Enclosure A

Seismic Probabilistic Risk Assessment
in Response to 50.54(f) Letter with Regard to NTTF 2.1,
Beaver Valley Power Station Unit No. 1

(203 pages follow)

FIRST ENERGY NUCLEAR OPERATING COMPANY

Seismic Probabilistic Risk Assessment in Response to 50.54(f) Letter with Regard to NTTF 2.1

Beaver Valley Power Station Unit 1

APPROVALS

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Reviewed by: Robert W Drsek — 5-22-17
Robert Drsek — Date
PRA Engineering

Reviewed by: [signature] — 5-31-17
Ryan Keck — Date
Design Engineering

Approved by: [signature] — 5/31/17
Nathan Walker — Date
Supervisor, Nuclear Mechanical/Structural Engineering

Approved by: [signature] — 6/5/17
K. Raymond Fine — Date
Supervisor, Nuclear Analytical Methods

Approved by: [signature] Alvi — 6-5-17
Mohammed Alvi — Date
Project Manager

Approved by: [signature] — 6/1/17
Patrick Pauvlinch — Date
Manager, Design Engineering

ABS Consulting

RIZZO ASSOCIATES

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Beaver Valley Power Station, Unit 1 Seismic Probabilistic Risk Assessment in Response to 50.54(f) Letter with Regard to NTTF 2.1

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Prepared for:
FirstEnergy Nuclear Operating Company

ABSG Consulting Inc. • 300 Commerce Drive, Suite 200 • Irvine, California 92602

BEAVER VALLEY POWER STATION, UNIT 1 SEISMIC PROBABILISTIC RISK ASSESSMENT IN RESPONSE TO 50.54(F) LETTER WITH REGARD TO NTTF 2.1

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RIZZO ASSOCIATES

ABS Consulting
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Originator:

Eddie Guerra, P.E.
Director
RIZZO Associates

May 11, 2017
Date

Reviewer:

Farzin Beigi, P.E.
Consultant
RIZZO Associates

May 11, 2017
Date

Reviewer:

Donald J. Wakefield
Donald J. Wakefield
Senior Consultant
ABSG Consulting Inc.

May 11, 2017
Date

Principal:

Nish Vaidya
Nishikant R. Vaidya, Ph.D., P.E.
Vice President
RIZZO Associates

May 11, 2017
Date

Approver:

Thomas R. Roche, P.E.
Vice President
ABSG Consulting Inc.

May 11, 2017
Date

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BEAVER VALLEY POWER STATION, UNIT 1

SEISMIC PROBABILISTIC RISK ASSESSMENT IN RESPONSE TO 50.54(F) LETTER WITH REGARD TO NTTF 2.1

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 (Reference 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 Beaver Valley Power Station, Unit 1 (BVPS-1) 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 NTTF Recommendation 2.1: Seismic" (Reference 2), and previously submitted to NRC (Reference 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 Hertz (Hz) range, and a seismic risk assessment is required. A seismic PRA (SPRA) has been developed to perform the seismic risk assessment for BVPS-1 in response to the 50.54(f) letter, specifically Item (8) in Enclosure 1 of the 50.54(f) letter.

This report describes the SPRA developed for BVPS-1 and provides the information requested in Item (8)B of Enclosure 1 of the 50.54(f) letter and in Section 6.8 of the SPID. The SPRA model has been peer reviewed (as described in ***Appendix A***) and found to be of appropriate scope and technical capability for use in assessing the seismic risk for BVPS-1, identifying which structures, systems, and components (SSCs) are important to seismic risk, and describing plant-specific seismic issues and associated actions planned or taken in response to the 50.54(f) letter.

This report provides summary information regarding the SPRA as outlined in ***Section 2.0***.

The level of detail provided in the report is intended to enable the NRC to understand the inputs and methods used, the evaluations performed, and the decisions made as a result of the insights gained from the BVPS-1 SPRA.

2.0 INFORMATION PROVIDED IN THIS REPORT

The following information is requested in the 50.54(f) letter (Reference 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 bin, including importance measures (e.g., Risk Achievement Worth, Fussell-Vesely (FV), and Birnbaum).
2. A summary of the methodologies used to estimate the SCDF and large early release frequency (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 Paragraph 1 through Paragraph 6, regarding the seismic hazard evaluation reporting, also apply, but have been satisfied through the previously submitted BVPS-1 Seismic Hazard Submittal (Reference 3). Further, 50.54(f) letter Enclosure 1 Paragraph 9 requests information on the spent fuel pool. This information has been submitted separately (Reference 86).

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

- Seismic Hazard Analysis
- Seismic Structure Response and SSC Fragility Analysis
- Systems/Accident Sequence (Seismic Plant Response) Analysis
- Risk Quantification

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

The BVPS-1 SPRA and associated documentation has been peer reviewed against the PRA Standard in accordance with the process defined in Nuclear Energy Institute (NEI)-12-13 (Reference 5), as documented in the BVPS-1 SPRA Peer Review Report (Reference 6). The BVPS-1 SPRA, complete SPRA documentation, and details of the peer review are available for NRC review.

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

The content of this report is organized as follows:

Section 3.0 provides information related to the BVPS-1 seismic hazard analysis.

Section 4.0 provides information related to the determination of seismic fragilities for BVPS-1 SSCs included in the seismic plant response.

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

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

Section 7.0 provides references.

Section 8.0 provides a list of acronyms.

Appendix A provides an assessment of SPRA Technical Adequacy for Response to NTTF 2.1 Seismic 50.54(f) Letter, including a summary of BVPS-1 SPRA peer review.

TABLE 2-1
CROSS-REFERENCE FOR 50.54(F) ENCLOSURE 1 SPRA REPORTING

50.54(f) Letter Reporting Item	Description	Location in this Report
1	List of the significant contributors to SCDF for each seismic acceleration bin, including importance measures	***Section 5.0***
2	Summary of the methodologies used to estimate the SCDF and LERF	***Section 3.0, Section 4.0, and Section 5.0***
2i	Methodologies used to quantify the seismic fragilities of SSCs, together with key assumptions	***Section 4.0***
2ii	SSC fragility values with reference to the method of seismic qualification, the dominant failure mode(s), and the source of information	***Table 5-9*** provides fragilities (Am and beta) and failure mode information, and method of determining fragilities for the top risk significant SSCs based on standard importance measures such as Fussel-Vesely (F-V). Seismic qualification reference is not provided as it is not relevant to development of SPRA
2iii	Seismic fragility parameters	***Table 5-9*** provides fragilities (Am and beta) information for the top risk significant SSCs based on standard importance measures such as F-V.
2iv	Important findings from plant walkdowns and any corrective actions taken	***Section 4.2*** address 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	***Section 5.1*** and ***Section 5.2*** provide this information
2vi	Assumptions about containment performance	***Section 4.3*** and ***Section 5.5*** address containment and related SSC performance
3	Description of the process used to ensure that the SPRA is technically adequate, including the dates and findings of any peer reviews	***Appendix A*** describes the assessment of SPRA technical adequacy for the 50.54(f) submittal and results of the SPRA peer review
4	Identified plant-specific vulnerabilities and actions that are planned or taken	***Section 6.0*** addresses this

TABLE 2-2
CROSS-REFERENCE FOR ADDITIONAL SPID SECTION 6.8 SPRA REPORTING

SPID SECTION 6.8 ITEM [1] DESCRIPTION	LOCATION IN THIS REPORT
A report should be submitted to the NRC summarizing the SPRA inputs, methods, and results.	Entirety of the submittal addresses this.
The level of detail needed in the submittal should be sufficient to enable NRC to understand and determine the validity of all input data and calculation models used.	Entirety of the submittal addresses this. The template attempts to identify key methods of analysis and referenced codes and standards.
The level of detail needed in the submittal should be sufficient to assess the sensitivity of the results to all key aspects of the analysis.	Entirety of the submittal addresses this. Results sensitivities are discussed in the following sections: • ***Section 5.7*** (SPRA Model Sensitivities) • ***Section 4.4*** Fragility Screening (Sensitivity)
The level of detail needed in the submittal should be sufficient to make necessary regulatory decisions as a part of NTTF Phase 2 activities.	Entirety of the submittal 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 (Reference 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 BVPS-1 SEISMIC HAZARD AND PLANT RESPONSE

The BVPS is a soil site located in Shippingport Borough on the south bank of the Ohio River in Beaver County, Pennsylvania, in the Appalachian Plateau Province. The bedrock in the area is the Allegheny formation of Pennsylvanian age consisting of shale and sandstone with several interbedded coal seams. The bedrock is overlain by about 100 feet (ft) of alluvial granular terraces that formed during the Pleistocene. Plant grade is elevation (EL) 735 ft and the top of bedrock is at approximate EL 625 ft.

Subsequent to the March 2014 submittal, the BVPS seismic hazard for hard-rock site conditions was updated to address SPRA peer review comments; this updated is summarized in ***Section 3.1.1***. The derivation of Foundation Input Response Spectra (FIRS) is completed for several elevations corresponding to the base of the critical structures located at the BVPS Site. The site response geotechnical model used to derive the FIRS is described in ***Section 3.1.1.2***, with site response analysis results described in ***Section 3.1.1.3***. The seismic hazard results used for the SPRA are described in ***Section 3.1.3***, while the derivation of horizontal and vertical FIRS are described in ***Section 3.1.4***.

3.1 SEISMIC HAZARD ANALYSIS

This section discusses the seismic hazard methodology, presents the final hard-rock seismic hazard results used in the SPRA, the site geotechnical model used to derive the FIRS, the site response analysis results, 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. More detailed information regarding the BVPS Site Probabilistic Seismic Hazard Analysis (PSHA) hazard was provided to NRC in the seismic hazard information submitted to NRC in response to the NTTF 2.1 Seismic information request (Reference 3) and can be found in Reference 23.

3.1.1 Seismic Hazard Analysis Methodology

For the BVPS-1 SPRA, the quantification of the seismic hazard utilizes RIZZO's in-house software, RIZZO-HAZARD (Reference 19). This software uses the characterization of seismic sources (NRC, 2012b) and ground motion models (GMM) (EPRI 2013a, referred to as the EPRI GMM update) to estimate the annual exceedance frequencies for various levels of pseudo- S_A at different spectral frequencies.

The final PSHA results reflect the resolution of SPRA peer review interactions as documented in peer review Facts and Observations (F&O). The specific resolution summaries are provided in ***Appendix A***. The final PSHA and supporting documentation includes the following elements addressing the peer review F&Os:

- Enhanced discussion of the potential for induced or triggered earthquakes and the impact of these earthquakes on the seismic hazard for the BVPS Site.
- Quantitative assessment of seismicity that has occurred since the end of 2008, the cut-off date for the earthquake catalog used to assess earthquake recurrence rates and maximum magnitudes (NRC, 2012b).
- Modifications to the scripts used to combine seismic hazard curves for hard-rock site conditions and updating the hard-rock mean and fractile hazard curves. This resulted in essentially no change to the mean hazard, and only minor changes to fractile hazard curves on which the SPRA is based.
- Enhanced assessment of site response amplification factor epistemic uncertainty to define the input for developing the soil hazard curves. Based on this assessment the soil hazard curves (mean and fractiles) were derived and used to develop FIRS at each foundation elevation.
- Assessment of the variance contribution to the total variance for each of the seismic hazard input parameters. This assessment quantifies which seismic hazard input parameter(s) dominates the epistemic uncertainty in seismic hazard for several mean annual frequencies of exceedance.
- Updating the approach used to assess vertical-to-horizontal ground motion ratios resulting in some reduction in the vertical ground motions at each foundation elevation on which the SPRA is based.

3.1.1.1 Hard-Rock PSHA Results

The hard-rock PSHA hazard curves at the BVPS Site are obtained for seven response spectral frequencies (100 Hz [equivalent to PGA], 25 Hz, 10 Hz, 5 Hz, 2.5 Hz, 1 Hz, and 0.5 Hz). In addition to the mean, the associated fractile (5 percent, 15 percent, 50 percent (median), 85 percent, and 95 percent) hazard curves are also obtained. ***Figure 3-1*** and ***Table 3-1*** present the PGA hard-rock hazard curves; the full set of hazard curves at the seven spectral frequencies associated with the hard-rock Ground Motion Model (EPRI 2013a) can be found in Reference 23.

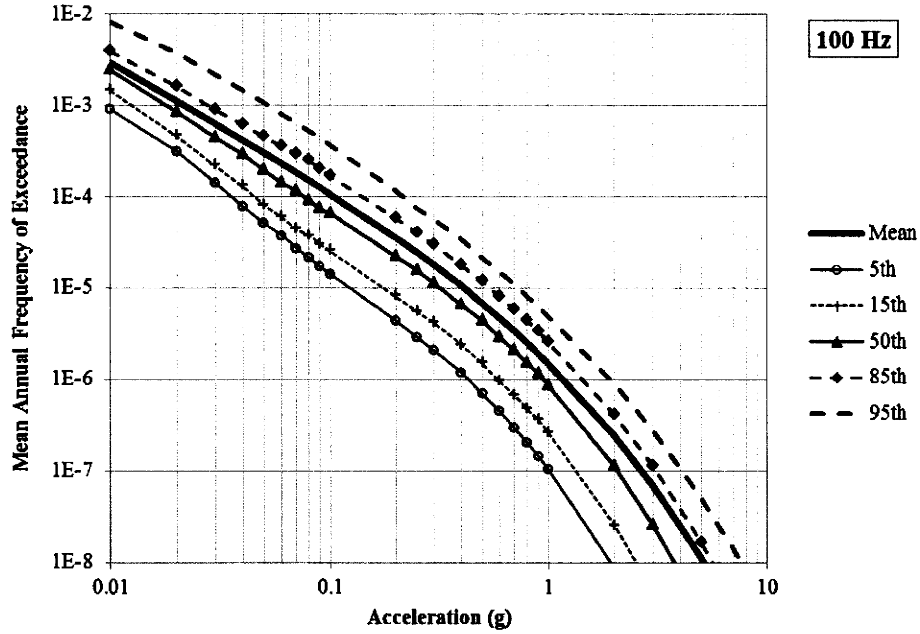


FIGURE 3-1
100 HZ SA MEAN AND FRACTILE HAZARD CURVES AT THE BVPS SITE FOR HARD-ROCK SITE CONDITIONS

The events controlling the hard-rock hazard provide the basis to develop smoothed UHRS at hard-rock based on the predicted hazard at the seven spectral frequencies. These controlling events are obtained by deaggregating the rock hazard for 1E-4, 1E-5, and 1E-6 mean annual frequency of exceedance (MAFE) into magnitude and distance bins following recommendations in Reference 24. The deaggregation results are used to identify controlling earthquakes at each MAFE.

TABLE 3-1
100 HZ S_A MEAN AND FRACTILE HAZARD CURVES AT THE BVPS SITE FOR HARD-ROCK SITE CONDITIONS

GROUND MOTION LEVEL (g)	ANNUAL PROBABILITY OF EXCEEDANCE					
	MEAN	5% FRACTILE	16% FRACTILE	50% FRACTILE	84% FRACTILE	95% FRACTILE
0.01	2.96E-03	9.06E-04	1.46E-03	2.43E-03	3.93E-03	8.13E-03
0.02	1.14E-03	3.16E-04	4.65E-04	8.56E-04	1.60E-03	3.80E-03
0.03	6.38E-04	1.44E-04	2.22E-04	4.50E-04	9.01E-04	2.18E-03
0.04	4.19E-04	7.85E-05	1.31E-04	2.90E-04	6.14E-04	1.47E-03
0.05	3.02E-04	5.10E-05	8.13E-05	1.95E-04	4.60E-04	1.06E-03
0.06	2.31E-04	3.77E-05	5.91E-05	1.44E-04	3.55E-04	8.00E-04
0.07	1.84E-04	2.71E-05	4.42E-05	1.16E-04	2.93E-04	6.36E-04
0.08	1.50E-04	2.14E-05	3.74E-05	9.04E-05	2.52E-04	5.29E-04
0.09	1.26E-04	1.72E-05	3.01E-05	7.53E-05	1.99E-04	4.45E-04
0.1	1.07E-04	1.43E-05	2.55E-05	6.56E-05	1.71E-04	3.67E-04
0.2	3.59E-05	4.46E-06	8.15E-06	2.22E-05	5.86E-05	1.17E-04
0.25	2.47E-05	2.94E-06	5.57E-06	1.55E-05	4.02E-05	7.57E-05
0.3	1.80E-05	2.09E-06	4.20E-06	1.13E-05	3.02E-05	5.62E-05
0.4	1.06E-05	1.20E-06	2.39E-06	6.63E-06	1.79E-05	3.42E-05
0.5	6.90E-06	7.07E-07	1.51E-06	4.42E-06	1.18E-05	2.15E-05
0.6	4.76E-06	4.51E-07	9.51E-07	2.94E-06	8.04E-06	1.52E-05
0.7	3.43E-06	2.95E-07	6.74E-07	2.09E-06	5.85E-06	1.09E-05
0.8	2.55E-06	2.04E-07	4.74E-07	1.53E-06	4.40E-06	8.12E-06
0.9	1.95E-06	1.46E-07	3.66E-07	1.16E-06	3.40E-06	6.32E-06
1	1.52E-06	1.05E-07	2.63E-07	8.72E-07	2.58E-06	4.92E-06
2	2.39E-07	8.39E-09	2.50E-08	1.13E-07	4.11E-07	8.88E-07
3	6.74E-08	1.34E-09	4.73E-09	2.58E-08	1.12E-07	2.74E-07
5	1.10E-08	7.81E-11	3.63E-10	2.98E-09	1.64E-08	4.94E-08
6	5.46E-09	2.61E-11	1.23E-10	1.22E-09	7.62E-09	2.54E-08
7	2.95E-09	9.07E-12	4.81E-11	5.46E-10	4.02E-09	1.41E-08
8	1.70E-09	3.89E-12	2.09E-11	2.72E-10	2.18E-09	8.16E-09
9	1.04E-09	1.58E-12	9.49E-12	1.43E-10	1.26E-09	4.94E-09
10	6.60E-10	7.21E-13	4.67E-12	7.41E-11	7.52E-10	3.25E-09

Because there is a significant contribution to hazard at low frequencies from distant earthquakes, the mean magnitude and distance are identified for the overall hazard and broken down by distance less than and greater than 100 km. ***Table 3-2*** identifies the controlling events in terms of the respective mean magnitude and distance for each of the distance bands. For the case in which contribution to hazard is examined separately for distance less than and greater than 100 km, the weight provided in ***Table 3-2*** represents the relative contribution to hazard from each distance range.

TABLE 3-2
CONTROLLING EARTHQUAKES FOR THE BVPS SITE

HAZARD	CONTROLLING EARTHQUAKE							
	OVERALL HAZARD R> 0 km		HAZARD FROM R< 100 km			HAZARD FROM R> 100 km		
	MAGNITUDE (M)	DISTANCE (km)	MAGNITUDE (M)	DISTANCE (km)	WEIGHT	MAGNITUDE (M)	DISTANCE (km)	WEIGHT
1E-4 MAFE 0.5 Hz	7.4	549	6.3	32	0.0941	7.5	737	0.906
1E-4 MAFE 1.0 Hz - 2.5 Hz	6.6	139	5.9	31	0.415	7.1	399	0.585
1E-4 MAFE 5.0 Hz - 10.0 Hz	5.9	46	5.7	31	0.777	6.4	176	0.223
1E-4 MAFE 25 Hz	5.8	41	5.7	30	0.829	6.3	168	0.171
1E-5 MAFE 0.5 Hz	7.3	292	6.5	27	0.252	7.6	651	0.748
1E-5 MAFE 1.0 Hz - 2.5 Hz	6.4	43	6.1	21	0.734	7.2	337	0.266
1E-5 MAFE 5.0 Hz - 10.0 Hz	5.9	17	5.8	16	0.967	6.9	163	0.0331
1E-5 MAFE 25 Hz	5.8	15	5.8	14	0.978	6.9	160	0.0221
1E-6 MAFE 0.5 Hz	7.0	70	6.7	23	0.623	7.5	452	0.377
1E-6 MAFE 1.0 Hz - 2.5 Hz	6.4	18	6.4	15	0.936	7.3	216	0.0635
1E-6 MAFE 5.0 Hz - 10.0 Hz	6.1	11	6.0	11	0.994	7.4	157	0.0061
1E-6 MAFE 25 Hz	6.0	10	6.0	10	0.996	7.4	155	0.00394

Note:

"Weight" is the percent contribution to overall hazard for the given distance range

Response spectral shapes for the controlling earthquakes are determined, following recommendations in Reference 77 for Central and Eastern United States (CEUS) earthquakes. Equally weighted single- and double-corner spectral shapes from Reference 77 are scaled up to the UHRS to define the controlling earthquake response spectra. Final hard-rock smoothed UHRS are determined by using the controlling earthquake spectral shape to interpolate and extrapolate the UHRS at response spectral frequencies other than those for which the GMM provides values.

For response frequencies less than 0.5 Hz, the controlling earthquake response spectrum for distances greater than 100 km is used. Similarly, for response frequencies between 0.5 Hz and 2.5 Hz, 2.5 Hz and 10 Hz, and greater than 10 Hz the controlling earthquake response spectra for 1.75-Hz S_A hazard, 7.5-Hz S_A hazard, and 25-Hz S_A hazard are used, respectively. The smoothed UHRS is derived for 36 spectral frequencies, which meets the minimum number of structural frequencies defined in Reference 24. ***Figure 3-2*** shows the smoothed UHRS with a 1E-4 MAFE.

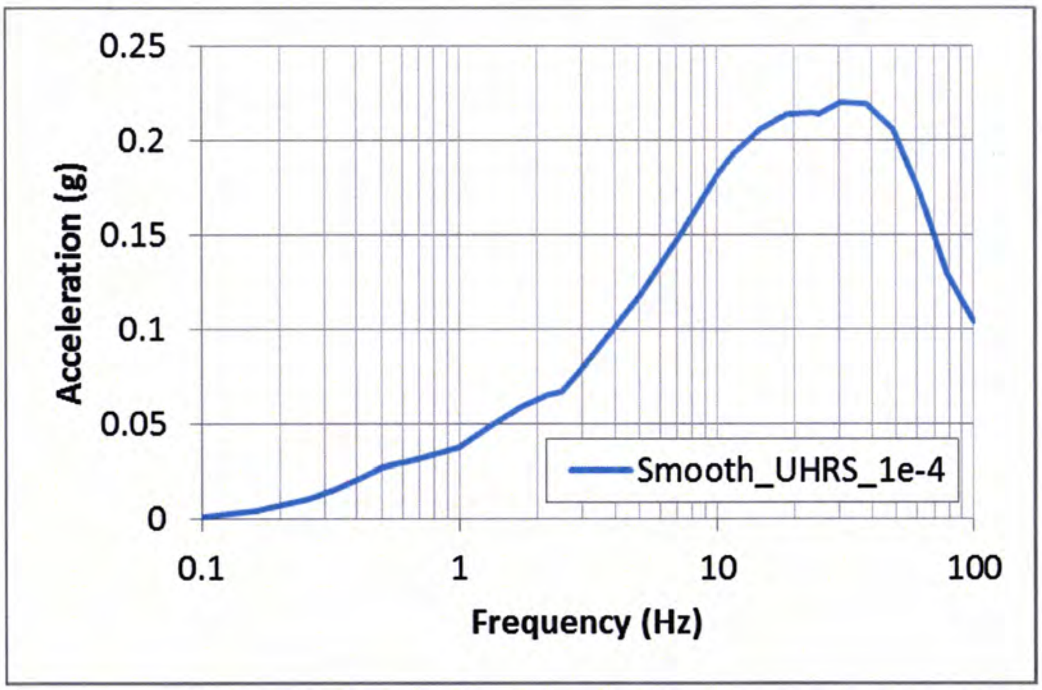


FIGURE 3-2
BVPS SITE 1E-4 MAFE SMOOTHED UNIFORM HAZARD
RESPONSE SPECTRA AT HARD-ROCK

3.1.1.2 Site Response Analysis Geotechnical Model

The BVPS Site is located in the Ohio River Valley, a flat-bottomed, steep-walled valley constructed by erosion. Bedrock underlying the BVPS Site and forming the hills, which rise to an elevation of about 1,100 ft adjacent to the BVPS Site to the north and south of the Ohio River Valley, is characterized by sandstones and shales interbedded with several thin coal seams and occasional thin limestone beds of the Pennsylvanian age Allegheny formation.

The terrace material at the BVPS Site, overlying bedrock, is characterized by three levels; high, intermediate, and low. The ground surface of the high terrace ranges between elevations (EL) 740 ft to EL 730 ft. The high terrace is composed of granular material mostly gravel and sand with some cobbles and rock fragments. The intermediate terrace ground surface elevation is approximately at EL 700 ft to EL 685 ft. The intermediate terrace is the result of flood control projects, which lowered the river level during the 1930s. The upper soils of the intermediate terrace consist of medium clays, which extend to about EL 660 ft. The low terrace being the most recent and closest to the river is located at a zone having a ground surface EL 675 ft towards the north. The shallow soils consist of soft clay and silt sediments of river showing some organic content.

The plant structures are located upon the high terrace of alluvial gravels. The nominal station grade is EL 735 ft. The ground surface grade elevation for the shared BVPS-1 and BVPS-2 Intake Structure is EL 675 ft.

The site response analysis is completed for several elevations corresponding to the bases of the critical structures located at the BVPS Site. Representative foundation elevations are selected for site response analyses considering that: 1) foundation elevation varies for some plant structures and 2) some plant structures are founded at similar elevations. Therefore, elevations for which site response analyses are performed may not coincide exactly with foundation elevations but are within a few feet. The approximations in elevation have a negligible effect on the structural response. These structures and representative foundation elevations are:

- EL 681: BVPS-1 Reactor Building (RCBX)
- EL 735: BVPS-1 Diesel Generator Building (DGBX)
- EL 723: BVPS-1 Fuel Handling/Decontamination Building (FULB)* and Safeguards Building (SFGB)
- EL 713: BVPS-1 Auxiliary Building (AXLB), Service Building (SRVB) and Main Steam Cable Vault (MSCV)
- EL 637: BVPS-1/BVPS-2 shared Intake Structure (INTS)

*Note: the Fuel Handling Building and Decontamination Buildings are separate structures separated by seismic shake spaces but are connected by steel superstructures and as a result are evaluated in the same seismic analysis model. The two buildings are herein referred to as the Fuel Handling/Decontamination Building, or FULB.

The quantification of site amplification of hard-rock motions takes into account the site-specific shear-wave velocity profile and other relevant dynamic properties for the site geologic material. These are based on available licensing documents and other relevant studies. Aleatory and epistemic uncertainties in the quantification of site amplification are explicitly addressed by defining alternative shear-wave velocity profiles, alternative shear modulus reduction and damping characteristics of the geologic materials, site attenuation (kappa), and the inherent random variation in these parameters.

Two conditions influence the site amplification factors (AF) for the BVPS Site: there is about 15 ft of compacted structural backfill surrounding several of the buildings and there is a significant V_S contrast between the soil materials at the site (compacted structural backfill and the terrace deposits) relative to the underlying sedimentary rock. Because of these two conditions, the calculation of the AFs at the various building elevations account for the potential influence of the soil confinement that surrounds the building. Guidance provided by Reference 25 accounts for these conditions.

The site response analysis for most of the structures at the BVPS Site is based on the full soil column extending from hard rock to plant grade (EL 735). The full set of strain iterated properties are retained for each of the layers modeled. The geologic column is then truncated at the appropriate building elevations and the site response analysis is repeated using the strain iterated properties from the full column, with no further strain iteration permitted. Because the soil column for the BVPS Site INTS is different, a second soil profile is developed for that structure, and the process outlined above is repeated.

