, RWST and Torus Water Storage Tank) were identified that are credited in the SPRA model but were not included in the SEL.	This is a Finding as potential failure modes have not been included in the model, which can impact the results. A review of other SELs can help ensure completeness of the SEL.	Revise the SEL to incorporate the identified SSCs and perform a review of other SELs to ensure completeness.	The items identified by the peer review team were added to the SEL for PBAPS. In addition, the SEL from another BWR was reviewed to determine if there were other components that should be added to the PBAPS SEL. As a result of this review, approximately ten SSCs were added to the SEL and documented in the SEL Notebook [44].
SPR-E2	6-2	A very limited review of non-dominant cutsets was performed.	A review of non-dominant cutsets is required. A very limited review of non-dominant SCDF and SLERF cutsets was	Perform additional reviews of non-dominant cutsets for CDF and LERF per the guidance	Additional reviews have been performed for non-dominant cutsets consistent with industry guidance.

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
SPR-E6			performed which is judged to not meet the intent required by this SR.	provided by the Owners' Groups.	The SPRA Quantification Notebook [45] was updated.
SPR-B4	6-3	Screening for relays was treated differently from other SSCs, with some relays being screened based on HCLPF values and some screened based on risk significance obtained from previous quantifications. In addition, some relay chatter events were screened out based on the fact that they are similar to others that are already modeled. Thus, only representative scenarios were modeled rather than a complete list of relay chatter events. This has the potential to underestimate the total risk from relay chatter events. Additionally, some relays were screened out on the basis that other relays that affect the same equipment were included. However, certain unique characteristics of the relays	By screening out relay chatter scenarios on the basis that the impacts are similar to those for other relay chatter scenarios already modeled, the additional risk contributions of these scenarios are also excluded from the model. The true total risk contribution to the plant due to relay chatter may not be appropriately characterized.	Include all relay chatter events in the model or justify their exclusion based on accepted screening criteria such as high capacity.	The methodology for screening relays has been reviewed and better documented. Additional relay chatter scenarios have been added to the model as judged appropriate. If multiple relay chatter scenarios have similar impacts and they all have fragilities that would result in a non-negligible contribution to risk, then all relay chatter scenarios with similar fragilities would be modeled. The SPRA Fragility Modeling Notebook [43] was updated.

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

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		that were screened out could be missed in this approach.			
SPR-B1	6-5	The success criteria used for the SPRA plant response model are the same as those used in the Internal Events model, with modifications to account for the impacts of a Very Small LOCA. However, diesel generator failure to run times were modeled using 8.2 hours and were not extended to 24 hours.	Diesel Generator failure to run events were modeled using a mission time of 8.2 hours and were not extended to 24 hours, despite the fact that recovery from loss of offsite power was not credited in the SPRA model. The total failure probability associated with these events is therefore underestimated.	Adjust the mission time for EDG failure to run and common cause failure to run basic events from 8.2 hours to 24 hours. Confirm that there are no other SSCs credited for 24-hour operation in the SPRA model have shorter mission times in the model.	The mission time for the independent and common cause EDG failure to run failure modes has been increased to 24 hours. No other SSC basic event probabilities in the SPRA required mission time adjustments. The SPRA Methods Notebook [39] has been updated.
SPR-F1 SPR-F2	6-8	Improvements are needed to the documentation to sufficiently document the process used to develop the seismic PRA model and incorporate them into the SPRA quantification.	A number of recommendations were made by the peer review team to improve the documentation PRA model, particularly with respect to the quantification and seismic methods notebooks. These recommendations are summarized below: 1. Two different methods were used for detailed HRAs. One approach gives a lower HEP than the alternate approach, which is more commonly used. The documentation did not sufficiently explain why this was done. The SPRA team provided information to explain the basis and it was determined to be	Enhance the documentation as suggested.	Each item has been enhanced consistent with the discussions with the comments from the peer review. The SPRA Notebooks updated include the following: • SPRA Methods Notebook [39] • SPRA Event Tree Notebook [41] • SPRA Human Reliability Analysis Notebook [42] • SPRA Fragility Modeling Notebook [43]

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
			reasonable; this should be added to the HRA notebook (PB-PRA-020.004). 2. The discussion of T-1/2 and T-rec in §2.2.6 and Table 3.1 in the HRA notebook (PB-PRA-020.004) did not comport with the actual HRA method used in the SPRA. The SPRA team provided a detailed description of what was done and why and this was deemed to be reasonable. This should be added to the HRA notebook (PB-PRA-020.004). 3. The HRA notebook stated that no recovery actions were credited in the SPRA. However, the roadmap notes a couple of exceptions. These should be documented in the SPRA notebook. 4. The Quantification notebook and the Seismic Methods Notebooks document the quantification process and discuss the results. While the presentation of the previous Quantifications and results was good information to include, it also greatly complicated understanding what specifically was done for the Final Quantification. A discussion on what specifically is done to build the final model, without having to march through the earlier quantification discussions, would be very beneficial,		• SPRA Seismic Equipment List Development Notebook [44] • SPRA Quantification Notebook [45] • SPRA Contact Chatter Assessment [58]

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SR	F&O	Description	Basis	Suggested Resolution	Disposition
			particularly with regard to the final screening criteria applied. 5. A clear disposition of the relay chatter events screened from the model was not provided. It is difficult to identify what were the specific criteria for inclusion in the final model quantification. A summary should be provided to identify these criteria. Furthermore, additional column(s) in Table 3-2 or 3-3 of the Fragility Modeling Notebook to identify the impacts for each relay chatter event, and clearly state if recovery is credited, would greatly assist in the review of the model inputs. 6. In the component chatter report, add a discussion that describes how the contacts of the time delay relays were themselves considered. 7. With respect to the applicability of the Internal Events Success Criteria, it was determined that certain seismic-unique scenarios that could have been modeled were not required due to the conservative treatment of these scenarios in the model. This information needs to be added to the documentation.		

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SR	F&O	Description	Basis	Suggested Resolution	Disposition
			8. The rules for assigning groups and associated representative fragilities were not adequately documented.		