The methodology described in Reference 2 guides the site response analysis. A logic tree is used to assess the epistemic uncertainties in site response input parameters, which includes the following:

- Hard-rock input ground motions are developed for two seismic source models with equal weights. The seismic source model is based on the point source model and uses both single-corner and double-corner input assumptions (Tables B-4 and B-6 of Reference 2).
- Use of three alternative base-case velocity profiles (BE [P1], LR [P2], and UR [P3]) to represent the shear-wave structure of materials underlying the Site.
- For each base-case profile, use of two scenarios to represent potential strain degradation of material properties of the Paleozoic rocks: materials behave nonlinearly in the top 500 ft of rock and linearly below the top 500 ft of rock to the profile base, and materials behave linearly for the whole profile.

The site parameter kappa describes the damping considered in the site response analysis. In the context of Reference 2, kappa is the profile damping contributed by both intrinsic hysteretic damping, as well as scattering due to wave propagation in heterogeneous material. The total site kappa consists of the kappa associated with the near-surface profile and kappa for the half-space; i.e., reference rock. The contribution to kappa from the half-space is taken as 0.006 seconds (s), consistent with the GMM. Both the hysteretic intrinsic damping and the scattering damping within the near-surface profile and apart from the crust are assumed frequency independent.

Based on review of available geotechnical data three base-case profiles were developed. The specified V_S profiles were taken as the mean or BE base-case profile (P1) with LR and UR base-case profiles P2 and P3, respectively. Consistent with the guidance from EPRI (Reference 2), the UR base-case profile is constrained to not exceed V_S of 9,200 ft/s. The BE profile is given a weight of 0.4 while the LR and UR profiles are each given a weight of 0.3. This is consistent with the guidance from Reference 2 where the weights are based on a 3-point approximation for a normal distribution reflecting the 10th and 90th percentile.

All three base-case profiles extend to a depth of 4,435 ft below the base of the ground surface at the BVPS Site. This depth is taken as the boundary where hard-rock site conditions exist. The basis for this selection considered guidance from Reference 2 which indicates that a sufficient depth should be selected such that hard-rock V_S is reached or the depth is greater than the criteria for no influence on response for spectral frequencies greater than 0.5 Hz. The base-case profiles

(P1, P2, and P3) are shown on ***Figure 3-3*** and listed in ***Table 3-3***, and represent the V_S profiles used for the site response analysis for all structures except the INTS.

To account for random variations in V_S beneath structure footprints, 30 randomized V_S profiles are generated utilizing the stochastic model developed from Reference 78. The range of Vs values for each of the geologic layers was reviewed to ensure that the Vs values modeled are realistic for the types of soils and Paleozoic rocks at the BVPS Site.

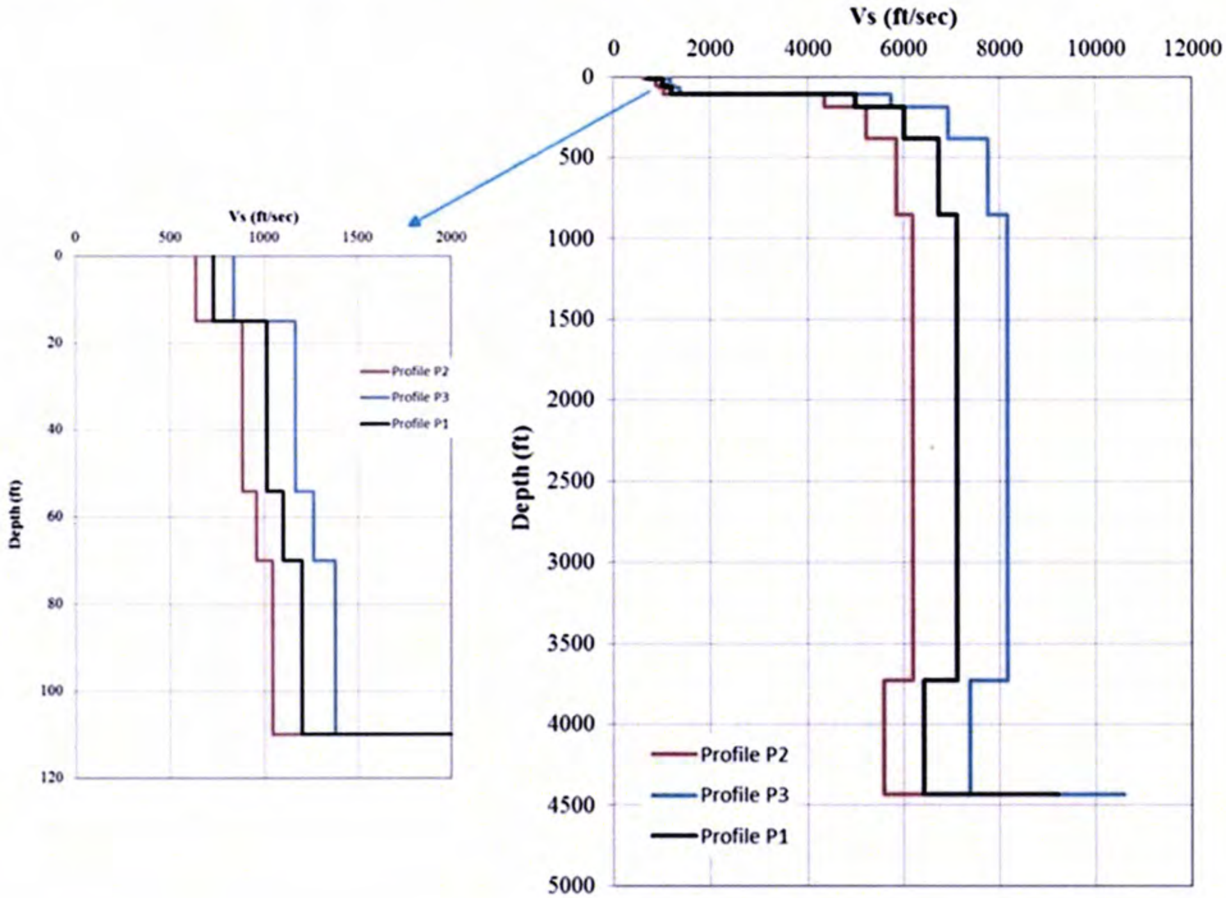


FIGURE 3-3
BASE-CASE V_S PROFILES, BVPS SITE

Consistent with the guidance from Reference 2, uncertainty and variability in material dynamic properties are included in the site response analysis. The soils at the BVPS are generally represented as sand and gravel so both the EPRI soil and Peninsular Range curves from Reference 2 are appropriate. Consideration was also given to the use of dynamic property curves developed for the proposed Bell Bend Nuclear Power Plant (NPP) site on the Susquehanna River in east-central Pennsylvania, which are also appropriate for sand and gravel; the Bell Bend curves are similar to the EPRI soil curves but allow for more significant non-linear site response as represented by higher shear modulus reduction. In summary, the Bell Bend dynamic property curves are associated with the most non-linear behavior (as expressed by the shear modulus reduction versus shear strain curves) while the Peninsular Range dynamic property curves are associated with the least non-linear behavior. Given this observation the selection of the soil dynamic property curves was directly linked to the stiffness (V_S) of each soil profile.

TABLE 3-3
BASE-CASE V_S PROFILES, BVPS SITE

Layer Elevation (ft)	Profile P1		Profile P2		Profile P3	
	Vs (ft/s)	Depth (ft)	Vs (ft/s)	Depth (ft)	Vs (ft/s)	Depth (ft)
735	730	0	635	0	840	0
720	730	15	635	15	840	15
720	1,015	15	883	15	1,167	15
681	1,015	54	883	54	1,167	54
681	1,100	54	957	54	1,265	54
665	1,100	70	957	70	1,265	70
665	1,200	70	1,043	70	1,380	70
625	1,200	110	1,043	110	1,380	110
625	5,000	110	4,348	110	5,750	110
550	5,000	185	4,348	185	5,750	185
550	6,026	185	5,240	185	6,930	185
350	6,026	385	5,240	385	6,930	385
350	6,744	385	5,864	385	7,756	385
300	6,744	435	5,864	435	7,756	435
300	6,744	435	5,864	435	7,756	435
-120	6,744	855	5,864	855	7,756	855
-120	7,112	855	6,184	855	8,179	855
-2994	7,112	3,729	6,184	3,729	8,179	3,729
-2994	6,416	3,729	5,579	3,729	7,378	3,729
-3700	6,416	4,435	5,579	4,435	7,378	4,435

For the rock material, uncertainty is represented by modeling the material as either linear or non-linear in its dynamic behavior over the top 500 ft of rock. This material primarily consists of shale and sandstone. The use of the EPRI rock curves from Reference 2, which exhibit a relatively high amount of low-strain damping (~3.2 percent), is limited to the upper 100 ft where the rock is considered as weathered and fractured. For the alternative linear analyses, the low-strain damping from the EPRI rock curves was used as the constant value of damping in the upper 100 ft.

Within the depth range of 100 ft to 500 ft, non-linear dynamic behavior is based on the unweathered shale dynamic properties from Reference 75 for the Y-12 Site at Oak Ridge, Tennessee. For these curves the low-strain damping is about 1 percent. For the alternative linear analyses, the low-strain damping from the Reference 75 unweathered shale curves were used as the constant damping value from 100 ft to 500 ft. Below a depth of 500 ft, linear material behavior is adopted, with the damping value specified consistent with the kappa estimate for the Site.

Near-surface site damping is described in terms of the parameter kappa. For the BVPS site, kappa was estimated following the guidance in Reference 2 using the approach for cases where the thickness of the sedimentary rock overlying hard-rock is greater than 3,000 ft. There is confidence, based on deep well sonic log data from the vicinity of the Site, that the hard-rock horizon is more than 4,000 ft below the top of rock. For each Vs profile, kappa was estimated using the equations from Reference 2 for the kappa contribution from the soil and the kappa contribution from the entire bedrock section. The kappa contribution for the Paleozoic rock section is defined as the bedrock kappa minus the kappa contribution from hard-rock (.006s).

The site kappa is used to establish the damping for the Paleozoic rock material below a depth of 500 ft. This is accomplished by using the low-strain damping and the Vs profiles to determine the remaining kappa contribution from the rock layers below a depth of 500 ft within the rock. Given the remaining kappa contribution for the deep rock layers and the Vs for those layers, the damping for these layers can be defined. The site response analysis is then completed assuming linear behavior for these deeper rock layers with appropriate low-strain damping values.

Using the kappa values obtained for the three velocity profiles and including a kappa of 0.006s for the underlying hard-rock the total site kappa is estimated to be 0.0167s for profile P1, 0.0191s for profile P2, and 0.0146s for profile P3. To complete the representation of uncertainty in kappa a 50 percent variation to the base-case kappa estimates was added for profiles P2 and P3. For profile P2, the softest profile, the base-case kappa estimate of 0.0191s was augmented with 50 percent increase in kappa to a value of 0.0286s, resulting in two sets of analyses for profile P2. Similarly uncertainty in kappa for profile P3, the stiffest profile, was augmented with a 50 percent reduction in kappa, resulting in kappa values of 0.0146s and 0.0097s. The suite of kappa estimates and associated weights is listed in ***Table 3-4***.

Consistent with the guidance in Reference 2, input Fourier amplitude spectra were defined for a single representative earthquake magnitude (M 6.5) using two different models for the shape of the seismic source spectrum (single-corner and double-corner).

TABLE 3-4
KAPPA VALUES AND WEIGHTS USED IN SITE RESPONSE ANALYSIS

VELOCITY PROFILE	PROFILE WEIGHT	KAPPA (S)	KAPPA WEIGHT
P1 Base-Case	0.4	0.0167	1.0
P2 Lower Range	0.3	0.0191	0.6
		0.0286	0.4
P3 Upper Range	0.3	0.0146	0.6
		0.0097	0.4

Parallel to the deviation of site response inputs for the power block area, site response inputs were also derived for the shared INTS. Epistemic uncertainty in Vs is modeled using three base-case profiles, the mean or BE base-case profile (P1) with LR and UR base-case profiles P2 and P3, respectively. Uncertainty and variability in material dynamic properties for the Pleistocene Terrace deposits are included in the site response analysis. The kappa for each of the base-case profiles uses the same V_S, layer thickness, and damping for the deeper geologic units, and adds above them the Pleistocene Terrace layers and their respective V_S and thickness values. Also, consistent with the site response analysis for the deeper geologic layers, equivalent-linear and linear damping represents the epistemic uncertainty in dynamic properties.

3.1.1.3 Site Response Analysis Results

The site response analysis uses an equivalent-linear method that is implemented using the Random Vibration Theory (RVT) approach. This approach utilizes a simple, efficient method for computing site-specific amplification functions and is consistent with Reference 24 and Reference 2. The input motion is applied at the top of the half-space as outcrop motion. The free-field peak responses at the top of any sub-layers are solved by using the RVT technique. The nonlinearity of the shear modulus and damping is accounted for by the use of equivalent-linear soil properties and an iterative procedure to obtain values for modulus and damping compatible with the effective shear strains in each layer.

Most major structures at the BVPS Site are founded in the Pleistocene Terrace deposits at foundation elevations of approximately 681 ft for the RCBX and 713 ft for the AXLB, SRVB, and MSCV. There are a few structures founded in the compacted granular structural backfill at approximate foundation elevations 723 ft for the FULB and SFGB and at 735 ft for the DGBX. The site response analysis for the BVPS-1 and BVPS-2 shared INTS has a different site profile than for the other structures. The approximate foundation elevation for the INTS is 637 ft while the top of the full soil column is at EL 675.

The seismic structural analysis will treat all of these structures as surface founded at the foundation levels ignoring the effects of embedment. The approach to developing FIRS for each elevation is based on the guidance provided by Reference 25. Each FIRS is provided as the Truncated Soil Column Response (TSCR). After the strain-compatible soil profiles are developed for the full soil column, the soil layers corresponding to the embedment depth of the structure are removed and a second round of soil column analysis is performed with the truncated soil columns with no further iteration on soil properties. The free surface outcrop motions from the second round truncated soil column analysis correspond to the required TSCR.

The results of the site response analysis consist of AFs that describe the amplification (or de-amplification) of reference hard-rock response spectra (5-percent-damped pseudo-absolute acceleration) as a function of frequency and input reference hard-rock PGA amplitude. AFs are determined for the appropriate control point elevation. Because of uncertainty and variability incorporated in the site response analysis, a distribution of AFs is produced. The AFs are represented by a median (i.e., ln-mean) amplification value and an associated log standard deviation (sigma ln) for each spectral frequency and input rock amplitude. Consistent with Reference 2, median total amplification was constrained to not fall below 0.5 to avoid extreme de-amplification that may reflect limitations of the methodology.

Table 3-5 provides the median site AFs and standard deviation of the logarithm of site AFs ($\sigma_{\ln(AF)}$) for the spectral frequencies of 0.5 Hz, 1 Hz, 2.5 Hz, 5 Hz, 10 Hz, 25 Hz, and 100 Hz (PGA) for BVPS Site EL 681. ***Figure 3-4 and Figure 3-5*** show the median site AFs and $\sigma_{\ln(AF)}$ versus S_A for each of the spectral frequencies. The complete set of site response results can be found in Reference 23.

TABLE 3-5
AMPLIFICATION FUNCTIONS FOR BVPS SITE AT EL 681

100 Hz SA [g]	MEDIAN AF	SIGMA Ln(AF)	25 Hz S_A [g]	MEDIAN AF	SIGMA Ln(AF)	10 Hz S_A [g]	MEDIAN AF	SIGMA Ln(AF)	5 Hz SA [g]	MEDIAN AF	SIGMA Ln(AF)
9.59E-03	2.50E+00	1.10E-01	1.24E-02	2.20E+00	9.59E-02	1.94E-02	2.00E+00	1.59E-01	2.21E-02	3.73E+00	2.54E-01
5.13E-02	2.13E+00	9.74E-02	1.00E-01	1.68E+00	1.60E-01	1.06E-01	1.87E+00	1.92E-01	8.85E-02	3.71E+00	2.44E-01
1.08E-01	1.79E+00	9.79E-02	2.12E-01	1.42E+00	1.73E-01	1.98E-01	1.82E+00	2.04E-01	1.54E-01	3.64E+00	2.36E-01
2.37E-01	1.47E+00	9.58E-02	4.45E-01	1.21E+00	1.75E-01	3.81E-01	1.80E+00	2.06E-01	2.84E-01	3.43E+00	2.35E-01
3.73E-01	1.28E+00	8.95E-02	6.77E-01	1.09E+00	1.75E-01	5.58E-01	1.78E+00	2.03E-01	4.09E-01	3.24E+00	2.37E-01
5.15E-01	1.16E+00	8.66E-02	9.13E-01	9.98E-01	1.77E-01	7.37E-01	1.75E+00	1.94E-01	5.35E-01	3.08E+00	2.42E-01
6.61E-01	1.08E+00	8.68E-02	1.15E+00	9.30E-01	1.80E-01	9.17E-01	1.72E+00	1.89E-01	6.61E-01	2.94E+00	2.51E-01
1.03E+00	9.42E-01	9.26E-02	1.75E+00	7.95E-01	1.90E-01	1.37E+00	1.61E+00	1.84E-01	9.74E-01	2.60E+00	2.72E-01
1.42E+00	8.48E-01	9.78E-02	2.38E+00	6.98E-01	2.02E-01	1.84E+00	1.48E+00	1.96E-01	1.30E+00	2.35E+00	2.83E-01
1.83E+00	7.81E-01	1.05E-01	3.04E+00	6.22E-01	2.08E-01	2.33E+00	1.35E+00	2.10E-01	1.64E+00	2.18E+00	2.87E-01
2.23E+00	7.33E-01	1.20E-01	3.66E+00	5.66E-01	2.15E-01	2.79E+00	1.25E+00	2.31E-01	1.97E+00	2.08E+00	2.95E-01

.5 Hz S_A [g]	MEDIAN AF	SIGMA Ln(AF)	1 Hz SA [g]	MEDIAN AF	SIGMA Ln(AF)	0.5 Hz S_A [g]	MEDIAN AF	SIGMA Ln(AF)	0.1 Hz S_A [g]	MEDIAN AF	SIGMA Ln(AF)
2.03E-02	1.72E+00	2.29E-01	1.39E-02	1.32E+00	1.78E-01	7.89E-03	1.26E+00	6.93E-02	3.56E-04	1.18E+00	9.00E-02
6.46E-02	1.81E+00	2.63E-01	3.53E-02	1.34E+00	1.81E-01	1.75E-02	1.27E+00	7.26E-02	6.64E-04	1.23E+00	9.56E-02
1.07E-01	1.87E+00	2.88E-01	5.56E-02	1.36E+00	1.83E-01	2.66E-02	1.27E+00	7.39E-02	1.01E-03	1.25E+00	9.78E-02
3.52E-01	2.16E+00	3.63E-01	1.72E-01	1.41E+00	1.97E-01	7.85E-02	1.29E+00	8.03E-02	3.07E-03	1.28E+00	9.73E-02
4.32E-01	2.25E+00	3.69E-01	2.10E-01	1.42E+00	2.02E-01	9.54E-02	1.30E+00	8.12E-02	3.75E-03	1.28E+00	9.67E-02
6.32E-01	2.48E+00	3.60E-01	3.04E-01	1.46E+00	2.20E-01	1.37E-01	1.30E+00	7.99E-02	5.44E-03	1.29E+00	9.66E-02
8.40E-01	2.69E+00	3.43E-01	4.02E-01	1.49E+00	2.30E-01	1.81E-01	1.31E+00	7.86E-02	7.21E-03	1.30E+00	9.57E-02
1.06E+00	2.84E+00	3.19E-01	5.04E-01	1.52E+00	2.50E-01	2.26E-01	1.32E+00	7.80E-02	9.05E-03	1.31E+00	9.47E-02
1.27E+00	2.92E+00	2.97E-01	6.02E-01	1.55E+00	2.82E-01	2.70E-01	1.32E+00	7.85E-02	1.08E-02	1.31E+00	9.26E-02

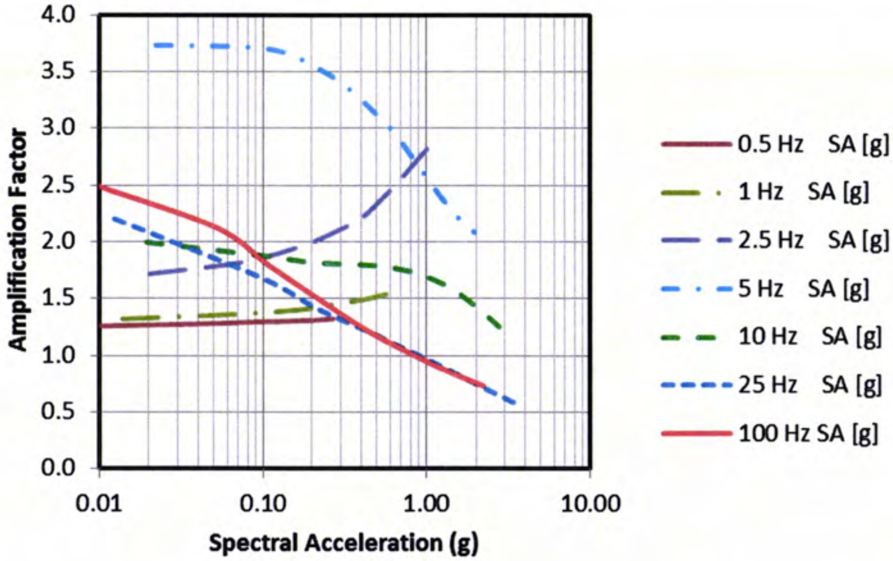


FIGURE 3-4
MEDIAN TOTAL AMPLIFICATION FACTORS VERSUS INPUT HARD-ROCK MOTION FOR BVPS SITE AT EL 681

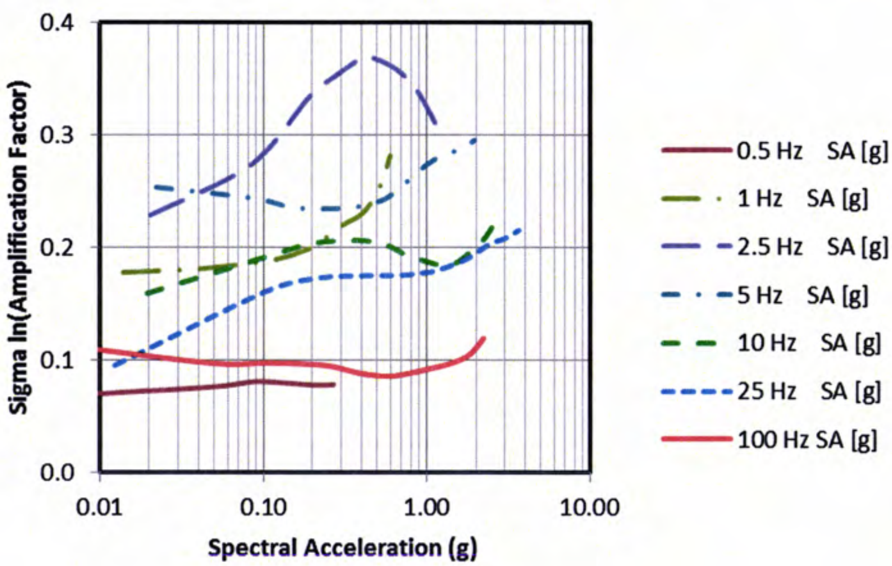


FIGURE 3-5
SIGMA LN (TOTAL AMPLIFICATION FACTORS) VERSUS INPUT HARD-ROCK MOTION FOR BVPS SITE AT EL 681

Given the complexity of the logic trees used to represent epistemic uncertainty in the CEUS-SSC model (Reference 21) and EPRI GMM (Reference 22), the computational demands of propagating all epistemic uncertainty in the site response logic tree into the PSHA is prohibitive. As a result, an assessment was performed to determine how the site response logic tree could be simplified without loss of accuracy in the hazard fractiles at the RCBX foundation elevation (EL 681 ft).

Sensitivity studies were performed to test the approach to grouping on the resulting surface hazard fractiles. Specifically, sensitivity testing assessed the impact of the AF grouping process, in which a portion of the epistemic uncertainty is transferred to aleatory uncertainty. The sensitivity study shows that the AF grouping approach has minimal impact on the mean hazard and on any of the hazard fractiles above the mean for all levels of ground motion. The steps for development of the surface control point hazard curves and hazard fractiles are:

- At each response frequency, group the site AFs according to the patterns observed in the site response logic tree branch of site AFs, as described below.
- Apply the grouped site AFs to all logic tree branches of the CEUS-SSC model and EPRI GMM used to derive the hard-rock hazard.
- Combine the surface hazard branches, using the same combinations as were used to derive the seismic hazard for hard-rock site conditions.

The assessment performed to determine if grouping of AFs was technically justified began with compilation of all AF branches for the seven response frequencies (0.5 Hz, 1.0 Hz, 2.5 Hz, 5 Hz, 10 Hz, 25 Hz, and 100 Hz). For each response frequency, the mean and standard deviation of AFs are saved, consistent with each end-branch of the site response logic tree. Based on the observed pattern in the trend of mean AF over each of the seven response frequencies, three grouped branches are determined for use calculating the control point hazard.

Figure 3-6 and Figure 3-7 display the 20 individual branches of AF from the logic tree together with the recommended three grouped amplification functions for two example response frequencies. On each figure P represents the site profile, M represents the material dynamic properties, K represents kappa, and 1C/2C represent the single-corner and double-corner input motions respectively. The full set of AF grouping results are listed in Reference 23.

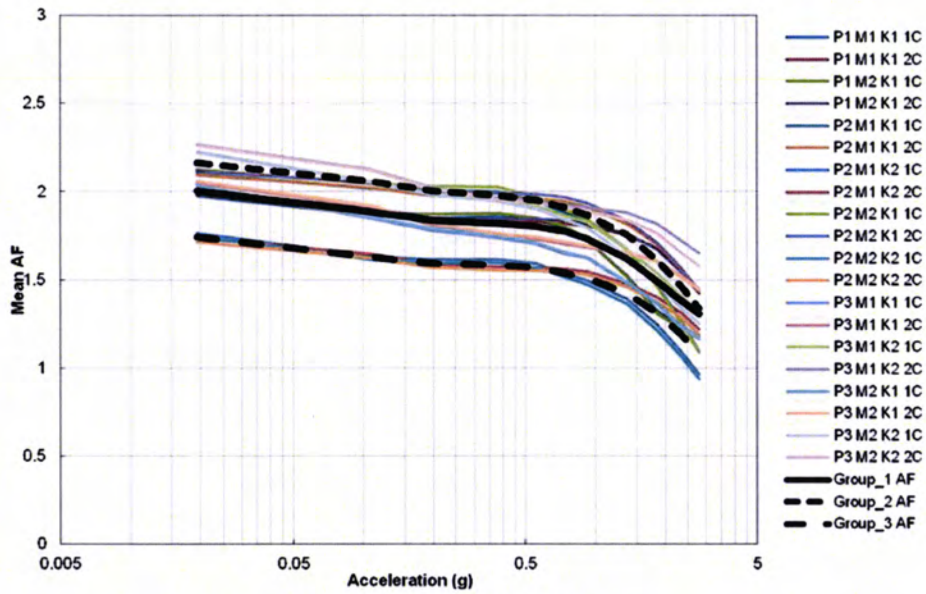


FIGURE 3-6
TOTAL SET OF MEAN SITE AMPLIFICATION FACTORS FOR 10 HERTZ SPECTRAL ACCELERATION TOGETHER WITH THE THREE AMPLIFICATION FACTORS FOR THE SELECTED GROUPINGS

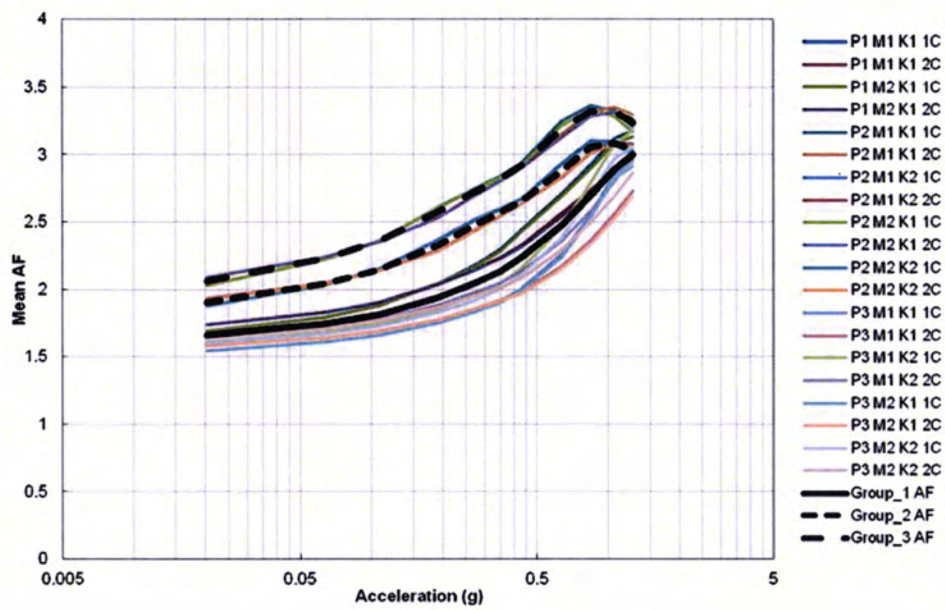


FIGURE 3-7
TOTAL SET OF MEAN SITE AMPLIFICATION FACTORS FOR 2.5 HERTZ SPECTRAL ACCELERATION TOGETHER WITH THE THREE AMPLIFICATION FACTORS FOR THE SELECTED GROUPINGS

3.1.2 Seismic Hazard Analysis Technical Adequacy

The BVPS-1 SPRA hazard methodology and analysis associated with the horizontal GMRS were submitted to the NRC as part of the BVPS-1 Seismic Hazard Submittal (Reference 3), and found to be technically acceptable by NRC for application to the BVPS-1 SPRA.

Subsequent to the March 31, 2014 (Reference 3) submittal, the seismic hazard was updated and FIRS were generated for each of the foundation elevations associated with critical structures as the BVPS for use in the SPRA. ***Figure 3-8*** presents the FIRS at the control point EL 681 ft and compares this to the GMRS reported in the BVPS-1 March 2014 submittal (Reference 3). The difference is attributed to:

- The material damping used for the rock material over the upper 500 ft. While the GMRS, reported in the March 2014, submittal is based on the low-strain damping of 3.2 percent over a 500-foot depth of bedrock, the FIRS used in the BVPS-1 SPRA limits this damping value to the upper 100 ft where the rock is considered as weathered or fractured. Within the depth range of 100 ft to 500 ft, a damping of 1 percent is used based on the unweathered shale dynamic properties from Stokoe et al., (Reference 75). Below a depth of 500 ft, linear material behavior is adopted with the damping value of 0.5 percent is specified consistent with the kappa estimate for the site.
- The subsurface profile used in the site amplification analysis. While the GMRS, reported in the March 2014, submittal is based on a profile which extends from the bottom of the RCBX foundation to at depth hard rock, the FIRS used in the SPRA develops from the analysis of the full soil column to plant grade, subsequently truncated to the RCBX foundation level, in accordance with NRC guidance (Reference 25).

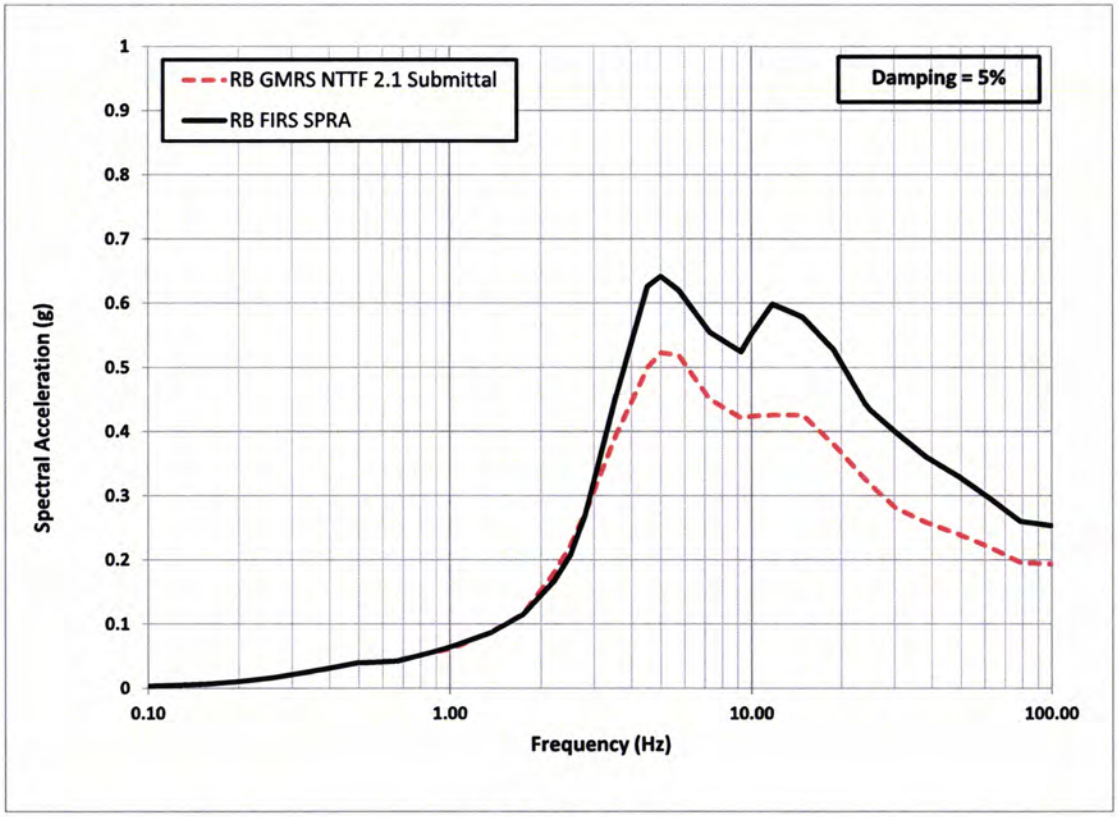


FIGURE 3-8
COMPARISON BETWEEN GMRS AT CONTROL POINT REPORTED IN MARCH 2014 SUBMITTAL AND FIRS USED AS BASIS FOR BUILDING SEISMIC RESPONSE AND FRAGILITY CALCULATION IN BVPS-1 SPRA PROJECT

The BVPS-1 hazard analysis was also subjected to an independent peer review against the pertinent requirements in the PRA Standard (Reference 4). 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

This section provides the final seismic hazard results used in the BVPS-1 SPRA.

The site AFs obtained from the site response analysis and the hard-rock PSHA curves are used to develop the seismic hazard curves and FIRS at the elevations of interest. The procedure to develop the seismic hazard curves follows the methodology described in Reference 2. This procedure, referred to as Approach 3, computes a site-specific control point hazard curve for a broad range of S_A given the site-specific bedrock hazard curve and site-specific estimates of soil or soft-rock response and associated uncertainties. The FIRS represent the performance-based ground motion used as input to the seismic analysis of the buildings.

The above procedure is executed to generate the mean hazard curve and the fractiles at EL 681. ***Figure 3-9*** presents the mean and fractile hazard curves at EL 681 for the spectral frequency of 100 Hz. ***Table 3-6*** presents numerical values of the mean hazard curve and the fractiles of the hazard distribution. The full set of hazard curves at EL 681 can be found in Reference 23.

The PSHA results were used to perform an assessment of the total hazard sensitivity to the epistemic uncertainty in the particular PSHA input variable (i.e., ground motion prediction equation (GMPE), seismicity of distributed sources, maximum magnitude of distributed sources, etc.), which is measured by the variance in the total hazard with contribution solely from the epistemic uncertainty in the specific input variable, normalized by the variance in the total hazard.

The results of this process are shown on ***Figure 3-10*** which displays the variance deaggregation for the spectral frequency of 100 Hz (PGA) at the RCBX control point at EL 681 ft. Deaggregation is shown for MAFE ranging from 1.40E-3 to 2.34E-8. The dominant contributor to the total variance is the epistemic uncertainty in GMPEs. As the MAFE gets lower, the epistemic uncertainty in maximum magnitude, the three magnitude-range cases used for deriving recurrence rates, and the eight recurrence rate realizations become more significant.

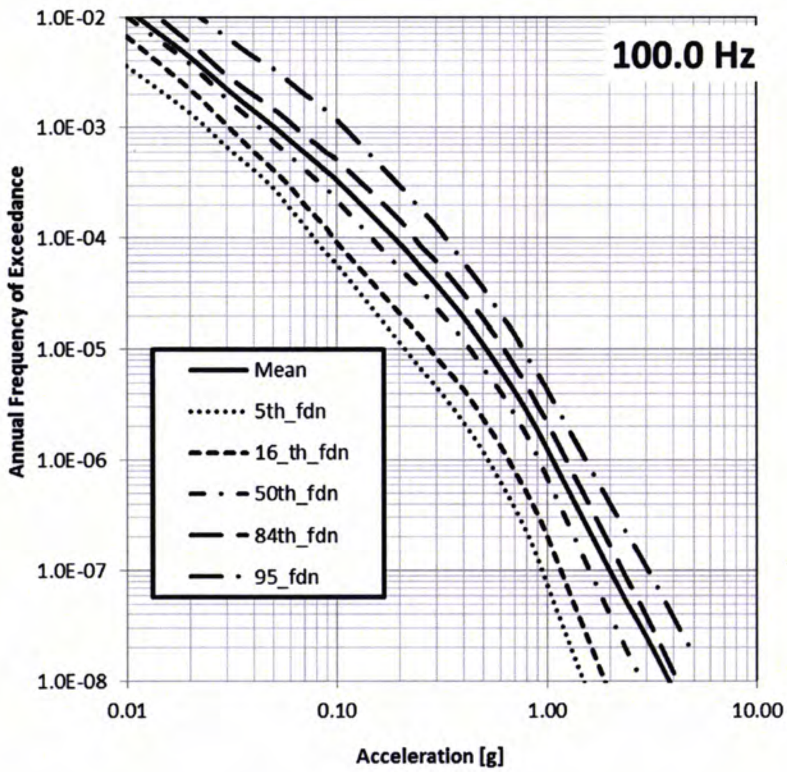


FIGURE 3-9
100-HZ S_A MEAN AND FRACTILE HAZARD CURVES FOR BVPS SITE AT EL 681 BASE OF BVPS-1 AND BVPS-2 REACTOR CONTAINMENT BUILDING FOUNDATION)

Note:

_fnd indicates the seismic hazard at the RCBX foundation level.

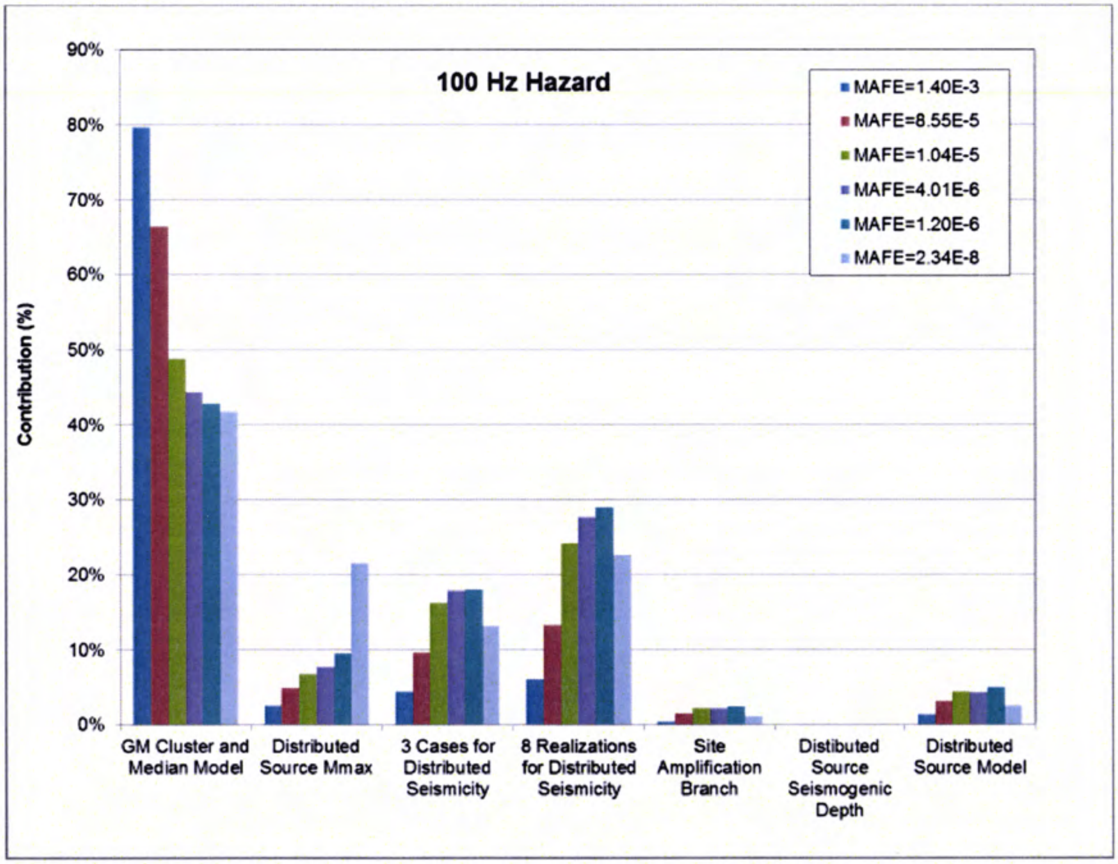


FIGURE 3-10
VARIANCE DEAGGREGATION OF THE BVPS SITE PSHA LOGIC TREE INPUTS FOR THE SPECTRAL FREQUENCY OF 100 HZ

TABLE 3-6
100-HZ S_A MEAN AND FRACTILE HAZARD FOR BVPS SITE AT EL 681 (BASE OF BVPS-1 AND BVPS-2 REACTOR CONTAINMENT BUILDING FOUNDATION)

SPECTRAL ACCELERATION [g]	ANNUAL FREQUENCY OF EXCEEDANCE					
	MEAN	5TH	16TH	50TH	84TH	95TH
0.01	1.19E-02	3.61E-03	6.72E-03	1.02E-02	1.72E-02	2.68E-02
0.02	4.33E-03	1.35E-03	2.22E-03	3.75E-03	5.83E-03	1.16E-02
0.03	2.23E-03	6.67E-04	1.05E-03	1.81E-03	2.96E-03	6.54E-03
0.04	1.44E-03	4.20E-04	6.16E-04	1.14E-03	1.99E-03	4.41E-03
0.05	1.03E-03	2.84E-04	4.16E-04	7.49E-04	1.48E-03	3.42E-03
0.06	7.69E-04	1.85E-04	2.94E-04	5.56E-04	1.12E-03	2.57E-03
0.07	6.02E-04	1.32E-04	2.03E-04	4.27E-04	8.52E-04	2.03E-03
0.08	4.86E-04	9.34E-05	1.55E-04	3.39E-04	7.13E-04	1.70E-03
0.09	4.02E-04	7.20E-05	1.22E-04	2.79E-04	5.91E-04	1.41E-03
0.10	3.38E-04	5.82E-05	9.23E-05	2.22E-04	5.18E-04	1.19E-03
0.20	8.92E-05	1.14E-05	2.08E-05	5.33E-05	1.45E-04	2.98E-04
0.25	5.54E-05	6.72E-06	1.31E-05	3.49E-05	8.60E-05	1.85E-04
0.30	3.77E-05	4.51E-06	8.30E-06	2.31E-05	6.09E-05	1.27E-04
0.40	1.95E-05	2.20E-06	4.38E-06	1.22E-05	3.19E-05	6.13E-05
0.50	1.09E-05	1.17E-06	2.32E-06	6.66E-06	1.86E-05	3.52E-05
0.60	6.58E-06	6.35E-07	1.35E-06	4.16E-06	1.13E-05	2.09E-05
0.70	4.21E-06	3.68E-07	8.02E-07	2.58E-06	7.11E-06	1.38E-05
0.80	2.78E-06	2.20E-07	5.01E-07	1.68E-06	4.75E-06	8.70E-06
0.90	1.87E-06	1.30E-07	3.20E-07	1.08E-06	3.13E-06	6.16E-06
1.00	1.26E-06	7.42E-08	2.04E-07	6.88E-07	2.20E-06	4.32E-06
2.00	9.70E-08	2.20E-09	7.37E-09	3.87E-08	1.64E-07	3.94E-07
3.00	2.44E-08	2.59E-10	1.11E-09	7.66E-09	3.75E-08	1.05E-07
5.00	3.65E-09	1.26E-11	6.77E-11	7.24E-10	4.89E-09	1.71E-08
6.00	1.74E-09	3.89E-12	2.28E-11	2.79E-10	2.21E-09	8.23E-09
7.00	9.40E-10	1.36E-12	8.22E-12	1.27E-10	1.14E-09	4.62E-09
8.00	5.61E-10	5.38E-13	3.93E-12	6.27E-11	6.25E-10	2.70E-09
9.00	3.41E-10	2.31E-13	1.77E-12	3.30E-11	3.63E-10	1.76E-09
10.00	2.19E-10	1.05E-13	8.24E-13	1.77E-11	2.19E-10	1.07E-09

Following the guidelines in Reference 24 the FIRS for the control point of interest are developed following a performance-based approach. The foundation level seismic hazard curves and UHRS provide the input to derive the performance-based FIRS. The performance-based FIRS are developed by scaling the mean 1E-4 MAFE UHRS by a design factor that is related to the ratio of the 1E-5 MAFE S_A to the corresponding 1E-4 MAFE S_A (Reference 24).

Figure 3-11 presents the performance-based horizontal FIRS at EL 681, and the 1E-4 and 1E-5 UHRS. ***Table 3-7*** presents numerical values of the S_A for the FIRS at EL 681. The horizontal FIRS at all other foundation elevations are presented in Reference 23.

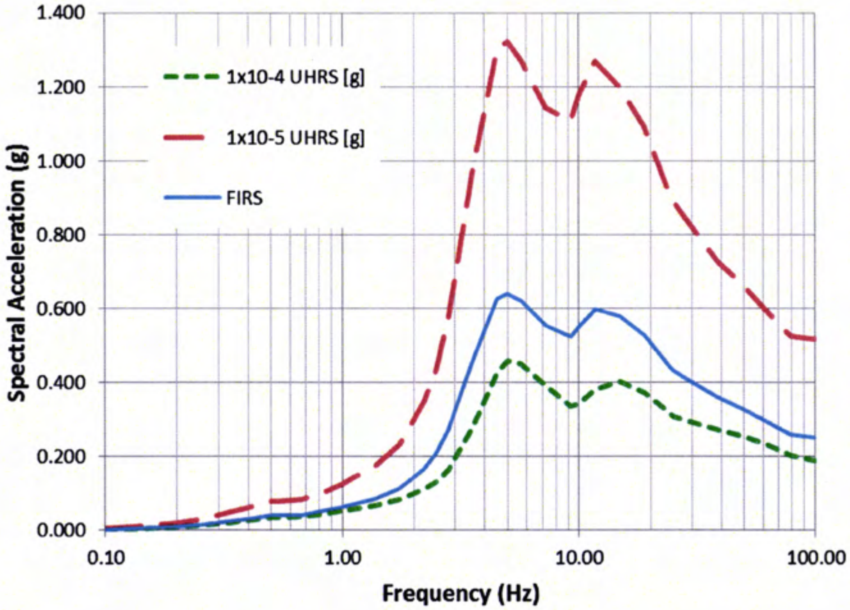


FIGURE 3-11
UHRS AND FIRS AT THE BVPS SITE AT EL 681

TABLE 3-7
UHRS AND FIRS AT THE BVPS SITE AT EL 681

Frequency (Hz)	Horizontal Spectral Acceleration (g) at the Foundation Elevation		
	$1x10^{-4}$ MAFE UHRS	$1x10^{-5}$ MAFE UHRS	FIRS
0.10	0.0028	0.0067	0.0034
0.13	0.0040	0.0097	0.0048
0.16	0.0058	0.0141	0.0071
0.20	0.0088	0.0211	0.0106
0.26	0.0136	0.0321	0.0162
0.33	0.0205	0.0474	0.0240
0.42	0.0285	0.0642	0.0328
0.50	0.0353	0.0780	0.0399
0.53	0.0352	0.0783	0.0400
0.67	0.0366	0.0831	0.0423
0.85	0.0459	0.1077	0.0545
1.00	0.0534	0.1264	0.0639
1.08	0.0573	0.1384	0.0696
1.37	0.0673	0.1756	0.0870
1.74	0.0829	0.2351	0.1145
2.21	0.1115	0.3495	0.1669
2.50	0.1310	0.4378	0.2064
2.81	0.1654	0.5799	0.2707
3.56	0.2802	0.9789	0.4573
4.52	0.4272	1.3039	0.6258
5.00	0.4574	1.3216	0.6413
5.74	0.4510	1.2712	0.6200
7.28	0.3927	1.1447	0.5545
9.24	0.3372	1.1085	0.5242
10.00	0.3429	1.1766	0.5517
11.72	0.3798	1.2689	0.5981
14.87	0.4039	1.1986	0.5785
18.87	0.3727	1.0896	0.5275
23.95	0.3193	0.9138	0.4443
25.00	0.3092	0.8922	0.4331
30.39	0.2919	0.8157	0.3985
38.57	0.2724	0.7286	0.3591
48.94	0.2575	0.6661	0.3305
62.10	0.2333	0.5956	0.2963

TABLE 3-7
UHRS AND FIRS AT THE BVPS SITE AT EL 681
(CONTINUED)

FREQUENCY (Hz)	HORIZONTAL SPECTRAL ACCELERATION (g) AT THE FOUNDATION ELEVATION		
	1x10^{-4} MAFE UHRS	1x10^{-5} MAFE UHRS	FIRS
78.80	0.2026	0.5244	0.2601
100.00	0.1885	0.5158	0.2530

Note:

MAFE = mean annual frequency of exceedance.

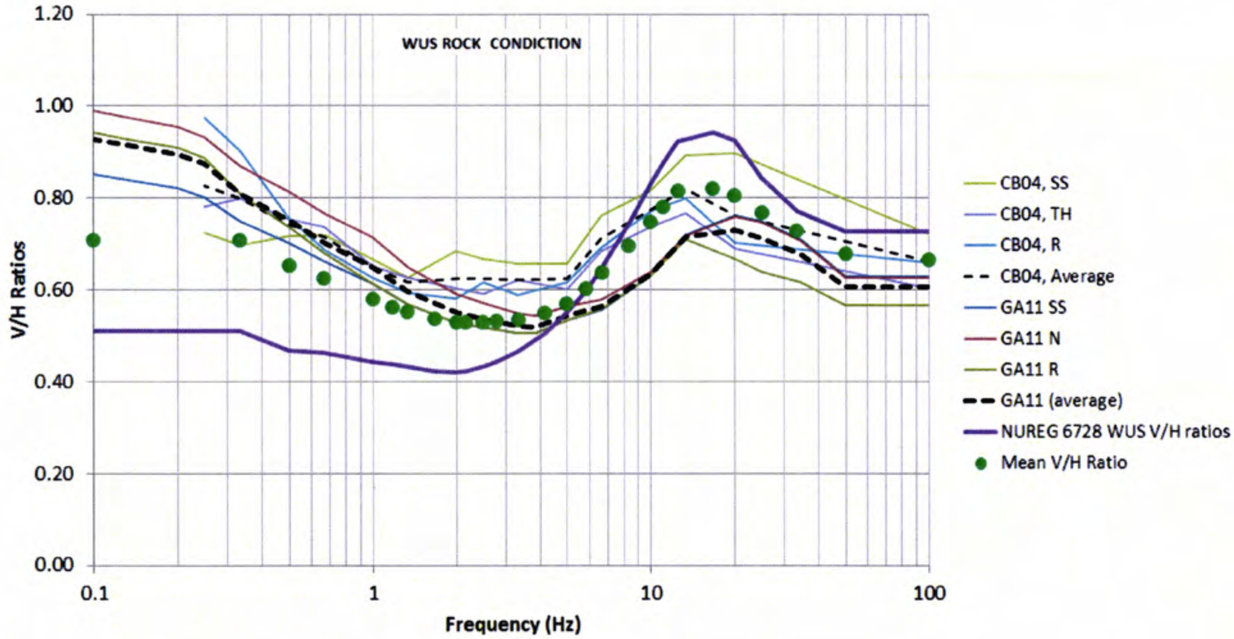
3.1.4 Horizontal and Vertical FIRS

This section provides the control point horizontal and vertical FIRS.

Vertical response spectra are developed at each foundation elevation by combining the appropriate horizontal response spectra and vertical-to-horizontal (V/H) response spectral ratios. The V/H response spectral ratios consider guidance provided in Reference 77 and Reference 79, which both provide approaches applicable to a range of CEUS or WUS sites.

For the BVPS Site three factors influence the approach used to derive V/H ratios: (1) the kappa values estimated for the site are significantly larger than the hard-rock kappa value of 0.006s reported for CEUS hard-rock sites in Reference 77, (2) the site-specific V_{S30} values for the site profiles are best associated with intermediate or soft sites as reflected in Reference 79, and (3) the shape of the horizontal FIRS at each of the foundation elevations peak at spectral frequencies closer to WUS spectral shapes. Given these factors the approach used to derive V/H ratios for the BVPS Site considers the generic V/H ratios from Reference 77 and the empirical GMPEs as described in Reference 79. For each foundation elevation a mean V/H ratio is derived by considering equal weights for WUS and CEUS rock site conditions, and equal weights on the V/H values derived by applying the GMPEs Reference 80 and Reference 81 and the generic V/H values from Reference 77.

The calculated V/H ratio for the RCBX foundation elevation is shown on ***Figure 3-12*** which displays the results separately for WUS rock conditions and CEUS rock conditions, showing the range of values for the models considered and the overall median V/H ratio from this range. On this figure the bottom plot displays the overall mean V/H ratio for WUS and CEUS rock conditions (from the top two figures) and the recommended V/H ratio based on averaging the mean V/H ratio for WUS and CEUS rock conditions. The vertical FIRS are derived using the V/H ratios and the horizontal FIRS. ***Figure 3-13*** shows the horizontal and vertical FIRS at the RCBX foundation elevation. The horizontal FIRS, the applicable V/H ratios, and the vertical FIRS for the RCBX foundation elevation are displayed on ***Table 3-8***. The full set of V/H ratios and vertical FIRS at other foundation elevations can be found in Reference 23.



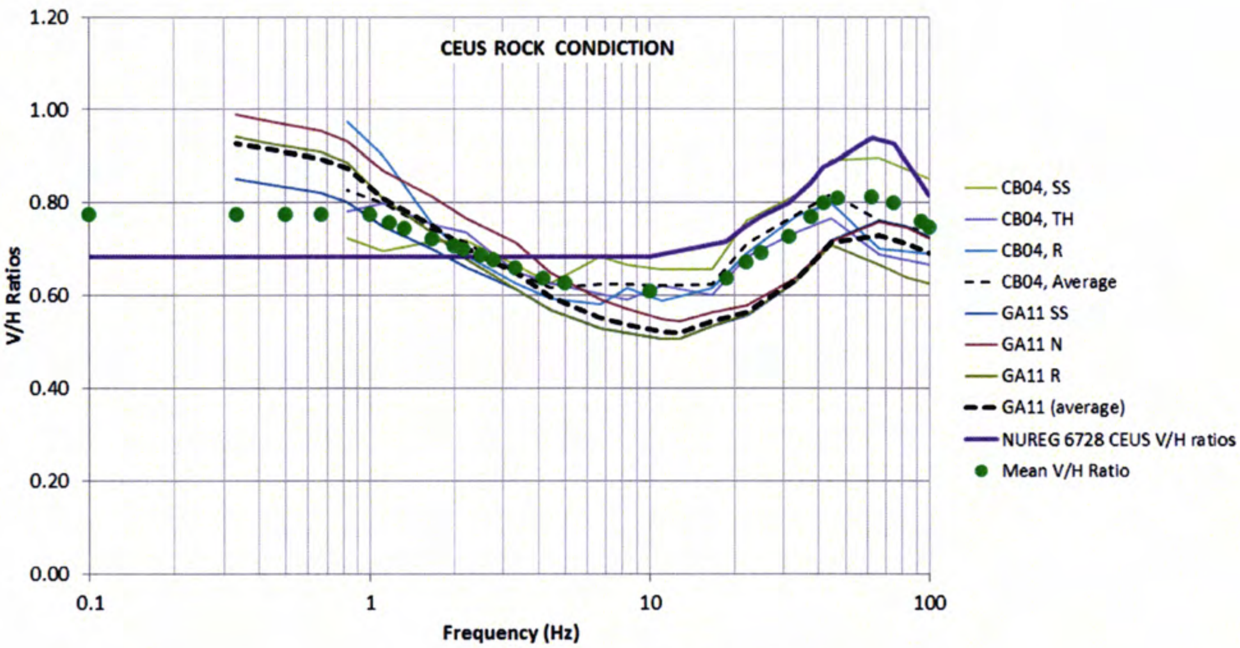


FIGURE 3-12
VERTICAL-TO-HORIZONTAL RATIOS FROM DIFFERENT MODELS AND THE RECOMMENDED MEDIAN VERTICAL-TO-HORIZONTAL RATIO FOR BVPS SITE EL 681

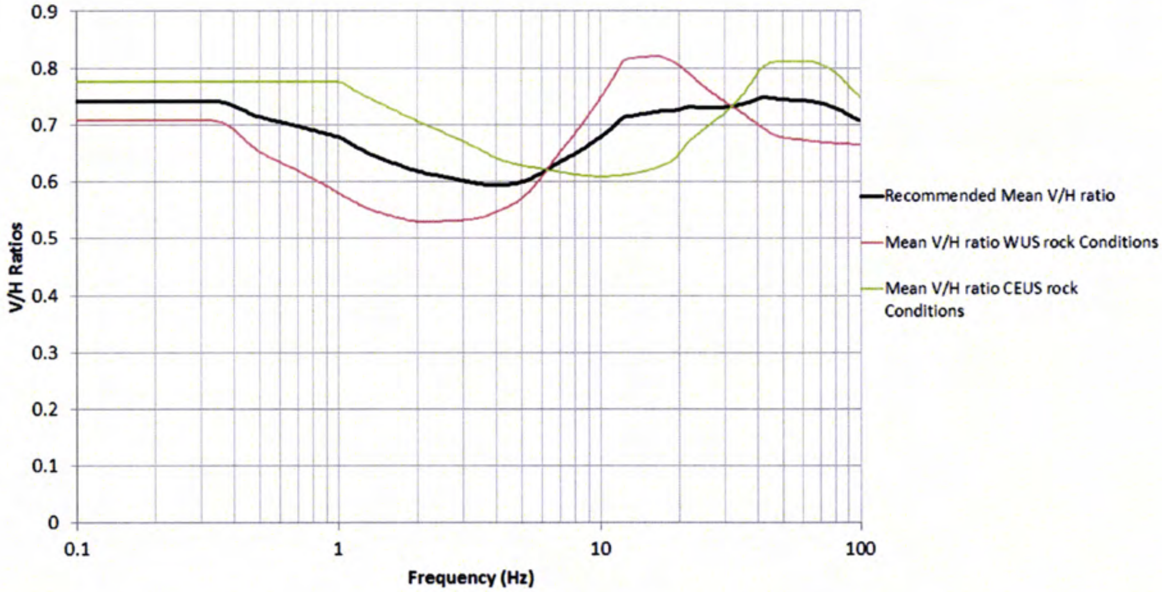


FIGURE 3-12
(CONTINUED)
VERTICAL-TO-HORIZONTAL RATIOS FROM DIFFERENT MODELS AND THE RECOMMENDED MEDIAN VERTICAL-TO-HORIZONTAL RATIO FOR BVPS SITE EL 681

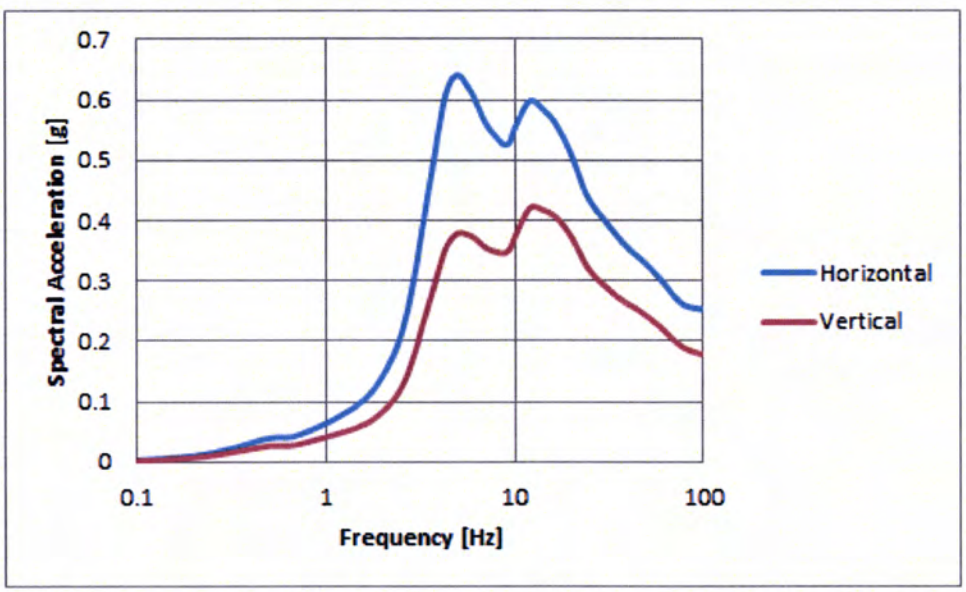


FIGURE 3-13
HORIZONTAL AND VERTICAL FOUNDATION INPUT RESPONSE SPECTRA AT THE BVPS SITE AT FOUNDATION EL 681

TABLE 3-8
HORIZONTAL AND VERTICAL FIRS AT THE BVPS SITE AT EL 681

FREQUENCY (Hz)	HORIZONTAL FIRS (g)	V/H RATIO	VERTICAL FIRS (g)
0.100	0.0034	0.7045	0.0024
0.200	0.0106	0.7045	0.0074
0.331	0.0242	0.7045	0.0170
0.501	0.0399	0.6754	0.0270
0.676	0.0425	0.6577	0.0280
1.000	0.0639	0.6480	0.0414
1.202	0.0770	0.6270	0.0483
1.413	0.0898	0.6108	0.0549
1.622	0.1047	0.6003	0.0629
1.820	0.1220	0.5919	0.0722
2.042	0.1464	0.5843	0.0855
2.188	0.1641	0.5812	0.0954
2.399	0.1911	0.5803	0.1109
2.630	0.2310	0.5806	0.1341
2.818	0.2726	0.5813	0.1585
3.020	0.3220	0.5820	0.1874
3.311	0.3987	0.5829	0.2324
3.631	0.4723	0.5821	0.2749
3.981	0.5484	0.5800	0.3181
4.266	0.5999	0.5811	0.3486
4.571	0.6284	0.5839	0.3669
4.786	0.6375	0.5867	0.3740
5.012	0.6413	0.5906	0.3787
5.248	0.6376	0.5954	0.3797
5.495	0.6291	0.6009	0.3780
5.754	0.6195	0.6067	0.3758
6.026	0.6087	0.6126	0.3729
6.457	0.5879	0.6218	0.3655
6.918	0.5663	0.6300	0.3568
7.413	0.5514	0.6376	0.3516
7.763	0.5436	0.6425	0.3492
7.943	0.5398	0.6450	0.3481
8.511	0.5297	0.6529	0.3459
8.913	0.5254	0.6584	0.3460
9.550	0.5318	0.6669	0.3547
10.000	0.5517	0.6730	0.3713

TABLE 3-8
HORIZONTAL AND VERTICAL FIRS AT THE BVPS SITE AT EL 681 (CONTINUED)

FREQUENCY (Hz)	HORIZONTAL FIRS (g)	V/H RATIO	VERTICAL FIRS (g)
12.023	0.5979	0.7030	0.4203
14.125	0.5849	0.7138	0.4175
16.218	0.5645	0.7226	0.4079
18.197	0.5372	0.7285	0.3914
20.417	0.5025	0.7325	0.3681
22.387	0.4676	0.7345	0.3435
23.988	0.4438	0.7328	0.3253
26.303	0.4229	0.7311	0.3092
28.184	0.4109	0.7298	0.2998
30.200	0.3995	0.7290	0.2913
34.674	0.3758	0.7317	0.2750
39.811	0.3549	0.7392	0.2623
44.668	0.3413	0.7411	0.2530
50.119	0.3273	0.7383	0.2417
54.954	0.3144	0.7375	0.2318
60.256	0.3007	0.7366	0.2215
70.795	0.2736	0.7324	0.2004
81.283	0.2587	0.7236	0.1872
100.000	0.2530	0.7017	0.1776

Dynamic properties of soil are degraded due to their non-linear response under a controlling earthquake motion propagated through the soil profile. This degradation is represented by strain-compatible dynamic properties obtained from the output of an equivalent-linear site response analysis. Epistemic and aleatory uncertainty of the input motion, V_S, thickness, damping etc., is included in the site amplification analysis. Three deterministic soil profiles that represent uncertainty in V_S, V_p, damping, and thickness are provided. The approach is consistent with Reference 82 and Reference 2.

A fully probabilistic approach is employed to develop the strain-compatible dynamic properties that preserve consistency with the ground motion hazard. Assuming the strain-compatible properties are lognormally distributed, this approach is analogous to Approach 3 described in Reference 77. The mean and standard deviation of logarithmic (ln*e*) strain-compatible properties are determined as a function of rock S_A for each soil layer in the same manner that a mean and standard deviation of logarithmic site AFs is determined. The soil S_A is determined from the soil hazard curve at the MAFE of interest, and the corresponding AFs and associated strain-compatible properties at the soil S_A are used.

Reference 2 considers the variation of the strain-compatible property for different response frequencies of the FIRS. The FIRS is not a response spectrum associated with a single earthquake, so the main contributor at a spectral frequency of 1.0 Hz could produce strains in the soil column different from those produced by the main contributor at a spectral frequency of 100 Hz (assumed to be PGA). To address this, Reference 2 states: "To examine consistency in strain-compatible properties across structural frequency, the entire process is performed at PGA (typically 100 Hz), and again at low frequency, typically 1 Hz. If the differences in properties at high- and low frequency are less than 10%, the high-frequency properties may be used since this frequency range typically has the greatest impact on soil nonlinearity. If the difference exceeds 10% [hazard-consistent strain-compatible properties] the hazard-consistent strain-compatible properties (HCSCP) developed at PGA and those developed at 1 Hz may be combined with equal weights."

To implement this requirement, two set of strain-compatible properties are obtained; one for a spectral frequency of 1 Hz and the other for 100 Hz S_A (PGA). If the differences between the means or standard deviations for the two spectral frequencies are larger than 10 percent, then the approach described above is used.

Once the BE strain-compatible shear modulus (G) and shear-wave damping (S) profiles and their standard deviations are determined, the upper- and lower-bound profiles are determined following Reference 82. The minimum requirement for coefficient of variation (COV) for site material in NRC, (2013) is 0.5 for well-investigated and 1.0 otherwise.

The resulting set of strain-compatible properties for the BVPS Site is provided in Reference 23.

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 BVPS-1 SPRA. The subsections provide brief summaries of these elements.

4.1 SEISMIC EQUIPMENT LIST

For the BVPS-1 SPRA, a seismic equipment list (SEL) was developed that includes those SSCs that are important to achieving safe shutdown following a seismic event, and to mitigating radioactivity release if core damage occurs, and that are included in the SPRA model. The methodology used to develop the SEL is generally consistent with the guidance provided in EPRI 3002000709 (Reference 15).

4.1.1 SEL Development

The BVPS-1 SEL was developed as follows:

Potential seismic-induced initiating events and consequential events were identified based on the internal events PRA and review of other potential seismic initiators. The following is a summary of items considered in developing the SEL.

The creation of the BVPS-1 SPRA SEL started with the SSCs listed in the existing BVPS-1 PRA, Internal Events Model. It further considered the list of SSCs developed much earlier for the BVPS-1 individual plant examination of external events (IPEEE [Reference 9]).

The following bases were used in the development of the BVPS-1 SPRA SEL:

1. The existing Internal Events PRA for BVPS-1 meets the Capability Category II requirements of the American Society of Mechanical Engineers (ASME) PRA Standard for PRA applications and complies with Regulatory Guide 1.200, Revision 1 (Reference 16).
2. The internal events PRA model used is as of July 2014; i.e., BV1REV6F. This is a working model update from BV1REV5a that was formally documented in Reference 26.
3. SSCs located in the turbine building are included in the SPRA SEL, although most are not credited in the SPRA sequence models. While the turbine building has some seismic capacity, it also contains numerous non-seismic SSCs that may fail in ways that fail other SSCs within the building and prevent operator access to the turbine building. Only the cross-tie cables and the portable generators used for steam generator level indication are located in the turbine building and currently credited. Future SPRA evaluations may choose to credit the turbine building at low seismic accelerations for all SSCs located there. Non-seismic electrical equipment which brings offsite power to the essential buses, are not located in the turbine building.

4.1.1.1 Use of the Internal Events PRA and IPEEE Lists of SSCs

The EPRI guidance document (Reference 2) says that using the previously developed IPEEE SEL as a starting point for listing the SSCs is acceptable. The ASME combined standard (Reference 4) says to use the existing internal events PRA model as the basis for building the seismic PRA logic model. The ASME Standard implies that the SSCs represented in the PRA logic model basic events would make up a starting point for such an SEL. As the IPEEE SEL includes some SSCs originally judged important for seismic risk, but that are not normally found in a PRA logic model for internal events, it was decided to combine the two lists of SSCs as a starting point for the development of the SPRA; i.e., the original IPEEE SEL and the SSCs from the current internal events PRA. This initial combined list does not mean that all SSCs listed in the IPEEE or PRA SEL lists will be explicitly represented in the seismic PRA. Rather, it means that they will be included for consideration during the seismic walkdown and their impact on plant response in an earthquake will then be considered.

For BVPS-1, the internal events PRA logic model (Reference 26) is well established, having evolved since the original individual plant examination in the early 1990s. For example, in parallel to this effort to construct an SPRA, the BVPS-1 PRA was also revised to update the PRA logic models for internal events, internal flooding, and for internal fire initiating events. The effort from these updates is considered in so far as they may impact the SPRA; e.g., especially in the identification of electrical cabinets and panels whose failures could impact the plant response in an earthquake and the listing of potential flood sources.

The internal initiating events were also reviewed for applicability to seismic sequences. ***Table 4-1*** presents all of the initiating events and how they are treated in the seismic PRA

TABLE 4-1
REVIEW OF INTERNAL INITIATING EVENTS FOR APPLICABILITY TO SEISMIC SEQUENCES

INITIATING EVENT CATEGORIES	MODELING OF INITIATOR FOR SPRA
1. Excessive LOCA (reactor vessel failure, not coolable by ECCS)	Reactor vessel included as EQ06, part of Top Event ZL1
2. Large LOCA (> 5" UP TO DBA) BVPS-1 per Rx Crit Yr	Screened out on high seismic capacity
3. Medium LOCA (1.5" TO 5") BVPS-1 per Rx Crit Yr	MLOCA assigned fragility curve; seismic failure leads to direct core damage via failure of Top Event ZL1
4. Small LOCA, Nonisolable (½ to 2-inch diameter)	Fail Top Event PR and assume CIA and CIB conditions
5. Small LOCA, Isolable (PORV train leakage) (0.5" to 1.5")	Screen out, not a seismic failure mode
6. Interfacing Systems LOCA	Screen out on high seismic capacity
7. Steam Generator Tube Rupture	Screen out, not a seismic failure mode
8. Reactor Trip	Assuming plant trip for every seismic initiator
9. Turbine Trip	Assuming plant trip for every seismic initiator
10. Loss of Condenser Vacuum	Assuming condensate lost for all seismic events; and that there is a resulting pressurizer PORV challenge
11. Closure of All Main Steam Isolation Valves (MSIV)	MSIVs not always required to close but likely will due to loss of station air
12. Steam Line Break Upstream of MSIVs	
a. Steam Line Break Inside Containment	Screened on high seismic capacity
b. Main Steam Relief or Safety Valve Opening	Valves modeled for seismic failure to open
c. Steam Line Break in Common Residual Heat Removal System (RHS) Valve Line	Screened on high seismic capacity
13. Steam Line Break Downstream of MSIVs (Outside Containment)	Turbine building collapse is assumed to shear the steamlines; fragility curves for MSIVs are assigned to top event ZMS, and failure of this top event would then fail top event MS and result in loss of steam supply to the TDAFW pump. To satisfy seismic PRA peer review F&O #7-1, this is accounted for in the GENTRANS tree STEAM macro, which includes ZTX=F*MS=F logic
14. Inadvertent Safety Injection	Screened on high seismic capacity
15. Miscellaneous Transients	
a. Total Main Feedwater Loss or Condensate	Assumed for all seismic events
b. Partial Main Feedwater Loss (one loop)	Bounded by total loss of MFW
c. Excessive Feedwater	Not possible since MFW failed for all seismic events
d. Closure of One Main Steam Isolation Valve	Model only seismic failure to close; valves do fail closed on loss of station air
e. Core Power Excursion	Reactor trip always assumed required; Pressurizer PORV assumed challenged anyway
f. Total Loss of Primary Flow (one or more loops)	Pressurizer spray lost anyway due to assumed loss of containment air
g. Main Feedwater Line Break	MFW and dedicated feedpump assumed failed anyway; pressurizer PORV assumed challenged
16. Loss of Offsite Power	Modeled in response to seismic event by acceleration dependent failure probability in ZOG. No credit for recovery of offsite power is given for seismic initiators

TABLE 4-1
REVIEW OF INTERNAL INITIATING EVENTS FOR APPLICABILITY TO SEISMIC SEQUENCES (CONTINUED)

INITIATING EVENT CATEGORIES	MODELING OF INITIATOR FOR SPRA
17. Loss of One 125V DC Emergency Bus	
a. 125V DC Bus 1-1, Orange	Modeled in response to seismic event by acceleration dependent failure probability in ZDC
b. 125V DC Bus 1-2, Purple	Modeled in response to seismic event by acceleration dependent failure probability in ZDC
18. Loss of River Water Headers	
a. Loss of Service Water Header A	Modeled in response to seismic event by acceleration dependent failure probability in ZRW or ZR4
b. Loss of Service Water Header B	Modeled in response to seismic event by acceleration dependent failure probability in ZRW or ZR2
19. Steam Line Break Downstream of MSIVs (Outside Containment)	Turbine building collapse is assumed to shear the steamlines; fragility curves for MSIVs are assigned to top event ZMS, and failure of this top event would then fail top event MS and result in loss of steam supply to the TDAFW pump. To satisfy seismic PRA peer review F&O #7-1, this is accounted for in the GENTRANS tree STEAM macro, which includes ZTX=F*MS=F logic
20. Inadvertent Safety Injection	Screened on high seismic capacity
21. Miscellaneous Transients	
a. Total Main Feedwater Loss or Condensate	Assumed for all seismic events
b. Partial Main Feedwater Loss (one loop)	Bounded by total loss of MFW
c. Excessive Feedwater	Not possible since MFW failed for all seismic events
d. Closure of One Main Steam Isolation Valve	Model only seismic failure to close; valves do fail closed on loss of station air
e. Core Power Excursion	Reactor trip always assumed required; Pressurizer PORV assumed challenged anyway
f. Total Loss of Primary Flow (one or more loops)	Pressurizer spray lost anyway due to assumed loss of containment air
g. Main Feedwater Line Break	MFW and dedicated feedpump assumed failed anyway; pressurizer PORV assumed challenged
22. Loss of Offsite Power	Modeled in response to seismic event by acceleration dependent failure probability in ZOG
23. Loss of One 125V DC Emergency Bus	
a. 125V DC Bus 1-1, Orange	Modeled in response to seismic event by acceleration dependent failure probability in ZDC
b. 125V DC Bus 1-2, Purple	Modeled in response to seismic event by acceleration dependent failure probability in ZDC
24. Loss of River Water Headers	
a. Loss of Service Water Header A	Modeled in response to seismic event by acceleration dependent failure probability in ZRW or ZR4
b. Loss of Service Water Header B	Modeled in response to seismic event by acceleration dependent failure probability in ZRW or ZR2

TABLE 4-1
REVIEW OF INTERNAL INITIATING EVENTS FOR APPLICABILITY TO SEISMIC SEQUENCES
(CONTINUED)

INITIATING EVENT CATEGORIES	MODELING OF INITIATOR FOR SPRA
c. Loss of Both Service Water Headers	Modeled in response to seismic event by acceleration dependent failure probability in ZRW or combination of failure of ZR2 and ZR4
25. Total Loss of Primary Component Cooling Water	Modeled in response to seismic event by acceleration dependent failure probability in ZCC
26. Loss of One Vital Instrument Bus	
a. Loss of Red Vital Bus	Modeled in response to seismic event by acceleration dependent failure probability in ZIO
b. Loss of White Vital Bus	Modeled in response to seismic event by acceleration dependent failure probability in ZIO
c. Loss of Blue Vital Bus	Modeled in response to seismic event by acceleration dependent failure probability in ZIO
d. Loss of Yellow Vital Bus	Modeled in response to seismic event by acceleration dependent failure probability in ZIO
27. Loss of One 4.16-kV Emergency Bus	
a. Loss of 4.16-kV Bus 1AE, Orange	Modeled in response to seismic event by acceleration dependent failure probability in ZAC
b. Loss of 4.16-kV Bus 1DF, Purple	Modeled in response to seismic event by acceleration dependent failure probability in ZAC
28. Loss of a Non-Emergency Bus	
a. Loss of 4.16-kV Bus 1A	Modeled in response to seismic event by acceleration dependent failure probability in ZOG. Since offsite power goes through the normal switchgear to the emergency switchgear a failure of the normal switchgear has the same effect as loss of offsite power
b. Loss of 4.16-kV Bus 1D	Modeled in response to seismic event by acceleration dependent failure probability in ZOG. Since offsite power goes through the normal switchgear to the emergency switchgear a failure of the normal switchgear has the same effect as loss of offsite power
29. Loss of Station Instrument Air	Assumed failed for all seismic events
30. Loss of Containment Instrument Air	Assumed failed for all seismic events
31. Total Loss of Emergency Switchgear Ventilation	Normal ventilation and operator potential action to align portable fans assumed lost due to chilled water pumps and portable fans being located in the turbine building; emergency fans modeled by acceleration dependent failure probability via ZBV

The details for the development of the final SEL can be found in Reference 32. Discussions are provided therein regarding items such as common-cause failure events, Human-Action related basic events and fire and flooding scenarios. Further in 2016, a model update included new basic events to represent the diverse and flexible mitigation strategies (FLEX). The added SSCs were included in the BVPS-1 SEL.

4.1.1.2 Additional SSCs Included in the SEL

Consistent with the ASME Standard (Reference 4), the BVPS-1 IPEEE documentation (Reference 28) and Updated Final Safety Analysis Report (UFSAR) (Reference 29, Table B.1 1) were first reviewed to identify plant structures that should be added to the BVPS-1 SPRA SEL. Such passive SSCs were not included in the internal events PRA models but are of special interest for SPRA. A total of 13 Seismic Category 1 structures were added. Seismic Category 2 and non-seismic structures (also 13 in all) were added if they housed SSCs already on the list. The following structures are included in the BVPS-1 SEL:

- Auxiliary Building (AXLB)
- Reactor Containment Building (RCBX)
- Diesel Generator Building (DGBX)
- Fuel Handling and Decontamination Building (FULB)
- Service Building (SRVB)
- Main Steam and Cable Vault (MSCV)
- Intake Structure (INTS)
- Safeguards Building (SFGB)
- Alternate Intake Structure (AISX)
- Chemical Addition Building (CABX)
- Control Building (CNTB)
- Emergency Response Facility Substation (ERFS)
- Emergency Response Facility (ERFX)
- North Pipe Trench (NPTX)
- Pipe Tunnels (PIPETUNNEL)
- Surrounding Shield Wall for Refueling Water Storage Tank (2QSS-TK21)
- Switchyard Relay House (RLYB)
- ERF Diesel Generator Building (RSGB)
- South Pipe Trench (SPTX)
- Storeroom (STOR)
- Solid Waste Building (SWBX)
- Turbine Building (TRBB)
- River Water Valve Pit Train A&B (VPA_VPB)
- Water Treatment Building (WTBX)
- Primary Plant Demineralized Water Storage Tank Pad and Enclosure (PDWS)

These Category 2 and non-seismic structures were considered further for BVPS-1 only when the fragility analysts determine whether they are likely to survive earthquakes that contribute to risk. The Emergency Response Facility (ERF) Diesel Generator Building fragility was evaluated in the earlier SPRA for BVPS-1 and so, notwithstanding the aforementioned, was retained on the SEL for potential walkdown. Reference 32 outlines other passive SSCs added to the BVPS-1 SEL such as nuclear steam supply system (NSSS) components, block walls, polar crane, and piping segments, among many others. The basis for including these additional passive SSCs is also provided in Reference 32.

In addition to adding passive equipment and structures, alternative lists of SSCs for the SEL were considered. These included SSCs such as those associated with the occurrence of a very small LOCA as well as those associated to LERF. A review of the SEL for both Diablo Canyon and Surry was performed to identify potential additions of non-passive SSCs into the BVPS-1 SEL. The complete list of additional non-passive SSCs is provided in Reference 32 along with their basis for inclusion into the BVPS-1 SEL.

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 BVPS-1 SPRA, in accordance with SPID (Reference 2), Section 6.4.2 and ASME/ANS PRA Standard (Reference 4), Section 5-2.2. The evaluation resulted in most relay chatter scenarios screened from further evaluation based on no impact to component function. One hundred eight relays did not screen based on relay chatter evaluation, however after fragility analysis all 108 relays have high confidence of a low probability of failures (HCLPF) greater than the screening HCLPF for inclusion into the PRA (i.e., they all screen based on seismic capacity). It should be noted that some relays did not screen based on seismic capacity until after the peer review in which the relay fragilities were refined to remove excess conservatisms documented in the peer review report. These relays are still in the PRA model logic, but are no longer among the top contributors to CDF due their increased HCLPF values.

For presentation of results circuit breakers and contactors that did not screen are addressed separately from the above relays. Four circuit breakers did not screen from the model and, therefore, were included in the PRA model for breaker malfunction which was conservatively treated as the trip open and subsequent failure to start of the corresponding pump. These four circuit breakers were for 480V pumps, specifically the A train and B train quench spray pumps and the A train and B train recirculation spray pumps.

Contactors identified through circuit analysis were evaluated through the GERS function during failure mode of the motor control center (MCC) that the contactor is housed in. Four MCC cabinets did not screen from inclusion into the PRA model based on seismic capacity. These four MCC cabinets are BV-MCC-1-E1, BV-MCC-1-E2, BV-MCC-1-E5, and BV-MCC-1-E6. Chatter of the contactors in these MCC cabinets would lead to a failure of river water due to motor-operated valves (MOV) repositioning closed (BV-MCC-1-E1 & BV-MCC-1-E2) or reposition various valves in the auxiliary feedwater or recirculation spray systems (BV-MCC-1-E5 & BV-MCC-1-E6).

The specific SSCs potentially affected by chatter of these relay types and how they are modeled in the PRA are summarized in Section 5.6 of Reference 38.

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

Walkdowns were performed in accordance with guidance in SPID Section 6.5 (Reference 2) and the associated requirements in the PRA Standard (Reference 4).

Several SEL items were previously walked down during the BVPS-1 Seismic IPEEE program. Those walkdown results were reviewed and the following steps were taken to confirm that the previous walkdown conclusions remained valid.

- A walk-by was performed to confirm that the equipment material condition and configuration is consistent with the walkdown conclusions and that no new significant interactions related to block walls or piping attached to tanks exist.
- If the SEL item was screened out based on the previous walkdown, that screening evaluation was reviewed and reconfirmed for the SPRA.

For some SEL SSCs walkdowns had recently been performed in support of resolution of NTTF 2.3 seismic (Reference 14) and the Expedited Seismic Evaluation Process (ESEP) (Reference 8), and information from those walkdowns was used where the appropriate level of detail needed for the SPRA was available.

The seismic walkdowns for equipment outside of the RCBX were performed from February 18 through March 1, 2013. Seismic walkdowns of equipment in RCBX were performed October 9 and 10, 2013, during a station refueling outage. A supplemental walkdown was performed on May 30, 2014, to further evaluate potential seismic-induced fire and seismic-induced flood. A second round of supplemental walkdowns was performed on February 8, 2016, and February 29, 2016, to address F&Os from the December 2014 SPRA peer review, evaluate recently installed FLEX equipment, and assess the lines connected to the spent fuel pool.

The following paragraphs summarize the preparation, procedure, and findings of the seismic walkdowns.

Structures, Systems and Components Walkdown

The BVPS-1 SEL consisting of approximately 2,300 SSCs was reviewed, analyzed, and then reduced to about 900 for walkdown and walk-bys. In addition to selecting representative samples of similar equipment, about 635 check valves and 260 penetrations were excluded as being seismically robust. Approximately 220 SSCs were excluded as being housed within other SSCs that were walked down, and 210 SSCs in the TRBB were excluded since this is a lower-capacity structure. An additional 65 components were excluded from walkdowns since they are not currently modeled in the SPRA. These components generally correspond to non-seismic or Seismic Category II systems. 11 SSCs on the SEL correspond to NSSS components. These items were not walked down, but fragility parameters were developed for them based on available drawings and calculations.

The BVPS-1 SEL also includes items needed to maintain containment (CTMT) functions. The RCBX and equipment that support the CTMT functions, and systems required for CTMT performance (e.g., CTMT fan coolers and CTMT isolation valves) were included in the walkdown list, as well as targeted for fragility analysis.

Table 4-2 presents the number of Walkdown components sorted in accordance with the EPRI Equipment Classes. Equipment Class 1 through Class 21 are assigned consecutively based on the SQUG/Generic Implementation Procedure (GIP) Walkdown Seismic Evaluation Work Sheets (SEWS). Class number (0) is assigned to the remaining components in the Walkdown SEWS as "other" components.

TABLE 4-2
BREAKDOWN OF EQUIPMENT WALKDOWN LIST BY EQUIPMENT CLASS

EPRI CLASS	DESCRIPTION	COUNT
0	Other	121
1	Motor Control Centers	20
2	Low Voltage Switchgear	7
3	Medium Voltage, Metal-Clad Switchgear	8
4	Transformers	14
5	Horizontal Pumps	28
6	Vertical Pumps	13
7	Pneumatic-Operated Valves	128
8A	Motor-Operated Valves	116
8B	Solenoid Valves	11
9	Fans	15
10	Air Handlers	3
11	Chillers	0
12	Air Compressors	0
13	Motor Generators	2
14	Distribution Panels	17

TABLE 4-2
BREAKDOWN OF EQUIPMENT WALKDOWN LIST BY EQUIPMENT CLASS (CONTINUED)

EPRI CLASS	DESCRIPTION	COUNT
15	Battery Racks	9
16	Battery Chargers And Inverters	13
17	Engine Generators	4
18	Instrument (On) Racks	33
19	Temperature Sensors	21
20	Instrument And Control Panels	225
21	Tanks And Heat Exchangers	46
-	Structures and Distribution Systems	44
	SEL Total	905

Walkdown Seismic Review Team

The seismic walkdowns were conducted by two Seismic Review Teams (SRT). Each Team was composed of at least two Seismic Capability Engineers (SCE) along with BVPS-1 Station personnel. All of the key individuals performing the walkdowns completed the 1-week walkdown training sponsored by SQUG. In addition, SCEs possess technical degrees with a structural/seismic background and nuclear-related experience. Furthermore, Mr. Farzin Beigi provided continuous support and expert input to each walkdown team throughout the full extent of the station walkdowns, as well as post-walkdown discussions to ensure consistency between walkdown teams.

Seismic Evaluation Walkdown Procedures

Prior to the walkdown, the SEL comprising the full scope of the seismic evaluations was reviewed by the SRT and Station Personnel. For the purpose of the equipment walkdown, the SEL was divided into mechanical and electrical (M&E) equipment and distribution systems. The locations of structures and components were determined from the station layout drawings. The walkdown sequence, including coordination with station operations, schedule, and route was developed to minimize affecting station operations.

The Walkdown of the SEL items was accomplished in two phases. The first phase was devoted to components that could be examined during normal station operation, while the second phase was planned for the remaining components accessible only during the station outage. Inaccessible components are addressed by inspection of photographs and existing design analysis documents.

Walkdown of Structures

The information required to develop structural fragilities is obtained primarily from design drawings. The seismic walkdown of the structures was limited to verification of the structural location, overall configuration, gross dimensions, and building separation, and any signs of degradation and distress.

Walkdown of Equipment and Distribution Systems

The seismic walkdown of the BVPS-1 M&E equipment was performed in accordance with the methodology of SQUG/GIP and EPRI NP-6041-SL (Reference 7).

The component-specific SQUG/GIP SEWS were utilized to record walkdown observations. Unlike the SQUG/GIP, the focus here was not to perform screening, but rather to document the specific sets of inclusion/exclusion rules or caveats and common bases in accordance with prescribed checklists so that the experience-based HCLPF in EPRI NP-6041-SL (Reference 7) can be supported.

The distribution systems comprising of piping, ducting, and cable trays were walked on a sampling basis, reflecting the industry experience that the distribution systems components generally perform well in a seismic event. The sample set of piping, heating, ventilation, and air-conditioning (HVAC) duct and cable trays segments represent the essential distribution systems in the BVPS-1. In general, the observations related to distribution systems focused on seismic vulnerabilities posed by potential excessive differential motion between structures and poor design of supports and their anchorage.

The walkdown procedures for different types of components are described in detail in the BVPS-1 Walkdown Report (Reference 40).

Additional Walkdown Considerations

In support of the plant walkdown, some added lists were developed for inspection by the walkdown team. Three general areas were considered:

- Operator Action Locations
- Fire Ignition Sources
- Potential Flooding Sources

Human failure events (i.e., models of operator actions) were identified in the BVPS-1 Internal Events PRA. The BVPS-1 SEL Development report (Reference 32) provides a summary of the room locations and path ways needed for recovery action following an earthquake. A total of nine (9) unique locations were identified where credit for operator actions performed outside the control room is taken. These locations were assessed as part of the human reliability analysis to determine which of these locations are likely to be accessible by the operators following a substantial earthquake. A listing of the transit routes for actions performed outside the control room is provided in the human reliability analysis notebook (Reference 36). Verifying that the locations were accessible helped assure that the actions credited in the internal events PRA were still feasible even considering the potential equipment failures that may occur following an earthquake.

Potential fire ignition sources were routinely evaluated by the walkdown team. These sources may coincide with SSCs on the seismic list or be in close proximity to SSCs that are on the list. Only those plant locations evaluated during the walkdown were considered because they contain SSCs on the SEL. However, to provide some assurance that potential sources were not overlooked, the walkdown team performed two informational searches focused on: (1) potential ignition sources involving flammable liquids and piping containing hydrogen or oil, and (2) electrical equipment that could be the source of a seismic-induced fire but are not already on the SEL.

To assist the plant walkdown, a list of potential flooding sources that should be considered during the walk-by was also developed. This list consisted of fire protection system piping, which is maintained "wet" during plant operation and tanks and coolers represented in the initial BVPS-1 Internal Floods PRA. The list of potential flooding sources is presented in the BVPS-1 SEL Development report (Reference 32).

4.2.1 Significant Walkdown Results and Insights

Consistent with the guidance from NP-6041 (Reference 7), no significant findings were noted during the BVPS-1 seismic walkdowns. Note that previous walkdowns for the NTTF Recommendation 2.3 did identify adverse conditions that were documented with their dispositions in a separate submittal (Reference 14).

Components on the SEL were evaluated for seismic anchorage and interaction effects in accordance with SPID guidance (Reference 2) and ASME/ANS PRA Standard (Reference 4) requirements. The walkdowns also assessed the effects of component degradation, such as corrosion and concrete cracking, for consideration in the development of SEL fragilities. In addition, the potential for seismic-induced fire and flooding scenarios was assessed. 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 BVPS-1 SPRA SEL development and walkdowns were subjected to an independent peer review against the pertinent requirements (i.e., the relevant SFR and SPR supporting requirements) in the PRA Standard (Reference 4).

The peer review assessment, and subsequent disposition of peer review findings, is described in Appendix A, and establishes that the BVPS-1 SPRA SEL and seismic walkdowns are suitable for this SPRA application.

4.3 Dynamic Analysis of Structures

This section summarizes the dynamic analyses of structures that contain systems and components important to achieving a safe shutdown, using fixed-base and/or Soil Structure Interaction Analyses (as applicable). The section describes the methodologies used, discusses responses at various locations within the structures and relevant outputs, important assumptions and sources of uncertainty.

4.3.1 Fixed-Base Analyses

No structure at BVPS-1 was analyzed using a fixed based methodology; i.e., SSI was performed for all structures analyzed for the SPRA. Note, however, that fixed-base analyses were performed as verification and validation step in the development of the SSI models, as described in the BVPS-1 Building Seismic Analysis report (Reference 43).

4.3.2 Soil Structure Interaction (SSI) Analyses

The building seismic analysis for BVPS-1 addresses the effects of SSI on the seismic response of the building structures. This analysis accounts for the foundation mat flexibility and its interaction with the flexibility of the supporting geotechnical medium. Both kinematic interaction due to the foundation mat stiffness and inertial interaction due to its mass are accounted for. The seismic incident waves are assumed to propagate vertically in the form of shear waves producing horizontal ground motion and compression waves producing vertical ground motion. Because the solution to the equations of motion is obtained in the frequency domain, the SSI analysis is linear. Strain-compatible soil properties obtained from the site response analysis (Reference 23) are used in the analysis without further modification.

The SSI analysis for BVPS-1 structures utilizes RIZZO's version of the System for Analysis for Soil-Structure-Interaction (SASSI) Program. This version is based on the original SASSI developed in the 1980s at the University of California, Berkeley (Reference 42).

The mean (BE) HCSCP are used in the SSI analyses. Although the site response analysis also develops mean-σ (lower bound) and mean+σ (upper bound) HCSCP, these are not considered in obtaining the seismic response used in the fragility analysis. Rather the effects of the SSI stiffness variation on the seismic demand are incorporated by peak shifting in accordance with the methodology in EPRI 103959 (Reference 11) and EPRI 1019200 (Reference 44). The justification for this approach is discussed in Reference 43 and Calculation 12-4735-F-140 (Reference 45), and summarized as follows:

- For a given input spectrum shape, the deterministic analysis with conservative structure and soil damping and BE structure and soil stiffness results in approximately 80 percent non-exceedance probability response which achieves the targeted demand conservatism for conservative deterministic failure margin (CDFM) evaluations.
- The American Society of Civil Engineers (ASCE) 4-98 (Reference 46) procedure of enveloping of lower-bound (LB) and upper-bound (UB) response and peak shifting provides a conservative design basis response for use in the seismic qualification of multi-mode subsystems. These procedures are conservatively biased and are consequently not used for fragility analysis.
- The LB and UB responses do not represent reasonable median-centered values. The use of LB and UB does not result in a CDFM value representative of 1 percent probability of failure on the composite fragility curve. If LB or UB response is used, then the βc may need to be re-examined so that the conditional failure probabilities are consistently described in quantification.

The ground motion inputs to the building seismic analysis are represented by a set of time histories (two horizontal and one vertical), each matching the appropriate FIRS (hereafter called the FIRS time histories). The FIRS time histories are based on seed (recorded) time histories selected based on similarity of their response spectral shapes to the spectral shapes of the FIRS. The seed time histories are conditioned to obtain FIRS time histories whose response spectra closely match the FIRS. This process implements the guidance in Reference 24 and Reference 82.

Selected records are checked to ensure that they meet criteria established by the NRC regarding the adequacy of time histories. Based on the Reference 82 the strong-motion duration is defined as the time required for the Arias Intensity Reference 83 to rise from 5 to 75 percent (D5-75). The uniformity of the growth of this Arias Intensity is reviewed. The minimum acceptable strong-motion duration should be 6s.

Prior to being used as input to seismic structural analyses, the seed time histories must be conditioned to match the FIRS. Spectral matching analysis is performed to generate spectral-compatible acceleration time histories using the spectral matching computer program, RspMatch09 (Reference 84, 85). RspMatch09 uses a time domain spectral matching method, where adjustment of initial time series (seed motions) is made by adding wavelet functions to the initial acceleration time history in the time domain. This adjustment is repeated until its response spectrum becomes comparable to the target spectrum over the desired frequency range.

Spectral matching analysis is performed by running RspMatch09 multiple times, which is specified in the RspMatch09 input file. The output file from the last run is used to confirm that the adjusted time histories meet the criteria stated in Reference 24 and Reference 82.

To confirm that there is no significant gap in the smoothed power spectral density (PSD) of the matched time histories, the computed PSD are compared to the minimum PSD requirement of Reference 82 which refers to the M and R bins from Reference 77. To comply with Reference 82 the minimum PSD are compared to 80% of minimum PSD in the frequency range of 0.3-24 Hz.

The full suite of time history information can be found in Reference 23.

The time histories described above and used as input to the building seismic analyses match the FIRS presented in Revision 1 of the BVPS PSHA/FIRS Report. They are not modified to match the FIRS presented in ***Section 3.1.4*** from Revision 4 of the BVPS PSHA/FIRS Report on the basis that the shapes of the FIRS utilized in the building seismic analysis reported in Reference 43 are very similar to those of the FIRS presented in ***Section 3.1.4. Figure 4-1*** compares the RCBX horizontal spectra normalized to the RCBX PGA. The comparison illustrates that the difference in the horizontal FIRS is relatively insignificant. However, the comparison on ***Figure 4-2*** shows that the vertical spectra are diminished in excess of 10% in the frequency range of about 8 Hz to 15 Hz because they are now based on the mean of the V/H ratios where previously, the envelope was used.

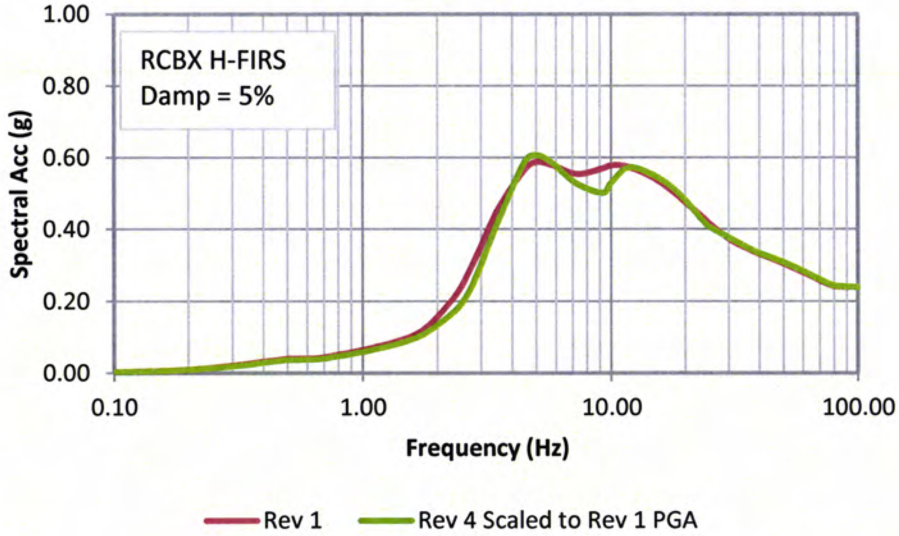


FIGURE 4-1
COMPARISON OF RCBX HORIZONTAL FIRS NORMALIZED TO PGA OF 0.24G

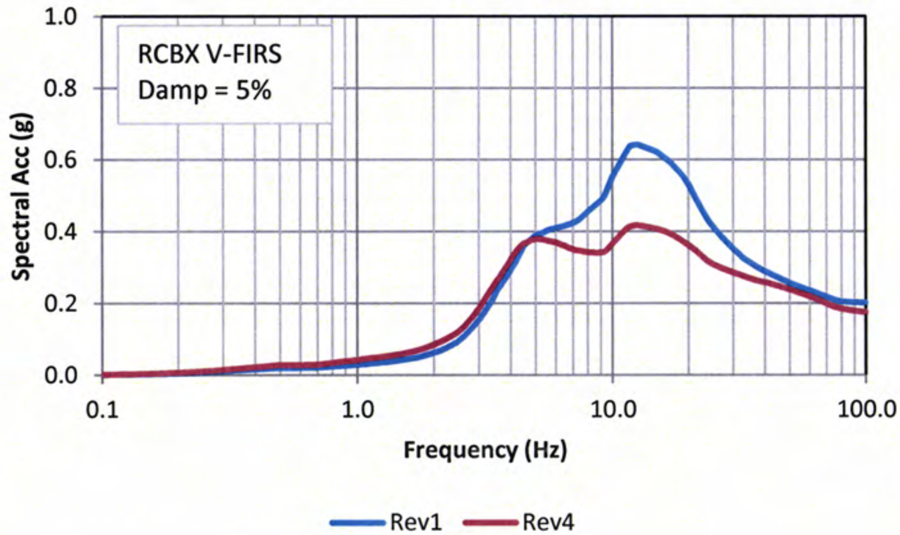


FIGURE 4-2
COMPARISON OF VERTICAL FIRS AT RCBX FOUNDATION LEVEL

Thus, the vertical direction ground motion time histories used in the building seismic analysis are conservatively biased. This conservative bias is justified on the basis that the fragilities of most of the SSCs are controlled by horizontal response, and are therefore not expected to be impacted significantly. However, when controlled by the vertical ISRS, fragilities could be improved on a selective basis; e.g., relay fragilities. Additionally, the bias is retained to allow for uncertainties in the regulatory acceptability of using mean

V/H ratios instead of the envelope. This is further discussed and justified in the Fragility analysis Report (Reference 41).

Details of the SSI analyses are provided in the BVPS-1 Building Seismic Analysis report (Reference 43).

A list of structures and descriptions of dynamic analysis approaches are presented in ***Table 4-3***.

TABLE 4-3
DESCRIPTION OF STRUCTURES AND DYNAMIC ANALYSIS METHODS FOR BVPS-1 SPRA

STRUCTURE	FOUNDATION CONDITION	TYPE OF MODEL	ANALYSIS METHOD	COMMENTS/OTHER INFORMATION
Auxiliary Building	Soil	FE	Deterministic SSI	BE case, 1 set of T-H in accordance with ASCE 4-98
Reactor Containment Building	Soil	FE	Deterministic SSI	BE case, 1 set of T-H in accordance with ASCE 4-98
Diesel Generator Building	Soil	FE	Deterministic SSI	BE case, 1 set of T-H in accordance with ASCE 4-98
Fuel Handling / Decon Buildings	Soil	FE	Deterministic SSI	BE case, 1 set of T-H in accordance with ASCE 4-98
Service Building	Soil	FE	Deterministic SSI	BE case, 1 set of T-H in accordance with ASCE 4-98
Main Steam & Cable Vault Building	Soil	FE	Deterministic SSI	BE case, 1 set of T-H in accordance with ASCE 4-98
Intake Structure	Soil	FE	Deterministic SSI	BE case, 1 set of T-H in accordance with ASCE 4-98
Safeguards Building	Soil	FE	Deterministic SSI	BE case, 1 set of T-H in accordance with ASCE 4-98

4.3.3 Structure Response Models

Details of the structural response models development are provided in the BVPS-1 Building Seismic Analysis report (Reference 43). The following subsections summarize the evaluation of existing lumped-mass stick models, analytical modeling procedure, and structure material properties, stiffness, mass and damping.

4.3.3.1 Evaluation of Existing Lumped-Mass Stick Models

The design basis seismic analysis of the BVPS-1 structures utilized lumped-mass stick models (LMSM). These models represent the entire mass of a floor slab concentrated at one point. The point masses are then connected with a beam or "stick" representing the respective story stiffness. These models are typical of the prevailing practice when BVPS-1 design was performed.

RIZZO assessed the acceptability of using stick models in the SPRA project in light of the ASME/ANS requirements (Reference 47). The report compares in-structure response spectra (ISRS) obtained using stick models to the ISRS based on independently developed finite-element models (FEM) for three representative buildings of the Davis-Besse Nuclear Power Station; namely Auxiliary Building Area 7, Reactor Building's Internal Structure, and the Reactor Shield Building.

Based on the comparisons of the ISRS, the Report concludes that the ISRS from the FEMs are not enveloped by the ISRS from the existing stick models over the entire range of frequencies of interest. However, improvements to the existing stick models to include appropriate representation of flexural stiffness, mass eccentricities, and rigid body rotations may result in acceptable response results.

Because of the significant effort expected to upgrade the existing stick models coupled with the possibility of such models being challenged, the study reported here develops new analytical models based on the FE method. These models represent state of the current practice. However, as a global verification, the total masses used in stick models have been compared to the values represented in the corresponding FE models. The differences are smaller than 10 percent.

4.3.3.2 Development of FE Structure Response Models

The building structure finite element models models are based on geometric information, such as building dimensions, wall and slab thicknesses, structural member locations, and size of openings, etc., taken from building structure layout drawings and details. The parametric information, such as the material properties, live loads, equipment loads, and boundary conditions are obtained on the basis of drawings, existing reports, and appropriate codes and standards.

Figure 4-3 presents the generic flow chart describing the procedure utilized to develop and check the FEMs. The structural FEMs are suitably modified for use with the program SASSI in the seismic SSI analysis.

The modeling effort for the building structure starts with the preparation of three dimensional (3-D) drawings representing the building geometry using software with a graphical interface, such as AutoCAD or RISA. This step develops the geometrical representation of the structural components of the building, such as the foundation and floor slabs, walls and openings, and defines the mid-planes of floors and walls. The geometric model is imported into SAP2000 for FE meshing, assigning element types, and material characteristics in support of developing the structural model. Loads, boundary conditions, and any other special analytical requirements are then incorporated to complete the analytical models.

Most of the building structures which house equipment are analyzed using models which represent the building geometry as described above, as well as the dynamic seismic interaction with the supporting geotechnical medium. The models are sufficiently representative to extract seismic forces on the structural components and to develop the ISRS at locations of interest for use in the analysis of the equipment supported in the buildings.

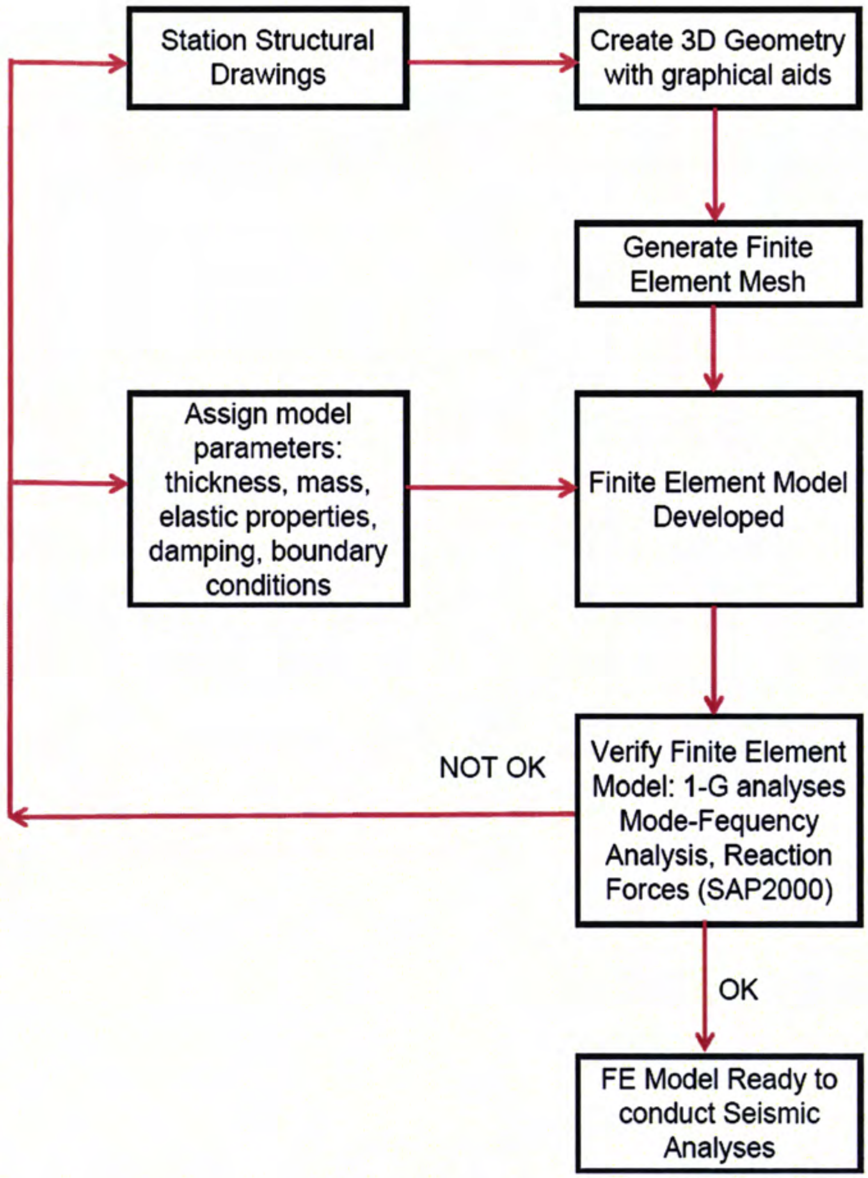


FIGURE 4-3
FLOW CHART DESCRIBING DEVELOPMENT OF FEM

4.3.3.3 Material Properties & Structure Stiffness and Mass

The building seismic analyses are performed using the best estimate values of structure stiffness and mass, the BE subsurface V_S profile compatible with the expected seismic shear strains, and "conservative estimates of median damping." In accordance with ASCE 4-98 (Reference 46), this approach is expected to develop approximately 84th percentile seismic response suitable for use in the CDFM analysis.

Table 4-4 presents the general material properties of the materials of construction. Information on the structure specific design drawings is also utilized to confirm the material strength.

TABLE 4-4
STRENGTH AND ELASTIC PROPERTIES OF MATERIALS OF CONSTRUCTION
BEAVER VALLEY POWER STATION - UNIT 1 STRUCTURES

MATERIAL	CONSTRUCTION	STRENGTH	ELASTIC MODULUS	POISSON'S RATIO
Concrete	Auxiliary Building Reactor Containment Building Diesel Generator Building Fuel Handling and Decontamination Building Service Building Safeguards Building Intake Structure Main Steam & Cable Vault	f_c'= 3000 psi	3.1×10^6 psi	0.25
Rebar ASTM A615, Gr 60	No. 3 to No. 18	F_y = 60 ksi	29.0×10^3 ksi	0.30
ASTM A 36 - Structural	Structural shapes, system supports, component supports	F_y = 36 ksi	29.0×10^3 ksi	0.30

Reference: BVPS-1 UFSAR (Reference 29)

The values of the Young's Modulus in ***Table 4-4*** are generally in agreement with those based on ACI 349-06 (Reference 48) for normal weight concrete ($E_c = 57{,}000\sqrt{f_c'}$). The value of the Poisson's ratio is taken to be 0.25 so that the concrete shear modulus $G_c = 0.4\ E_c$, which is consistent with ASCE Standard 43-05 (Reference 49). A unit weight of 150 pounds per cubic foot (pcf) has been adopted for analyses. This value corresponds to normal weight concrete used in the building construction. Consistent with the expected Response (damage) Level, full or effective stiffnesses are used for concrete members recommended in ASCE/SEI 43-05 (Reference 49) as shown in ***Table 4-5***.

TABLE 4-5
EFFECTIVE STIFFNESS OF REINFORCED CONCRETE ELEMENTS
(REFERENCE 49)

Member	Flexural Rigidity	Shear Rigidity	Axial Rigidity
Beams—Nonprestressed	$0.5\ E_cI_g$	G_cA_W	
Beams—Prestressed	E_cI_g	G_cA_W	
Columns in compression	$0.7\ E_cI_g$	G_cA_W	E_cA_g
Columns in tension	$0.5\ E_cI_g$	G_cA_W	E_sA_s
Walls and diaphragms—Uncracked	E_cI_g $(f_b < f_{cr})$	G_cA_W $(V < V_c)$	E_cA_g
Walls and diaphragms—Cracked	$0.5\ E_cI_g$ $(f_b > f_{cr})$	$0.5\ G_cA_W$ $(V > V_c)$	E_cA_g

The shear stiffness of walls and diaphragms is represented assuming cracked section properties (***Table 4-5***) for in-plane shear. Subsequent to the SPRA quantification, a selected sample of the shear walls for the plant structures was assessed to confirm the assumption. This assessment shows that the shear demand corresponding to the median failure capacities of controlling SSCs (HCLPF of about 0.5g PGA) exceeds the concrete shear capacity, which is 2 $\sqrt{f'_c}$ in accordance wit ASCE 43-05. The assessment shows that most walls are cracked.

4.3.3.4 Structural Damping

Dynamic analyses of BVPS-1 structures use a concrete structural damping of 4 percent of critical for concrete members and 2 percent for steel structural members. This level of damping considers that the buildings will enter only into Response Level 1 as defined in ASCE/SEI 43-05 (Reference 49). An assessment of damage state in accordance with ASCE/SEI 43-05 (Reference 49) for a selected sample of walls shows that most walls remain in Response Level 1.

4.3.4 Seismic Structure Response Analysis Technical Adequacy

The BVPS-1 SPRA Seismic SSI Analysis and the Structure Response were subjected to an independent peer review against the pertinent requirements in the PRA Standard (Reference 4).

The peer review assessment, and subsequent disposition of peer review F&Os, is described in ***Appendix A***, and establishes that the BVPS-1 SPRA Seismic Structure Response and SSI Analysis are suitable for this SPRA application.

4.4 SSC Fragility Analysis

The seismic fragility analysis develops the probability of SSC failure for a given value of the PGA. The fragilities are developed for all of the SSCs that participate in the SPRA accident sequences and included on the SEL. The fragility analysis for the significant risk contributors is particularly based on plant-specific information, and actual current conditions of the SSCs in the plant, as confirmed through the detailed walkdown of the plant, so that the resulting fragility estimates are realistic.

This section summarizes the fragility analysis methodology, presents a tabulation of the fragilities (with appropriate parameters, 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 5.0***). Important assumptions and important sources of uncertainty, and any particular fragility-related insights identified, are also discussed.

4.4.1 SSC Screening Approach

In the context of a SPRA, high capacity components may be screened if their HCLPF capacity is in excess the PGA at very low exceedance frequency (e.g., $2x10^{-7}$) on the site-specific hazard curve. The items screened out in this manner require no further fragility analysis as the screening level capacity already contributes negligibly to the CDF. However, the associated screening level at the Beaver Valley site is relatively high, and very few items can be screened out.

A more appropriate screening level is established on a quantitative basis so that the maximum possible increase in CDF/LERF that can be added from accelerations greater than the screening threshold does not exceed $2x10^{-7}$ to CDF, or $1x10^{-8}$ addition to LERF. This quantitative approach uses the CDF/LERF interval success frequency (i.e., the hazard frequency which does not go to core-damage/large early release) and results in a screening threshold of 0.6g for excluding SSCs from the level 1 PRA model and 2.0g for excluding SSCs from the level 2 PRA model.

The screening strategy implemented for the BVPS-1 Fragility Analysis is based on the following considerations and is supported by the walkdowns:

- The screening is based on the site-specific seismic hazard for the BVPS-1.
- The fragility analysts focus most of their analytical resources on equipment likely to govern the seismic risk, and to minimize their efforts on more robust equipment, or on equipment judged seismically so weak as to not provide any benefit.
- To demonstrate that all seismic risk contributors to CDF and LERF are eventually included in the SPRA, the final screening criterion was adjusted upward based on the fragility estimates for evaluated equipment. The intent is to show that at most, the equipment not evaluated in detail contribute, in aggregate, no more than three to four percent to CDF or LERF.
- Sensitivities were performed on the highest risk fragile components to assess the impact of possible refinement of fragilities. This is documented in the quantification notebook section 6.3.2 (Reference 17).
- SPRA is expected to be used in the future for making risk-informed decisions. For this purpose, it is useful to keep in the system model all the components whose failure may lead to some important accident sequences. In this way, one could judge the impact of upgrading any particular component or even relaxing the test frequency requirements. If the component is screened out and not in the model, the analyst would have to introduce the subject component into the SPRA model for future risk-informed applications.

Where appropriate, the SRT used caveats in the screening tables in EPRI NP-6041-SL (Reference 7) to justify assigning the respective screening level capacities to high seismic capacity components.

The general approach classifies equipment on the SPRA SEL into ranges of HCLPF capacity so as to identify a set of equipment that are seismically strong enough to mitigate risk, yet not so strong that they do not contribute to seismic CDF and LERF. The approach used is as follows:

1. Initially screen from fragility analysis all SSCs that are not Seismic Category 1, as being seismically weak.
2. Screen out all Seismic Category 1 SSCs that are judged seismically no stronger than the fragility for loss of offsite power, again as being seismically weak; i.e., a HCLPF of 0.1g PGA.

3. Screen out rugged SSCs judged to have a seismic HCLPF greater than the screening level as being seismically robust and; therefore, potentially less likely to contribute to seismic CDF or LERF.
4. Evaluate the fragilities for the remaining Seismic Category 1 SSCs judged to have a seismic fragility with HCLPF's between 0.1g and the screening level.
5. Incorporate the evaluated fragilities in Step 4 above into an initial SPRA model to determine the seismic CDF and LERF as a function of seismic hazard level.
6. Subtract the CDF contribution from each seismic range from the seismic hazard frequency curve to obtain the remaining frequency of seismic events that do not result in core damage as a function of PGA. Identify the seismic magnitude in PGA, at which the adjusted exceedance frequency curve corresponds to 3 percent to 4 percent of the computed CDF. Repeat this step for LERF.
7. If the PGA values for maximum added seismic CDF and LERF obtained in Step 6 are less than the screening level, then no additional SSC fragilities need be evaluated. All other unanalyzed SSCs have been shown to have seismic capacities greater than the screening level, or are seismically weak and not credited in the analysis.
8. If the PGA values corresponding to 3 percent to 4 percent of the computed CDF and LERF as derived in Step 6 are greater than the screening level, then additional SSCs should be evaluated. The choice of which SSCs are to be evaluated next is to be decided by discussions between the fragility analysts and the PRA analysts. Most likely SSCs selected from those initially judged to have HCLPFs greater than the screening level are to be evaluated next. The collaboration between the fragility analysts and PRA modeling team is to also consider how the initial contributors to CDF and LERF can be mitigated by SSCs not yet credited; e.g., by SSCs screened because they were not Category 1. After the fragility analyses of more SSCs, repeat Step 4 through Step 6 until the CDF and LERF PGA values in Step 6 are less than the screening level, or some higher acceleration level that the fragility analysts can justify that all other SSCs meet.

The assignment of SSCs to ranges of HCLPFs is supported by EPRI NP-6041-SL (Reference 7). Therein caveats are provided for equipment to meet in order to assign a generic seismic capacity. The generic seismic capacity is based on seismic experience as well as results from prior SPRAs. The screening level to be applied to BVPS-1 components that meet the EPRI caveats is 1.8g S_A per References 7, 44, and 50. This screening level capacity is a HCLPF capacity level and assures the survival of the equipment and function after the earthquake. Anchorage must be verified to also have a HCLPF capacity of at least 1.8g S_A.

Fragilities of components, based on the screening level HCLPFs, were developed as follows:

1. The clipped peak of the 84th percentile non-exceedance probability (NEP) spectra at the equipment location, or the S_A at or greater than the lowest estimated/calculated/tested equipment frequency was compared to the 1.8g screening level to determine the ratio of the screening level to the 84th percentile NEP demand.
2. The HCLPF of the component was determined as the ratio in Step 1 times the site-specific Control Point PGA; i.e., 0.24g PGA.

3. Anchorage HCLPF was determined in accordance with EPRI NP-6041-SL (Reference 7) procedures and using the 84th percentile NEP floor spectra as the demand.
4. The governing HCLPF was determined as the component screening level, or component's demonstrated test capacity or the anchorage capacity. If the component was subjected to seismic interaction effects, then the resulting HCLPF was the lowest HCLPF, including the HCLPF due to seismic interaction effects.
5. In accordance with the recommendations in Reference 2 a generic composite uncertainty, β_C, ranging from 0.35 to 0.45 was assumed.
6. The median ground acceleration capacity of the screened component was calculated from the governing HCLPF as:

   $A_m = HCLPF(e^{2.33*\beta_C})$

β_C was broken down into a β_R of 0.24 to represent randomness in the ground motion and response and β_U ranging from 0.26 to 0.38 to represent uncertainty in response and capacity per Reference 2 Table 6-2.

Based on the walkdown observations and past SPRA experience, we conclude the following:

- SEL items deemed to meet the 1.8g S_A limit can be assigned a generic seismic fragility.
- Manually-operated valves on the SEL, are judged to have high seismic capacities. They were removed from the SPRA systems model.

For the SEL items not "screened out" specific seismic fragilities were developed using the design data and walkdown observations.

4.4.2 SSC Fragility Analysis Methodology

For the BVPS-1 SPRA, the following methods were used to determine seismic fragilities for SSCs included in the SPRA. Overall, fragilities of Seismic Category 1 structures were calculated following the separation of variables method whereas the remainder of SSCs not screened out was established using the CDFM method considering betas recommended in Table 6-2 of the SPID (Reference 2). The following subsections describe the implementation of the technical approach in developing the seismic fragilities for the BVPS-1 SSCs.

4.4.2.1 Fragility Evaluation Standards and Guides

The standards and guidelines used to develop the fragilities of SSCs are identified below.

1. EPRI TR-103959 "Methodology for Developing Seismic Fragilities" (Reference 11).
2. EPRI 1002988 "Seismic Fragility Application Guide" (Reference 50).
3. EPRI 1019200, "Seismic Fragility Applications Guide Update" (Reference 44).
4. EPRI NP-6041-SL, "Nuclear Plant Seismic Margin" (Reference 7).
5. ASCE/SEI 43-05, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities" (Reference 49).

6. ASCE 4-98, "Seismic Analysis of Safety Related Nuclear Structures" (Reference 46).
7. EPRI 1025287, "Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic" (Reference 2).

4.4.2.2 CDFM Method

The CDFM method is described in detail in Reference 7. The CDFM HCLPF values are determined using the following expression:

$$HCLPF = F_S \cdot F_\mu \cdot PGA$$

where,

F_S = Strength factor derived from the comparison between seismic demand (ISRS) and the conservative estimate of seismic capacity (e.g., GERS or design analysis),

F_μ = Inelastic energy absorption factor (taken as 1.0 for brittle failure modes),

PGA = Peak ground acceleration of the BVPS-1 control point (i.e., Reactor Containment Building foundation at EL 681 ft) FIRS = 0.24 times the acceleration of gravity (g)

The median capacity A_m developed in terms of the CDFM approach was estimated by using the following equation:

$$A_m = HCLPF \cdot e^{2.33(\beta_C)}$$

where,

β_C = Composite logarithmic standard deviation due to randomness and uncertainty,

The median capacity estimates, A_m are developed using β_c values recommended in Table 6-2 of the SPID (Reference 2) for various types of SSCs. These values are shown below in ***Table 4-6*** along with the corresponding β_r and β_u values.

TABLE 4-6
RECOMMENDED LOGARITHMIC STANDARD DEVIATIONS FOR SSC
(SPID GUIDELINE, TABLE 6-2)

TYPE SSC	COMPOSITE β_C	RANDOM β_R	UNCERTAINTY β_U	$C_{50\%}/C_{1\%}$
Structures & Major Passive Mechanical Components Mounted on Ground or at Low Elevation Within Structures	0.35	0.24	0.26	2.26
Active Components Mounted at High Elevation in Structures	0.45	0.24	0.38	2.85
Other SSCs	0.40	0.24	0.32	2.54

4.4.2.3 Separation of Variables Method

The direct method, using separation of variables, develops median capacity on the basis of the median factor of safety (FOS), F_M, which defines the relationship between A_m and the value of the ground motion parameter corresponding to the analysis spectra (EPRI TR-1002988 Seismic Fragility Application Guide, 2002, EPRI TR-103959 Methodology for Developing Seismic Fragilities):

$$A_m = F_M \times A_{RLE}$$

where,

F_M is the seismic safety factor

A_{RLE} is the peak ground acceleration (g)

For structures, F_M is defined by:

$$F_M = F_{RS} \times F_C$$

F_C is the seismic capacity factor defined as:

$$F_C = F_S \times F_\mu$$

where,

F_S is a factor associated with strength

F_μ is a factor associated with ductility

F_{RS}, the structural response factor, was calculated by SOV as a combination of several factors that affect the seismic response:

F_{GM} = Ground Motion Factor,

F_D = Damping Factor,

F_M = Modeling Factor,

F_{MC} = Modal Combination Factor,

F_{TH} = Time History Simulation Factor,

F_{SSI} = Soil-Structure Interaction Factor,

F_{EC} = Earthquake Component Combination Factor,

F_{HD} = Horizontal Direction Peak Response,

F_{VC} = Vertical Component Response.

Thus, F_{SR} is defined as:

$$F_{SR} = F_{GM} \cdot F_D \cdot F_M \cdot F_{MC} \cdot F_{TH} \cdot F_{SSI} \cdot F_{EC} \cdot F_{HD} \cdot F_{VC}$$

Combining the capacity and the response factors the overall median FOS is:

$$F_M = F_C . F_{RS}$$

$$\beta_R = (\beta^2_{R,C} + \beta^2_{R,RS})^{1/2}$$

$$\beta_U = (\beta^2_{U,C} + \beta^2_{U,RS})^{1/2}$$

4.4.2.4 Seismic Demand

The FIRS developed in Reference 23 are of significantly different shapes than the design basis earthquake (DBE) Safe Shutdown Earthquake (SSE) response spectra. Therefore, scaling of the DBE seismic response and the floor response spectra was not considered adequate to obtain median capacities. Instead, the fragility calculations reported here are based on seismic re-evaluation of facility structures using the new evaluation basis earthquake ground motion. This re-evaluation also updates the analytical models of the structures as described in ***Section 4.3***.

The seismic demand on the plant SSCs (in terms of forces and moments on building structural components, and in-structure floor response spectra) is obtained on the basis of seismic soil-structure-interaction analysis of selected buildings as reported in the BVPS-1 Building Analysis Report (Reference 43). The seismic SSI analysis is performed following the methodology in ASCE 4-98 (Reference 46), and results in the approximate 84th percentile seismic demand.

For structure fragilities evaluated using the separation of variables approach, the median demand is obtained on the basis of the calculated 84th percentile NEP forces and moments resulting from the SSI analyses, and the median demand conservatism ratio factor from the equation in EPRI Report 1019200 (Reference 44). A seismic demand logarithmic standard deviation of 0.2 is used in the equation based on an interpretation of data presented as part of probabilistic SSI studies in literature (References 51 and 52). The resulting median demand conservatism ratio is 1.18.

The seismic demand on equipment is evaluated independently using the 84th percentile NEP floor response spectra at selected points close to the equipment support location. Unlike design analysis, the equipment response used in the CDFM approach is typically based on un-broadened in-structure response spectra (ISRS) and frequency shifting. EPRI NP-6041-SL (Reference 7) recommends the damping values to calculate the equipment seismic demand for use in the CDFM method. These damping values are presented here in ***Table 4-7***.

TABLE 4-7
RECOMMENDED EQUIPMENT DAMPING FOR ANCHORAGE BASED ON EPRI NP-6041-SL

EQUIPMENT TYPE	DAMPING
Electrical Cabinets Bolted or Welded to Floor	5%
Light, Welded Instrument Racks	3%
Massive, Low-Stressed Components (Pumps, Motors)	3%
Piping	5%
Cable Trays	15%
Fluid Containing Tanks - Impulsive Mode	5%
Fluid Containing Tanks - Sloshing Mode	0.5%

4.4.2.5 Fragility Evaluation of Seismic Category 1 Structures

The building structures listed below are included in the fragility analysis. The fragilities of these structures are based on new analysis using the separation of variables method previously summarized. The method is described in detail in EPRI TR-103959 (Reference 11). Other structures are evaluated on the basis of simplified analysis.

- Auxiliary Building (AXLB)
- Reactor Containment Building (RCBX)
- Diesel Generator Building (DGBX)
- Fuel Handling and Decontamination Building (FULB)
- Service Building (SRVB)
- Safeguards Building (SFGB)
- Intake Structure (INTS)
- Main Steam and Cable Vault (MSCV)

The seismic capacity of a structure is typically controlled by the capacity of the shear walls, which are the primary lateral load resisting elements. Floor diaphragms are screened on the basis that the seismic margins for these components are generally higher than for the shear walls. The diaphragm shear develops only from the lateral forces on the floor, while the shear walls particularly near the base are subjected to lateral forces accumulated from the stories above. Based on the typical floor slab thickness (two feet) and span configurations of the floor diaphragms of the BVPS structures, it is judged that their fragilities do not govern over in-plane shear or flexure fragilities of shear walls near the base.

Within each structure, critical walls are selected for evaluation of fragility. Critical structural members are major walls which failure poses a potential failure of the structure. Yielding of minor walls is not a concern since loads in these walls will be redistributed to the major shear walls. Of these critical walls selected for evaluation, the one calculated to have the lowest safety factor is taken to represent the fragility of the building.

Critical walls of a building are generally located at stories which exhibit the most significant inter-story drift based on the displaced shape of the structure under horizontal seismic loads. Typically, two or more floor levels of the building are considered where representative walls are evaluated. One is at the foundation level, where the walls are expected to carry the largest shear forces accounting for the total base shear for the structures. A second story level is based on observable inter-story drift. This story is expected to introduce the largest shear deformations in the shear walls.

The fragility of a reinforced concrete wall reflects the strength of the wall accounting for the ultimate strength of the concrete, the yield strength of the reinforcing steel and the energy absorption as the component is cycled in the inelastic range.

The strength capacity calculations follow consensus codes and industry guides such as ACI 318 and EPRI 103959 to evaluate potential failure modes, such as diagonal shear cracking, flexure, and shear friction in walls. In general, the critical failure modes of concrete shear walls in Seismic Category I buildings of the BVPS-1 are diagonal shear and flexure. Shear friction is not considered to be a credible failure mode for the BVPS shear walls. This is because there are either no horizontal construction joints, or because the joints are prepared to result in bonding between concrete placed at different times. Similarly, due to heavy reinforcement, the failure mode involving compression failure of the shear wall end sections is not predicted.

The inelastic energy absorption is related to the hysteresis as the structure describes inelastic displacements in sustaining loads up to the ultimate strength of the structural elements. The fragilities of the buildings are evaluated considering two limit states, according to ASCE/SEI 43-05 (Reference 49).

1. Limit State C (LS-C) defined as limited permanent deformation, and
2. Limit State A (LS-A) defined as short of collapse, but structurally stable.

ASCE 43 LS-C corresponds to the point where the structure exhibits sufficient strain to induce cracking and cause incipient failure of the anchorage of mounted components. ASCE 43 LS-A corresponds to an advanced limit state allowing permanent inelastic deformations short of collapse, but structurally stable. This limit state is more representative of gross failure of the structure, whereas LS-C represents a failure of equipment housed within the structure. Inelastic energy absorption factor values presented in Table 5-1 of ASCE/SEI 43-05 (Reference 49) consistent with the limit state being evaluated are selected and converted to median level for use in the separation of variables fragility evaluation of the walls.

With the exception of structural damping, all other variables in the building seismic analysis are median values. A conservative value of structural damping (4 percent of critical) is used to develop the ISRS for use in the CDFM calculations. However, a higher damping is used in the fragility analysis of the structure itself with the value depending on the limit state being evaluated. For LS-C, 7 percent of critical damping is considered as median. A higher damping of 10 percent of critical is selected for LS-A consistent with the advanced degree of damage.

4.4.2.6 Grouping of Equipment for Seismic Evaluation

The equipment screened-in for evaluation are grouped to condense the equipment list into a reasonable number of groups containing similar equipment based on several attributes, including the following:

- Equipment types (SQUG GIP Classes), such as horizontal and vertical pumps
- Associated systems
- Potential concerns encountered which could impact the seismic capacity
- Location, such as building and floor elevation
- Size
- Manufacturer

Observations made during walkdowns are also utilized to assess if components included in a group need a specific evaluation (as opposed to generic approaches) to establish a capacity. For example:

- Component does not meet all caveats of respective GIP class; e.g. valves with excessively cantilevered actuators.
- Supplemental supports, such as snubbers, rigid struts, or hangers for valve yokes.
- Potential of seismic interactions.

Where differences in physical characteristics, such as the dimensions, weight, manufacturer, etc., are observed for components included in EPRI equipment classes, additional sub-groups were created so that representative HCLPF values could be developed. Finally, components within groups are subdivided based on building and elevations to address the differences in floor response spectra.

In some instances, a relatively large number of components were grouped together and represented by a component that reasonably bounds the seismic capacity of other components in the group. The inherent conservatism in this approach is justified on the basis that the bounding capacity exceeds the risk significance level. Therefore, the seismic fragilities of all of the components bounded by this representative component also have a negligible quantitative impact on the PRA results.

4.4.2.7 Fragility Evaluation of Mechanical and Electrical Equipment

In general, fragilities are evaluated for the equipment functional and structural/anchorage capacities, as well as relay and potential interactions where applicable. Functional fragility is typically established by comparing the ISRS near the equipment, clipped according to EPRI 6041, to a capacity spectrum in a frequency range of interest. Most equipment functional capacities are established on the basis of experience data, generic equipment ruggedness spectra (GERS), or qualification test data. These capacities do not represent the anchorage capacity of the equipment and accordingly anchorage fragility evaluation is also necessary where these approaches are used. Anchorage fragility is typically calculated by scaling design basis analyses or by new analysis. For passive equipment such as tanks, only a structural/anchorage fragility is evaluated.

4.4.2.8 Fragility Evaluation Based on Experience Data

A HCLPF capacity based on earthquake experience data, when used, is justified by documenting that the associated caveats are satisfied. EPRI NP-7149-D (Reference 56) and its supplement EPRI NP-7149-D-S1 (Reference 57) document the development of these caveats based on extensive surveys and cataloging of the effects of strong ground motion earthquakes on various classes of equipment mounted in conventional power stations and other industrial facilities. The seismic experience database presented in these reports reflects detailed investigations of some 120 sites within the strong-motion regions of some 23 earthquakes from 1971 to 1993 by SQUG, EPRI, and EQE International.

EPRI 1019200 (Reference 44) presents further analysis for the earthquake experience database. It concludes that components satisfying the requirements to assign a 1.2g capacity in terms of PGA will exhibit HCLPF capacities developed as follows:

For ground-mounted items, the mounting level capacity is 1.32g for comparison to either free-field demand or clipped in-structure demand spectra.

- For structure-mounted items, the mounting level capacity is 1.80g for comparison to clipped ISRS demands.
- These experience-based capacities of 1.80g and 1.32g can be used to develop a component functional HCLPF capacity in a manner similar to a capacity response spectrum developed by testing, such as a GERS.

The ISRS, which reflect the calculated floor response spectra, often exhibit highly amplified narrow frequency content. These narrow peaks are not well correlated with potential structural or functional failure. Therefore, when comparing peaked floor response spectra with an experience-based capacity, the peaks in the floor response spectra are clipped as described in Appendix Q of EPRI NP-6041-SL (Reference 7).

4.4.2.9 Fragility Evaluation Based on Test Data

The seismic capacity of components qualified on the basis of tests (e.g., electrical cabinets) may utilize either specific qualification testing or generic test data. The seismic capacity is determined as the ratio of the TRS to the required response spectra (RRS) associated with the evaluation basis earthquake. In order to bias the capacity to CDFM level of conservatism, the selected TRS is associated with a 99 percent exceedance probability. Depending upon whether the testing is assembly based or device-based, local amplification may be incorporated to obtain device-based capacities (using, for example, in-cabinet response spectra).

Several reference documents, such as EPRI NP-6041-SL (Reference 7), EPRI TR-103959 (Reference 11), EPRI NP-5223-SL (Reference 58), and SQUG/GIP (Reference 18), present the methodology to develop CDFM level capacities based on Test Response Data for specific classes of M&E equipment. These documents specify the conditions (caveats), under which the GERS may be used. Available TRS for specific equipment are also considered to develop seismic capacities. However, the TRS are taken to represent a LB of the capacity of the respective equipment.

Where CDFM level capacities were assigned based on generic test data, the walkdown observations provided the basis for considering that the associated caveats are satisfied.

The ISRS, which reflect the calculated floor response spectra, often exhibit highly amplified narrow frequency content. These narrow peaks are not well correlated with potential structural or functional failure. Therefore, when comparing peaked floor response spectra with a TRS capacity, the peaks in the floor response spectra are clipped as described in Appendix Q of EPRI NP-6041-SL (Reference 7).

4.4.2.10 Fragility Evaluation Based on New Analysis or Scaling of Existing Analysis

Typical codes and standards used in the qualification of equipment by analysis include those published by ASME, American Institute of Steel Construction (AISC), ACI and Institute of Electrical and Electronics Engineers (IEEE) Standard. Additionally, EPRI NP-6041-SL (Reference 7) identifies load combinations and stress limits for pressure retaining components, supports, and anchorage.

When equipment is qualified based on design analysis, it was recognized that the component design capacity is determined by code specified stress and design displacement limits. The CDFM capacity, on the other hand, is obtained for as-built conditions using stress limits corresponding to the code specified minimum stress or the material yield strength with a 95 percent exceedance probability. However, for non-ductile materials EPRI NP-6041-SL (Reference 7) suggests using 70 percent of the material yield as the stress limit.

The evaluation of M&E components based on generic and seismic experience capacities are supplemented with the verification of the equipment anchorage. For anchorage fragility evaluation, approaches include scaling of existing analysis or new analysis. Scaling of existing analyses is performed considering the guidance of EPRI 6041 (Reference 7). New analysis is performed in accordance with the procedure outlined in the SQUG/GIP (Reference 18). This procedure follows a static equivalent approach, where the inertial load of the equipment is applied at the equipment center of gravity. The inertial load in each direction is equal to the product of the Sa, an equivalent static coefficient, and the mass of the equipment. An equivalent static coefficient of 1.0 is used for the anchorage analysis of M&E equipment.

The seismic demand on the equipment anchorage in terms of tension and shear is developed consistent with the following equipment characteristics:

- Mass of the Equipment
- Location of the Center of Gravity
- Natural Frequency
- Equipment Damping
- Equipment Base Center of Rotation

The equipment mass defines the inertial loads, while the location of the center of gravity determines the overturning moment caused by the inertial loads. The seismic anchorage demand of the equipment is determined by shifting the appropriate floor response spectrum to account for the effects of the uncertainties in the structural frequencies, according to EPRI NP-6041-SL (Reference 7). Then, the lowest natural frequency of the equipment is used to determine the amplified acceleration of the equipment from the shifted ISRS.

4.4.2.11 Fragility Evaluation of Distribution Systems Components

Distribution systems, piping, cable trays and supports, and HVAC are typically treated on a sampling basis and are evaluated using generic data and earthquake experience data. A conservative 0.50g PGA HCLPF value is assigned to distribution systems in the BVPS-1 (i.e., piping, HVAC ducts, and cable trays and conduits) on the basis of earthquake experience data.

Experience from past strong-motion earthquakes in industrial facilities throughout the world indicated that, in general, distribution systems are seismically rugged. The seismic experience data shows that most types of piping systems exhibit extremely good performance under strong-motion seismic loading, with the pressure boundary being retained in all but a handful of cases. The BVPS-1 Walkdown report (Reference 40) presents a summary of walkdown observations, which provide the basis to assign a 0.50g PGA HCLPF value to distribution systems.

4.4.2.12 Fragility Evaluation of Relays

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 BVPS-1 SPRA, in accordance with SPID (Reference 2) and ASME/ANS PRA Standard (Reference 4). The evaluation resulted in most relay chatter scenarios screened from further evaluation based on no impact to component function. The relays that were not screened were addressed in the SPRA with appropriate seismic fragility.

The seismic fragility for the relay chatter mode is developed based on test reports for specific relay models. For the relay chatter evaluation, the CDFM methodology is followed as described in EPRI 6041 (Reference 7).

Appropriate cabinet amplification factors, AF_C, are considered to scale the ISRS to an estimated mounting point spectrum. In general, amplification factors from Table Q-1 of EPRI 6041 (Reference 7) are used for the horizontal direction and EPRI 3002004396 (Reference 39) for the vertical direction. The recommended factors are shown in ***Table 4-8*** below. As stated in EPRI 6041 (Reference 7), the amplification factors are generally conservative for most location within the cabinets. In the case of medium voltage switchgears at BVPS-1, the maximum in-cabinet amplification factors are obtained from actual shake table tests, which are 3.9, 5.3, and 1.8 for the front-to-back, side-to-side, and vertical directions, respectively (Reference 87).

TABLE 4-8
RECOMMENDED CABINET AMPLIFICATION FACTORS
(EPRI 6041 (REFERENCE 7), EPRI 3002004396 (REFERENCE 39)

DIRECTION	CABINET TYPE	AMPLIFICATION FACTOR, AF
Horizontal	Motor Control Centers	3.6
	Low and Medium Voltage Switchgears	7.2
	Stiff Panels and Control Boards	4.5
Vertical	All	4.7

A knockdown factor, F_k, has been considered to obtain about a 99 percent exceedance level capacity. Representative knockdown factors are presented in Table Q-2 of EPRI 6041 (Reference 7) and reproduced in ***Table 4-9*** below. Knockdown factors corresponding to IEEE C37-98 Relay Fragility Tests, GERS – Relays, and Component-Specific Qualification Test: Function During are used for the BVPS-1 relay evaluation.

TABLE 4-9
RECOMMENDED TRS KNOCKDOWN FACTORS (EPRI 6041 (REFERENCE 7))

DATA SOURCE	KNOCKDOWN FACTOR, F_K
HCLPF Capacities	1.0
GERS – Non-Relays	1.2
GERS – Relays	1.5
IEEE C37-98 Relay Fragility Test	1.08
Component-Specific Qualification Test: Function During	1.2
Component-Specific Qualification Test: Function After (No Anomalies)	1.0
Component-Specific Qualification Test: Function After (Anomalies)	1.1 – 1.6

TRS are all broad banded and are not clipped, but RRS were clipped as appropriate. Therefore, C_T factor is 1.0 with no uncertainty. Per EPRI 6041 (Reference 7), when the TRS are for multi-axis excitation, and the RRS is predominantly a single-axis excitation, as is the case for relays and contactors mounted on panels in cabinets, then the TRS should be increased by a multi-axis to single-axis correction factor, F_{MS}, to remove the unnecessary conservatism. EPRI 6041 (Reference 7) suggests F_{MS} = 1.20.

4.4.2.13 Fragility Evaluation of NSSS Components

Ten NSSS components are included in the SEL: Pressurizer, three Reactor Coolant Pumps, Reactor Internals, Control Rods, Reactor Vessel, and three Steam Generators. The fragilities for these NSSS components are based on new analysis, design basis criteria, scaling available seismic calculations, and earthquake experience data.

4.4.2.14 Fragility Evaluation of Block Walls

The evaluation of the masonry block walls is based on the Elastic and Reserve Energy methodologies presented in Section 10.5 of DOE/EH-0545 (Reference 76). This approach is used to estimate the HCLPF, median seismic capacities, and associated uncertainty and randomness.

The seismic walkdown summarized in the Walkdown Report (Reference 40) identified masonry walls that are judged to present potential interaction concerns to nearby components on the SEL. Such walls are subjected to seismic evaluation.

Concrete Masonry Unit (CMU) walls evaluated in BV1 are either 12- or 24-inch thick unreinforced masonry blockwalls. The 24-inch thick walls are double-wythe walls with 12-inch blocks in each wythe. Anchorage to floor and/or ceiling was not shown in available documentation; therefore, it is conservatively judged that no anchorage is present. Boundary conditions of the walls are determined on a case-by-case basis.

4.4.2.15 Fragility Evaluation of Non-Seismic Category 1 SSCs

A 0.10g HCLPF capacity is assigned to all Non-Cat 1 SSCs prior to any fragility calculation unless a higher capacity was requested by the PRA analyst. The basis for this capacity is that it corresponds to the HCLPF recommended for loss of offsite power (LOOP) per the EPRI SPRAIG Report 3002000709 (Reference 15), NUREG-1738 (Reference 59), and NUREG-CR-3558 (Reference 60). The representative failure mode for LOOP is the brittle failure of the ceramic insulators on transformers per NUREG-CR-4334 (Reference 61) and NUREG-CR-3558 (Reference 60). A key function of non-Cat 1 equipment relates to bringing offsite power into the Station. The equipment that supports this function is judged to have HCLPFs greater than or equal to that of offsite power. Therefore, the seismic capacity of off-site power constitutes the weak link in the system. The equipment that supports systems that bring off-site power into the Station are limited by the seismic capacity of LOOP and accordingly may be assigned the same capacity. Other Non-Seismic Category 1 SSCs not related to LOOP are assigned a conservative low HCLPF capacity of 0.1g on the basis that they have such low impact on the SPRA results and risk quantification is not sensitive to the conservatism in their fragilities.

4.4.2.16 Fragility Refinement Process

The objective of refining seismic fragilities is to assess unintended conservatism in the fragility parameters to subsequently achieve an acceptable risk level quantified in terms of CDF or LERF. The refinement of seismic fragilities for SSCs constitutes an iterative process between the fragility analyst and PRA systems modeler. This iterative process can be summarized as follows:

1. The fragility analyst develops seismic fragilities based on generic methods (i.e., earthquake experience or GERS) and scaling of existing anchorage analysis.
2. This initial set of seismic fragilities is provided to the PRA systems modeler in the form of HCLPF capacities, logarithmic standard deviations, median capacities, and controlling failure modes.

3. By performing initial risk quantification, the PRA modeler records the CDF and LERF values achieved with this initial set of fragilities.

4. The PRA modeler will then proceed to evaluate the risk level and determine if the resulting CDF and LERF fall within an acceptable risk level.

5. In case the resulting CDF and LERF does not represent an acceptable risk level, say greater than 10^{-6}, the PRA modeler will identify and rank the SSCs with the highest contribution to CDF and/or LERF.

6. This list of top contributors is then provided to the fragility analyst with the intent to refine the SSCs seismic fragilities. In order to refine or provide more representative fragilities, the fragility analyst will recur to several methods including:

    - Creating new groups and selecting new representative components.
    - Refining of seismic demand through the development of computer models.
    - Inclusion of a higher ductility factor.
    - Performing a new fragility calculation following the separation of variable approach.

7. After refinement of seismic fragilities, the fragility analyst will convey the newly refined fragilities to the PRA systems modeler for new risk quantification.

8. This process is repeated until an acceptable CDF and LERF risk level has been achieved.

The refinement of seismic fragilities for several SSCs in the BVPS-1 PRA model was performed by following this process until an acceptable CDF or LERF was achieved.

4.4.3 SSC Fragility Results and Insights

Table 5-10 and Table 5-11 in ***Section 5.0*** provide lists of fragilities for SSCs at BVPS-1 determined to be top contributors to risk, based on Fussell-Vesely importance (FVI) from the final SPRA quantifications of CDF and LERF. The Median acceleration capacity A_m and associated variabilities β_r and β_u are provided for each SSC along with their calculation method, and failure mode addressed in the PRA plant model.

4.4.4 Fragility Analysis Technical Adequacy

The BVPS-1 SPRA SSC Fragility Analysis was subjected to an independent peer review against the pertinent requirements in the PRA Standard (Reference 4).

The peer review assessment, and subsequent disposition of peer review findings, is described in ***Appendix A***, and establishes that the BVPS-1 SPRA SSC Fragility Analysis is suitable for this SPRA application.

5.0 PLANT SEISMIC LOGIC MODEL

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

5.1 DEVELOPMENT OF THE SPRA PLANT SEISMIC LOGIC MODEL

The BVPS-1 seismic response model was developed by starting with the BVPS-1 internal events at-power PRA model of record as of January 2013, and adapting the model in accordance with guidance in the SPID (Reference 2) and PRA Standard (Reference 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 BVPS-1 SPRA, the following methods were used to develop the seismic plant response model:

The BVPS-1 PRA is comprised of two major areas of analysis: (1) the identification of seismically-induced sequences of events that could lead to core damage and the estimation of their frequencies of occurrence (the front-end analysis); and (2) the evaluation of the potential response of containment to these sequences, with emphasis on the possible modes of containment failure and the corresponding radionuclide source terms (the back-end analysis).

The overall methodology for both the front-end and back-end analysis can be characterized as the "linked-event tree" approach. Under this approach, a set of linked-event trees was developed for the plant responses needed to model the impacts from seismic initiating events. The model for these plant responses was developed starting from the General Transient event tree set developed for internal events (see Reference 62). This event tree set also considers transient-induced small LOCA. An updated seismic pre-tree was developed to replace the one previously linked to the General Transient event tree set. These event trees allow the safety functions that must be achieved to keep the core cooled to be organized in a way that defines accident sequences that lead to core damage. The potential for failure of each of the safety functions is defined through the construction of a fault tree. The fault trees carry the modeling from the level of safety functions down to the basic hardware failures and human actions (or inactions) that can contribute to a core-damage sequence. Using reliability data assembled from a review of operating experience at BVPS-1 and on an industry-wide basis, the integrated models can be evaluated to yield estimates of the frequencies of the core-damage accidents of concern.

As described in Reference 38, the SPRA model builds upon the internal events PRA accident sequence models documented in Reference 62. A cross-reference between the top events considered in that model and the system notebooks where the analysis for top events is documented is provided in Table B-1 of Reference 63. The internal events PRA consists of many notebooks listed in Table 3.1 1 of Reference 63 document the models used as a starting point. The portion of the internal events PRA sequence models used in the SPRA and the changes made to incorporate seismic failure events are documented in Reference 2.

The back-end analysis is essentially the same as that performed for the internal events PRA, as documented in Reference 64. The back-end analysis performed for the internal events PRA employed both deterministic and probabilistic analysis tools to follow the progression of the core-damage accidents. Computer codes were used to simulate the meltdown of the core, the failure of the reactor vessel due to contact with molten core materials, and the transport and interactions of core debris in the containment. Because of the large uncertainties associated with the progression of a core-damage accident, these deterministic calculations were supplemented with assessments that considered the potential for phenomena different from or more severe than those treated in the computer codes (see Reference 64). The results of that analysis included an assessment of the potential for a variety of containment failure modes for each type of core-damage sequence, and an estimate of the magnitude of the radionuclide release that would be associated with each.

The seismic hazard curve for BVPS-1 is shown on ***Figure 5-1*** below, taken from Figure 6-7 of Reference 23. The 100 Hz spectral acceleration is selected to represent the zero period PGA at the analysis Reactor Containment Building control point. All SSC fragilities are also developed with referenced to this same control point. The BVPS-1 SSE at 0.125g has a mean hazard exceedance frequency of 1.5E-4 per year. The hazard exceedance frequency of 1E-5 is at 0.43g and the exceedance frequency is still at 1E-6 per year at 1.0g.

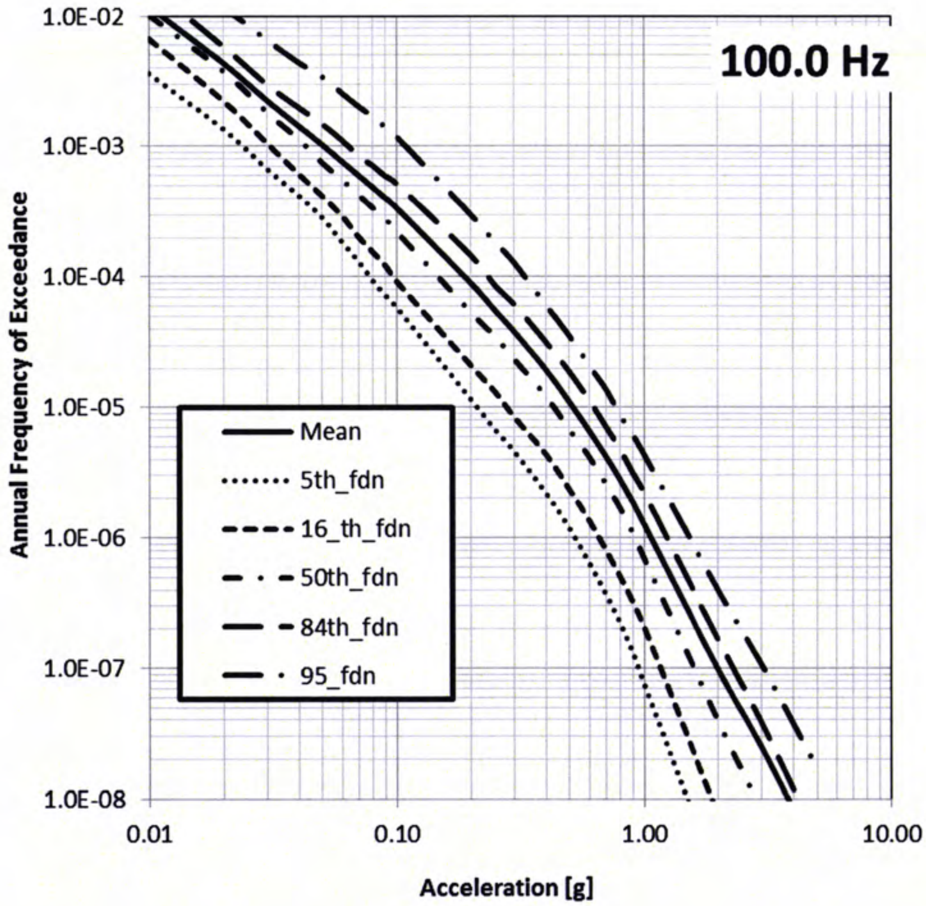


FIGURE 5-1
SEISMIC HAZARD EXCEEDANCE CURVES FOR BEAVER VALLEY SITE AT THE REACTOR CONTAINMENT BUILDING FOUNDATION, INCLUDING UNCERTAINTIES

The BVPS-1 seismic exceedance curves shown on ***Figure 5-1*** are in units of per calendar year. The SPRA model is to assess the risk of at-power plant operation. Therefore, the exceedance curves are scaled by the Unit 1 specific availability factor of 0.927, to obtain the mean exceedance frequency curve for at-power conditions; i.e., the rest of the time the plant is not at-power and the SPRA model does not apply. ***Table 5-1*** lists the scaled and unscaled mean seismic hazard exceedance frequencies at the accelerations provided from Reference 23.

TABLE 5-1
MEAN SEISMIC HAZARD EXCEEDANCE FREQUENCIES SCALED BY PLANT AVAILABILITY

ACCELERATION (g)	EXCEEDANCE FREQUENCY	SCALED BY PLANT UNAVAILABILITY (.927)
	MEAN CURVE	MEAN CURVE
0.01	1.19E-02	1.10E-02
0.02	4.33E-03	4.01E-03
0.03	2.23E-03	2.07E-03
0.04	1.44E-03	1.33E-03
0.05	1.03E-03	9.55E-04
0.06	7.69E-04	7.13E-04
0.07	6.02E-04	5.58E-04
0.08	4.86E-04	4.51E-04
0.09	4.02E-04	3.73E-04
0.1	3.38E-04	3.13E-04
0.2	8.92E-05	8.27E-05
0.25	5.54E-05	5.14E-05
0.3	3.77E-05	3.49E-05
0.4	1.95E-05	1.81E-05
0.5	1.09E-05	1.01E-05
0.6	6.58E-06	6.10E-06
0.7	4.21E-06	3.90E-06
0.8	2.78E-06	2.58E-06
0.9	1.87E-06	1.73E-06
1	1.26E-06	1.17E-06
2	9.70E-08	8.99E-08
3	2.44E-08	2.26E-08
5	3.65E-09	3.38E-09

The seismic initiating event frequencies and their associated acceleration intervals are found in ***Table 5-2***. The analysis acceleration for computing SSC fragilities is also listed. Finally, the four human reliability analysis (HRA) analysis intervals are also associated with the 10 seismic analysis intervals chosen. The basis for this assignment is provided in Reference 38.

The lowest acceleration for the SPRA (0.06g) was selected so that the geometric mean of the acceleration interval would be roughly 0.1g; i.e., the HCLPF value for the off-site power fragility. This same selection has been made for the SPRA models for other FirstEnergy Nuclear Operating Company (FENOC) plants. Relatively narrow acceleration intervals were selected for those ranges of acceleration where the conditional core-damage probability was expected to change most quickly, and to aid in the demonstration that adding new SSC fragilities with higher capacity would not significantly impact the computed CDF. Therefore, constant interval widths of 0.1g were selected for the range between 0.4g to 0.8g. Above 0.8g, the acceleration widths of the remaining seismic initiating event intervals were broadened. The higher range of the acceleration intervals is retained to evaluate LERF. With the exception of G08, the uneven acceleration interval widths still result in the initiating event frequencies decreasing for each interval.

TABLE 5-2
SEISMIC INITIATING EVENT INTERVALS

IE NAME	PGA LOWER	PGA HIGHER	IE FREQ
G01	0.06	0.15	5.33E-04
G02	0.15	0.25	1.09E-04
G03	0.25	0.4	3.31E-05
G04	0.4	0.5	7.91E-06
G05	0.5	0.6	3.99E-06
G06	0.6	0.7	2.19E-06
G07	0.7	0.8	1.32E-06
G08	0.8	1	1.40E-06
G09	1	2	1.07E-07
G10	2	5	8.59E-08
		Freq. Sum =	**6.93E-04**

5.1.1 Seismic Initiating Event Impacts

The purpose of this section is to document the potential initiating event impacts that may be caused by seismic events so that a suitable plant response model to respond to each of the impacts is accounted for in the SPRA. Fortunately, the BVPS-1 Internal Events PRA includes a long list of initiating event impacts and a number of unique plant response models. These plant response models take the form of linked-event tree sets wherein each set contains a seismic pre-tree, a fire analysis tree, a support tree, one or more frontline trees and a containment tree. The event tree sets are best distinguished by their frontline tree names since the other event trees mentioned previously are common to each event tree set, resulting in the following event tee sets:

1. Excessive (e.g., Reactor Vessel Rupture) LOCAs
2. Large LOCAs
3. Medium LOCAs
4. General Transient/ Small LOCAs
5. Steam Generator Tube Ruptures
6. Anticipated transient without Scram (ATWS), for Transients Involving Failure to Trip
7. Interfacing Systems LOCAs

The sequences for these plant response models are created by linking the frontline tree to the other trees in the set, including the containment event tree so that Level 1 and Level 2 end states may be calculated; i.e., where the CDF from seismic events is a sequence group (SEIS) defined as the sum of all release categories at the end of the containment event tree. The sequence group

(LERFS) for large early release from contributed by seismic events is the sum of just those release categories acknowledged to result in a large early release; i.e., release categories BV01, BV01S, BV02, BV02S, BV03, BV03S, BV04, BV04S, BV18, and BV19. The trailing "S" in these release categories indicates that a small containment penetration fails to isolate. Large, early releases result from containment bypasses (BV18 or BV19), or from large, early containment failures (BV01, BV02, BV03, or BV04) with or without an accompanying small containment isolation failure (i.e. as represented by bin name suffix "S").

The basic events included in the internal events PRA models were used in large part to develop the BVPS-1 SEL. These events form a large portion of the SEL. Therefore, the seismic impacts of most SSCs are already accounted for in the internal events PRA models. What has been added to the SEL, are the passive SSC failures and potential relay chatter effects. These passive failures need only be added to the list of seismic impacts affecting a plant response if they are new, cannot be modeled by an existing plant response model, and if the seismic SSC failure probabilities fall below the screening criterion for inclusion in the SPRA model. We have adopted an SSC screening criterion of 0.6g for the SSC HCLPF. SSCs with HCLPFs higher than 0.6g may still be added to the model so long as the required plant response model is available.

For the BVPS-1 seismic PRA, we settled on including only the General Transient/ Small LOCAs event tree set. The reasons are seen in ***Table 4-1*** of ***Section 4.1.1*** where a review of the full list of internal events initiating events is documented for applicability to seismic events.

In summary, the following assumptions and bases are used in the development of the BVPS-1 systems model:

1. The Internal Events PRA was last formally documented in 2013 (Reference 26). An updated version of the model, frozen in July 2014, served as the foundation for the seismic PRA, and a model update was performed in parallel with the seismic PRA that serves as the foundation for the seismic PRA to be finalized and documented at the same time. This new effective reference model is BV1REV6.
2. The Internal Events PRA is used as the technical basis for both CDF and LERF. All assumptions and success criteria in the Internal Events PRA are retained in the SPRA for the portions of the sequence models that apply (see Section 2 of Reference 62). This assumption provides continuity between the Internal Events PRA and the SPRA. Any future changes to the Internal Events PRA success criteria would be addressed as part of the maintenance and update process of the integrated PRA.
3. An SSC HCLPF of 0.35g is used as the screening criterion for excluding potential seismic-induced fires. Please see Section 5.5.3 of Reference 38.
4. The portions of the internal events PRA model that apply to seismic events are: Transients (which include small and very small loss of coolant accidents [SLOCA and VSLOCA] and losses of offsite power) and seismic events assumed to lead directly to core-damage and/or large early release.
5. ATWS sequences are excluded from the SPRA model, on the basis of low frequency; based on multiple redundant trip signals resulting from ground acceleration, as well as highly reliable operation action to trip the plant, it is assumed that the reactor would

successfully trip. Seismic capacity of the control rod drive mechanism was evaluated, but ultimately screened based on high seismic capacity.

6. The spurious, random reactor vessel rupture event sequence model is screened from the SPRA but the seismic failure of the reactor vessel is included. However, seismic capacity of the reactor vessel itself is evaluated, and seismic damage to this component is assumed to lead to core damage.

7. Sequences involving seismic SSCs failures judged to lead directly to core damage (e.g., polar crane in the Reactor Containment Building falling onto the reactor vessel) are guaranteed to be binned to core damage through inclusion of certain event tree rules. These SSCs are represented by Top Event ZL1 (see Section 4.5.1 of Reference 38). However, systems that may have an impact on radiological release categories (e.g., containment spray systems) are still evaluated probabilistically even if Top Event ZL1 fails; i.e., not guaranteed failed. Medium LOCAs have HCLPFs less than the screening criterion but still very high capacity. Therefore, medium LOCAs are conservatively assumed to cause core damage directly and are included in Top Event ZL1 also.

8. Seismic SSCs failures judged to lead directly to core damage plus a large early release (e.g., selected building failures) bypass the usual General Transient initiator event trees, and through the inclusion of certain event tree rules, these sequences are guaranteed to lead to core damage and to a large early failure of the containment, which is always mapped to a large early release category. These SSCs are represented by Top Event ZL2 (see Section 4.5.1 of Reference 38).

9. Sequences involving steam generator tube rupture as a direct result of the seismic motion were not included in the SPRA because no seismic failures that cause a steam generator tube rupture (SGTR) without otherwise failing the steam generator were identified. Pressure- and temperature-induced SGTR following core damage are still evaluated in the containment event tree, and may have an impact on radiological release.

10. The Interfacing systems LOCA (ISLOCA) initiating events model from the Internal Events PRA was reviewed for applicable SSCs, but none were found applicable to seismic failure modes and so the associated sequence model was not used in the SPRA model.

11. The CDF model screening criterion used for excluding SSCs from the SPRA logic models is an SSC HCLPF value of 0.6g or higher. See Reference 17 for a further explanation.

12. The LERF model screening criterion used for excluding SSCs from the SPRA logic models is an SSC HCLPF value of 2.0g or higher. See Section 5.1 of Reference 38 for a further explanation.

13. Large LOCAs are screened from the final SPRA since the minimum acceleration at which they may occur is above the screening criterion for including SSC failures. All Beaver Valley Unit 1 specific NSSS components (Reference 32), large enough to result in these larger breaks, were found seismically robust enough to be excluded. The generic

fragility for large breaks suggested by EPRI (Reference 15) has a HCLPF above the 0.6g screening criterion.

14. SSCs located in the turbine building are not credited in the SPRA sequence models, with exception for the cross-tie cables and the portable air-conditioning (AC) generators. The turbine building is a non-seismic design and so is not resistant to extreme shaking. Further, it contains many SSCs that are also susceptible to seismic shaking. Therefore, while it is expected that the turbine building and SSCs have some seismic capacity to respond to low accelerations, no credit was assumed for the turbine building SSCs with the two exceptions listed above. A fragility was developed for the turbine building to be used in conjunction with the credited SSCs.

15. Although components in the turbine building are assumed failed for all seismic initiators, there are also cables that pass through the turbine building and portable generators off the turbine deck, but these SSCs are not assumed to fail. See Section 4.5.3 of Reference 38 for a further discussion on this topic.

16. Seismic SSC failures are assumed to be complete failures, in that the SSC fails to perform its function, or not. Degraded states of equipment (e.g., where only the equipment failure rates differ from the internal events model) for the period following the seismic initiator are not represented.

17. The assumed SSC seismic failure mode depends on the SSC type and whether the fragility applies to functional failure, structural failure, or impact by adjacent block wall or interaction failure. See Section 5.2 of Reference 38 for a further explanation. Relay chatter failure modes are a function of the specific relay and SSC control circuit itself. See Reference 37 for more discussion of relay chatter.

18. Inadvertent actuation of the Safety Injection (SI) signal or other Engineered Safety Feature Actuation System (ESFAS) functions may occur as a result of seismic failures in the actuation logic, or functional failure of the associated cabinets. However, the primary and secondary process racks and reactor protection racks all screen at high seismic capacity; i.e., greater HCLPF than 0.6g.

19. The alternate river water system is in a Category II building (Alternate Intake Structure [AISX]) and so preliminarily assigned a low seismic capacity, and thus the alternate service water pumps have a high probability of failure for even the lower seismic events. This conservatism is not expected to be significant because of the redundancy offered by the Category I river water system and the similarity of support systems both systems require for success.

20. The steam generator atmospheric relief valves and safety valves are highly redundant for steaming the steam generators. It is conservatively assumed that if they fail seismically, they would all fail to open; i.e., that the strong motion occurs before they are called on to open. This assumption is conservative because it would fail all steam generator cooling.

21. Seismic failures of buildings that are adjacent to the Reactor Containment Building were assumed to fail in a way that opened flow paths around the containment penetrations into each building. The flow area was assumed large enough to lead to a large early release

should a core-damage sequence also occur. The buildings applicable to this scenario are represented in Top Event ZL2 (Section 4.5.1 of Reference 38).

22. Seismic failures of the containment spray nozzles or discharge headers were assumed not to affect the transfer of water from the Refueling Water Storage Tank (RWST) into the containment. Such failures would affect the spray function but this function is not required to protect the containment.

23. Credit for the reactor coolant pump (RCP) shutdown seal has been taken since the Westinghouse Generation III RCP seal have been installed.

24. Correlation is assumed between SSCs assigned to the same EPRI capacity analysis category if in the same building and on the same floor. Credit for SSCs being arranged orthogonal to each other was not considered sufficient to break such correlations, except in the case for relays in the emergency switchgear (see section 5.6 of Reference 38 for further details).

25. Many other SSCs are seismically rugged, and therefore their seismic failure probabilities are unchanged from the internal events PRA; e.g., check valves, manual valves, cable trays, conduits, junction boxes, and local starter boards.

26. The existing Internal Events PRA meets the Capability Category II requirements of the ASME PRA Standard for PRA applications. Table 2-1 of Reference 38 lists the upgrade and update history of the Beaver Valley Unit 1 PRA through the years since it was first issued as an IPE PRA model in October of 1992.

27. The impacts of several Internal Events initiating events are conservatively assumed to occur simultaneously during a seismic initiating event.

28. Equipment failure data for random failures, test and maintenance unavailabilities, and plant configuration data are unchanged from the internal events PRA model. All seismic correlation sets and seismic initiating events are stored in the RISKMAN™ software (Reference 69) model data. The increasing SSC seismic failure probabilities with acceleration interval are computed from the fragility curves reported in Reference 41 within the Fragility Module of RISKMAN. The Am, βr, and βu parameters of the SSC seismic fragility curves are used to compute the acceleration interval dependent failure probabilities and then combined with other fragility curves which lead to the same plant impacts to generate the seismic pre-tree top event failure probabilities as appropriate. The seismic pre-tree accounts for the seismic SSC failures while the existing event trees account for the random SSC failures.

5.1.2 Seismic Event Trees for Large Early Release

The Level 2 PRA Notebook (Reference 64) documents the containment event trees used, the mapping of sequences from the Level 1 plant response into plant damage state bins, the assignment of sequences into release categories, and their categorization into large/small and early/late release states. The same containment event tree (CET) which models the containment response is used here for the SPRA. The LERF sequences are one such categorization of releases and are used for the SPRA calculation of LERF due to seismic events.

During SEL development SSCs related to LERF were identified to prevent inadvertent screening due to the large HCLPF screening cutoff for LERF. These SSCs include but not limited to the containment structure and any SSC that could affect the function of the containment pressure boundary, as well as SSCs that have a role in containment isolation failures.

The release categories assigned to LERF in the LERF analysis for internal events are presented in the PRA Notebook (Reference 64).

A discussion of seismic containment failures resulting in flow paths large enough, should core damage occur, to potentially lead to a large early release is provided in Section 4.5.1 of Reference 38. Seismically-induced large holes in the Reactor Containment Building are represented by Top Event L1 in the containment event tree, CET. Failure of Top Event L1 represents a large hole in the Reactor Containment Building prior to or at the time of Vessel Breach.

Regarding containment isolation failures on smaller lines, caused by seismic accelerations, see also the discussion of containment isolation failures in Section 4.5.2 and Table 4.5-1 of Reference 38. Seismic fragility assessments were performed on the containment isolation valves of the normally open lines of interest. Relay chatter analysis was also performed for the potential opening of isolation valves. These normally open lines, if failed, are modeled in GTRECIRC Top Event CI. CI failure represents openings too small to lead to a large early release and so do not impact the calculation of LERF at BVPS-1.

5.1.3 Relay Chatter Modeling

The investigation into SSCs susceptible to relay chatter during a seismic event is documented in Reference 37. Circuit analysis was performed for identified SSCs (MOVs, Pumps, pressure-operated relief valves (PORV), EDG Loading Circuits etc.). The evaluation of relay chatter considers chatter of not only relays, but also other non-relay contact devices as electro-mechanical contactors and motor starters (main and auxiliary contacts); circuit breakers (main and auxiliary contacts); manually-operated control switches; limit, torque, and position switches; and mechanical sensor switches including pressure, level, flow, and temperature switches, etc. This includes all the devices identified to be susceptible to high-frequency motion identified in EPRI Phase 2 testing (Reference 90). The circuit analysis evaluated the impact of the contact device (relay) on the SSC and screened out devices based on the following:

1. Relays that were located in non-seismically designed buildings were screened out as long as the components they were associated with were also located in a non-seismic building. The assumption is that both the component and relay fail when the building fails.
2. The relay impacts indication or annunciation only. Such relays will not physically alter the state of the SSCs. This also includes relays for post-accident monitoring.
3. The relay is not a lock-out relay and does not impact a seal-in or lock-out. Impacts to seal-in and lock-out relays are the principal concern in this study as these relays are the most likely to have an impact on PRA-related SSCs.
4. Due to the lack of mechanically moving parts, solid state relays are not prone to chatter.
5. Timing relays with settings greater than one second are not affected by chatter of upstream relays because they will be de-energized for sufficient time to reset the timing mechanism. However, a timing relay's output contacts may still chatter in response to seismic input.

Those relays that could not be screened had fragilities developed as described in ***Section 4.1.2*** of this submittal. The seismic failure of the relays that did not screen based on capacity was included in the SPRA. Each relay equipment group in the table below represents a correlation group of relays or contact devices that if chatter occurred (based on calculated fragility) would fail the top event(s) presented in the table below. The following ***Table 5-3*** lists the relays or contact devices that were modeled and their effect on the model if chatter were to occur.

TABLE 5-3
SUMMARY OF RELAY CHATTER CONSEQUENCES

Top Event	Top Event Name	Relay Equipment Groups	Relays in Group	Effect on Model if Seismic Top Event Failed
ZWC	All River water - Pumps REJs; HVAC DUCTS	EQ113	Contactors in MCCs BV-MCC-1-E1(E2) for MOVs: 1RW-102A2(B1,C1,C2)	If failed fails WA and WB
ZM5	Contactors in MCCs BV-MCC-1-E5(E6) Chatter	EQ116	Contactors in BV-MCC-1-E5 & BV-MCC-1-E6 for MOVs: 1FW-151B(D,F) 1MS-105 1RC-535(536,537) 1RS-155A(B) 1RS-156A(B) 1RW-104A(B,C,D) 1RW-105A(B,C,D)	Fails AF, A3, RA, RB, RS. OD and PR are not failed but part of the fault tree is affected and thus certain split fractions are selected
ZRS	Recirculation Spray – pumps & HXs & header	EQ115	Circuit Breakers for 1RS-P-1A (B)	Fails RS; RA; and RB
ZQS	QSS - pumps	EQ114	Circuit Breakers for 1QS-P-1A (B)	Fails QA and QB and when combined with ZLP failed then no branch at ZMO
ZR2	RELAY CHATTER -PUMPS DF Bus	EQ87	COM-5 Relays for pumps: BV-1WR-P-1B(C) BV-1CH-P-1B(C) BV-1SI-P-1B BV-1RS-P-2B BV1CC-P-1B(C) BV-1FW-P-3B	Fails WB, LB, and RB. Uses the split fractions for only A train failed for tops CC:HH;HC;LC;RS. But fails these tops for a combined failure with ZR4
ZR4	RELAY CHATTER -PUMPS AE Bus	EQ99	COM-5 Relays for pumps: BV-1WR-P-1A(C) BV-1CH-P-1A(C) BV-1SI-P-1A BV-1RS-P-2A BV1CC-P-1A(C) BV-1FW-P-3A	Fails WA, LA, and RA. Uses the split fractions for only A train failed for tops CC:HH;HC;LC;RS. But fails these tops for a combined failure with ZR4

5.1.4 Correlation of Fragilities

SSCs not screened by potential impact on the plant were then assigned to correlation sets in part by their seismic capacities. It is important to account for dependencies between the probabilities of seismic failure modes, as appropriate. Past SPRA's have assumed that all identical and redundant equipment located in the same or at least seismically similar response locations, are 100 percent correlated, while assuming that equipment which is identical, but not redundant, (i.e., perform their functions in series) are uncorrelated. Here, by 100 percent correlated we mean that if one equipment item in the redundant set fails seismically, all others in that redundant set are also assumed to fail and via the same failure mode. This is a much stronger linkage than simply saying their failure probabilities are the same yet the failure probabilities themselves are independent. This 100 percent correlation approach conservatively minimizes the advantages of redundancy; partial correlation is not modeled.

The approach to defining correlation groups in this study is explained below.

All SSCs on the SEL have been screened as seismically rugged, are judged not to impact the PRA model, or have had their seismic capacities assessed. Those assessed have been assigned to one of the EPRI seismic analysis categories as an initial step in computing seismic equipment fragilities. These categories were further broken down into analysis groups which contain the SSCs sufficiently similar in anchorages as to be expected to all be evaluated in roughly the same way. For example, for BVPS-1 the equipment assigned to the EPRI Category 21 for tanks and heat exchangers was further divided into nine analysis groups due to perceived differences in the analysis needed to assess their seismic capacities.

A further consideration is in the final assessment of equipment capacities. In this study the equipment's HCLPF is used as a measure of equipment capacity, although it is recognized that the capacity is defined by the entire fragility curve, including its parameters for median capacity and variability assigned. The HCLPF assigned is a function of many things, including the equipment type, seismic design classification and the exact SSC location within the building.

The general approach to correlating SSCs into correlation groups was to group those SSCs that of the same equipment types, have roughly the same seismic capacity, and subject to the same seismic accelerations. Reasons for not grouping such SSCs are as follows:

1. SSCs in different EPRI categories are assigned to different correlation groups.
2. SSCs in different buildings are assigned to different correlation groups.
3. SSCs on different floors of the same building are assigned to different correlation groups.
4. SSCs which seismic capacities are evaluated differently according to their different analysis groups are assigned to different correlation groups, though sometimes the analysis groups are sufficiently similar that they still should be grouped.

The approach to correlation was first to divide the full list of equipment into partial lists of nearly identical equipment. The lists of all equipment in the same EPRI category were separated out, one category at a time. If multiple types of equipment are assigned to the same EPRI category (for example air-operated valves (AOV) and relief valves are assigned to the same EPRI

Category 7), then the list reviewed was further broken up by types of equipment within a given EPRI Category.

The next step was to sort the list of equipment within the EPRI category by capacity as measured by their assessed HCLPFs.

Correlation groups were then assigned based primarily on similarity of the assessed HCLPFs. While they need not be identical, the grouping into correlation sets was only performed for those SSCs with nearly the same HCLPF; i.e., within say 0.05g of each other. Grouping equipment with substantially different HCLPFs can be problematical, because then it is unclear which HCLPF to assign to all the SSCs within the correlation set. For this study, the lowest HCLPF within the correlation set was assigned to all SSCs within the set, though most often equipment assigned to the same correlation group had identical HCLPFs. SSCs of the same equipment type with HCLPFs that differed by more than 0.05g were generally found to be designed to different seismic design classifications, located in different buildings, were in notably different elevations within the same building, or belonged to a different analysis group of the same EPRI category, indicating that their anchorage design maybe different.

Exceptions to the above rules for assigning SSCs to correlation groups were made for this study and are documented in Reference 38. Generally these assumptions reflect differences in the depth for fragility analysis for each SSC and the relative importance of the SSCs. The correlation groups defined are presented in Table 5.4-1 of Reference 38 along with the SSCs assigned to each. Nearly 450 SSCs are explicitly grouped into 133 correlation groups. Since the SSCs may have a slightly different capacity than that assigned to the entire correlation set, the individual SSC HCLPFs are also listed in the table. Note that these HCLPFs are for the minimum HCLPF values for the different failure modes of the same SSC; i.e., from among the failure modes of functionality, structural/anchorage, relay chatter, block wall impacts, or interaction failures.

5.1.5 Human Reliability Analysis

The list of post-initiator human actions for the internal events model was analyzed for modification due to seismic affects. Some human failure events (HFE) were excluded from the analysis due to not being associated with the sequence models used to represent seismic initiators; e.g., HFEs for SGTR initiators.

Every post-initiator HFE retained in the SPRA sequence models was evaluated for the impacts of seismic events. The degree of impact was assumed dependent on the seismic acceleration level. At very high accelerations, the human error probabilities (HEP) were set to 1.0. The seismic impacts on every post-initiator HFE in the SPRA sequence models is accounted for by the HFE specific, performance shaping factors and selected minimal values that increase with acceleration as a function of plant damage state. The adjusted HFEs use the internal events name with the suffix of "Sn" where n ranges from 1 to 4; i.e., four separate seismic acceleration ranges were evaluated for varying seismic impacts, but in SEIS4, all post-trip actions are set to failed. Further discussion of the modeling changes made to account for acceleration dependent HEPs is provided in Section 6.0 of Reference 38. A summary of the SPRA HRA HFE HEP Evaluation Process is provided in ***Table 5-4*** below.

TABLE 5-4
SPRA HFE HEP EVALUATION PROCESS SUMMARY

SEISMIC GROUP	APPROXIMATE G LEVEL	SEISMIC INITIATING EVENT(S)	CORRESPONDS TO FOR BV1	AFFECTS CR HFE	AFFECTS FIELD HFE	COMMENTS
SEIS1	0-0.15	G1	SSE (and slightly over)	• Add 2 min to Tdelay- • no other affect	• Add 2 min to Tdelay- • no other affect	Plant is designed for SSE- should be little effect; 2 minutes to account for initial shock. Note, that if adding time delays for SPRA also increases the EPRI recommended floor values of dependency, this updated floor value for dependency is applied in the cognitive and execution recovery portions of the HEP evaluation (this is applied in all cases where the EPRI recommended dependency level has changed, including for SEIS1, SEIS2, and SEIS3 evaluations)
SEIS2	0.15-0.8	G2 - G7 (ZO3=S and ZO4=S)	Accelerations greater than the SSE in which control room indication is not lost and the control room ceiling is still intact.	• Add 2 minute to Tdelay and • increase cognitive workload and • execution stress level to high • If HCR/ORE Cognition, increase CP level to UB	• Add 2 minute to Tdelay and • increase cognitive workload and execution stress level to high and • increase Texe to 2x • If HCR/ORE Cognition, increase CP level to UB	Although control indication is still available seismic events greater than the SSE are likely to cause additional failures that would increase cognitive workload and stress as well as execution time

TABLE 5-4
SPRA HFE HEP EVALUATION PROCESS SUMMARY
(CONTINUED)

SEISMIC GROUP	APPROXIMATE G LEVEL	SEISMIC INITIATING EVENT(S)	CORRESPONDS TO FOR BV1	AFFECTS CR HFE	AFFECTS FIELD HFE	COMMENTS
SEIS3	0.15-0.8	G2 - G7 (ZO3=F and ZO4=S)	Accelerations greater than the SSE in which control room indication is lost but the control room ceiling is still intact.	• Add 15 minute to Tdelay and • increase cognitive workload and • execution stress level to high • use "monitored, not alarmed" for pcb, • no ERF recovery credit • If HCR/ORE Cognition, increase CP level to UB	• Add 15 minute to Tdelay and • increase cognitive workload and • execution stress level to high; • use "monitored, not alarmed" for pcb, • no ERF recovery credit and • increase Texe to 4x • If HCR/ORE Cognition, increase CP level to UB	When controls are being lost in the control room; there should be a step change. Difficult to navigate to work site; many components already failed. **USE FLOOR OF 1E-02 FOR INDIVIDUAL HFEs**
SEIS4	Greater than 0.8	G8, G9, and G10	High Accelerations	Fail 1.0	Fail 1.0	Most CAT 1 buildings fail above 1.0g
	All	G01 – G10 (Z04=F, PT=TOX, or ZL1=F)	Catastrophic; failure of the control room ceiling, failures of SSCs leading to direct core-damage, or toxic failure of the propane tank farm.	Fail 1.0	Fail 1.0	CR ceiling fails at about 0.7056g

Human Failure Events were also developed for the FLEX mitigation actions. These are not specific to the seismic PRA as they are designed for an extended loss of offsite power scenario, and specifically account for high levels of plant damage and operator stress. The FLEX operator actions were developed using the same methodology as other internal events HFEs. Execution step durations were obtained from the timing validation study performed by BVPS. These actions are failed in the seismic model for the "SEIS4" or high acceleration group identified above.

The use of the same method from the internal events model for HRA dependency analysis is valid for the SPRA HRA. The SPRA HRA Notebook (Reference 36) discusses the method used to assess HFE dependency. The SPRA Quantification Notebook also has details of how the HRA dependency analysis was performed for the SPRA (Reference 17). The FENOC HRA Dependency Database (Reference 70) is used to determine the level of dependency between HFE Pairs assigned to the same HRA seismic interval since only such pairs can appear in the same accident sequence; i.e., SEIS1, SEIS2, and SEIS3. These pairs with other than zero dependence are then examined individually to see if the dependence need be included in the accident sequence model. Section 4.2 of Reference 17 discusses the HRA dependency analysis further.

Pre-initiator actions are not affected by seismic events and so were not changed from the internal events PRA model.

5.1.6 Seismic-Induced Floods

The evaluation of seismic-induced floods was a compilation of three activities. First, the internal flooding PRA, Reference 27, was utilized to provide a risk-based screening of flood-significant scenarios. The second activity was the use of the walkdown team to identify flood sources in and around components that were on the SEL; this is documented in the Seismic Walkdown of BVPS-1, Reference 40. The third activity was to review the tanks not on the SEL and the "wet" fire suppression system and do a walk-by of the components to determine if the assets would screen; this is documented in the SEL Notebook, Reference 32.

As discussed in Section 3.3.8 of Reference 40, the piping evaluation was risk informed. The systems of interest and pipe segments selected were those that had the greatest risk contributions as evaluated in the Internal Flood PRA.

In addition to those pipe segments identified in Section 3.3.8 of Reference 40, Table 3-4 of Reference 40, identifies three additional pipe segments that merited additional specific walk downs. This list was derived from a list of important flood scenarios minus those pipe segments that had previously been walked down. The list of important flood scenarios is given in Table 5.5-1 of Reference 38.

During the plant walk downs, piping in general, and non-seismic piping in particular were examined to see if there were any unique vulnerabilities in proximity to any of the SSCs examined; see Reference 27. A summary of specific seismic-induced flooding interactions is provided in Section 3.3.8 of Reference 40. Appendix B of Reference 40, presents pictures and the walkdown team's conclusions for the piping segments called out as having the highest conditional probability of core damage given a pipe break occurs.

Table 5-5 presents risk significant flood scenarios.

TABLE 5-5
RISK SIGNIFICANT FLOOD SCENARIOS

SSC ID; Scenario Designator	SSC Description	Building Names	Eleva-tion	Seismic Category	SAP System Names	Flood Scenario CCDP
PA1C-FWLP-2	Volume Control Tank Cubicle; 1.5 inch diameter,5 feet long, not insulated CVCS pipe; 8700-RP-10D -	Primary Auxiliary Building	752'	Cat 1	Chemical and Volume Control System	3.68E-04
PA1E-FWLP-4	General Area E; 1.5 inch diameter, 20 feet long, not insulated, SI pipe	Primary Auxiliary Building	735'	Cat 1	Safety Injection System	1.35E-04
QP1-FWLP-2	Quench Spray Pumps Room; 10 inch diameter, 50 feet long, not insulated, QS pipe	Safeguards Building	735'	Cat 1	Containment Depressurization System (Recirculation Spray and Quench Spray)	1.35E-04
MS1-SE-1	Main Steam Valve Area; 30 inch diameter, 75 feet long, insulated, main steam pipe	Safeguards Building	756'	Cat 1	Main Steam System	2.74E-05
MS1-SE-2	Main Steam Valve Area; 18 inch diameter, 90 feet long, insulated, FW pipe	Safeguards Building	756'	Cat 1	Steam Generator Feedwater System	2.74E-05
PA1E-HI-1	Primary Auxiliary Building General Area E - Component Cooling Water Pump Area; 18 inch diameter RW pipe, 10.75 feet not insulated, Inlet CCR HX, Downstream of Inlet Valves	Primary Auxiliary Building	735'-6"	Cat 1	River Water System	2.11E-01
PA1E-SP-3D	Area outside A&B Degasifier Cubicles; 4 inch diameter CCR pipe, 20 feet not insulated	Primary Auxiliary Building	735'	Cat 1	Reactor Plant Component and Neutron Tank Cooling Water Systems	2.11E-01
AF1-FWLP-1	Auxiliary Feed Water Pumps Room; 6 inch diameter, 12 feet long, not insulated water treatment system piping	Safeguards Building	735'	Cat 2	Water Treating System	1.73E-03
AF1-FWMP-2	Auxiliary Feed Water Pumps Room; 7 inch diameter, 96 feet long, not insulated Aux Feedwater pipe	Safeguards Building	735'	Cat 1	Steam Generator Feedwater System	1.73E-03
PA1E-FWMP-1	General Area E; 6 inch diameter, 265 feet long, not insulated, fire protection system piping	Primary Auxiliary Building	735'	Cat 1	Fire Protection System	5.66E-04

5.1.7 Risk Significant Flood Scenarios

As a supplement to the SSCs in the internal events PRA, a list of all tanks and coolers at the plant was obtained for review for potential seismic-induced flood sources. This list was reduced by excluding those tanks in plant rooms that contain no SSCs on the SPRA SEL, and to eliminate duplicates that are already on the SPRA SEL. The reduced list of potential flood sources is also shown as Table 3-6 in Reference 32.

The reduced list of potential sources was then filtered by building and those located in the turbine building were also then excluded. For the SPRA only the cross-tie cables and the portable generators are credited. Both of these had no flooding susceptibility identified in the internal flooding PRA. Also, failure of flooding sources in the TRBB do not propagate to adjoining buildings.

To ensure that no important tanks were missed, the SPRA SEL list of tanks, coolers/heat exchangers, and pumps (which have coolers) was reviewed. Those not already on the list were added if the tanks and coolers were not located in the yard or containment, and contained liquids rather than air.

The walkdowns performed by the Seismic Review Team screened these from further consideration either due to their seismic ruggedness, presence of dikes around the tanks, or lack of proximity to SEL components. All tanks were screened based on either: information provided in the internal events flooding analysis, or based on no impact to PRA equipment in the flood area, or too small of a flood source to cause an impact. The small coolers also were screened from either of these screening criteria.

The flood sources from tanks and heat exchangers, although technically screened, were sampled and walked down to validate the assumptions made for their screening. These include the fire protection engine cooler on the diesel driven pump and the spent fuel pool heat exchangers as examples.

No potential flooding sources have been identified for inclusion in the BVPS-1 seismic model.

5.1.8 Seismic-Induced Fires

Appendix A in Reference 38 contains a white paper on the subject of seismic-induced fires. The presentation describes ways that seismic-induced fires may be screened, both qualitatively and quantitatively from further consideration. The flow chart presented at the end of Appendix A in Reference 38, summarizes the variety of ways that screening can be performed on a fire compartment by compartment basis.

The following are some key conclusions from the suggested approach in Appendix A in Reference 38:

1. The list of equipment of interest as potential fire sources caused by seismic events are:
    a. Tanks, Bottles, and Piping (including turbine-generator, auxiliary boiler) That Contain Hydrogen, Propane, and any Other Flammable Gases
    b. Above-Ground Tanks and Piping That Contain Diesel Fuel Oil

c. Tanks, Equipment, and Piping That Contain Lubricating Oil

- Turbine-Generator
- Turbine Lube Oil Storage Tank
- Oil-Filled Transformers
- Pumps (especially large pumps)
- Compressors
- Piping

d. Equipment with Electrical Wire or Bus Bar Connections at 480V and Above

- Pumps
- Oil-Filled Transformers
- Compressors
- Switchgears/Buses/MCCs
- Others (e.g., other applicable NUREG/CR-6850 fire source bins from Fire PRA that are unique and significant for specific plants)

2. Seismic-induced fires are believed possible only if structural failure of the SSC occurs; i.e., we neglect the functional failure limit if it is lower.

3. Based on data from other industries, the conditional probability of fire ignition given seismic failure of a potential seismic-induced fire source is bounded by 0.1. An individual seismic-induced fire frequency leading to core damage for a single SSC of 1E-7 per year is assumed as sufficiently small as to be neglected. Due to frequency overlap between the potential seismic-induced fire and other contributions to core damage, a single, SSC seismic-induced fire frequency of 5E-7 per year is sufficiently small as to be negligible.

For this study of Beaver Valley, we adopt the above methodology conclusions and apply the qualitative and quantitative screening of potential seismic-induced fire sources, including the use of walkdown observations to eliminate seismic-induced fires from inclusion in the SPRA logic models. The case for this screening is provided below.

Table 5.5-2 of Reference 38 presents the failure frequencies of SSCs with typical HCLPFs ranging from.1g to 2.0g. The total frequency column was obtained by summing the convolution of the Beaver Valley mean seismic hazard curve over all seismic intervals. The frequency of seismic failure of 1E-7 per year corresponds to an SSC HCLPF of just greater than 1.0g.

However, this acceleration level has not yet accounted for the conditional probability of ignition given the SSC fails, or of the potential overlap of seismic-induced fires with other contributors to core damage. At Beaver Valley, the conditional core-damage probability for accelerations of 0.6g and higher is close to 1.0. Therefore, seismic-induced fires at frequencies greater than 0.6g cannot add significantly to the CDF total. The HCLPF acceleration corresponding to a failure frequency of 1E-7 per year, only from accelerations less than 0.6g is then between 0.5g and 55g. This is an approximate approach, as other contributors to core damage at accelerations less than 0.5g do occur and so there is some potential overlap at lower accelerations that is not credited.

An ignition probability of 0.1 reduces the frequency of SSC failures to just those that also ignite, resulting in a fire. A corresponding HCLPF value just more than 0.35g would result in a potential fire source adding approximately 1E-7 per year to the existing seismic CDF. We observe that this acceleration is selected conservatively both because of the potential for frequency overlap at accelerations less than 0.6g, and because it is implicitly assumed by this screening calculation that any seismic-induced fire leads to core damage. Further, the results for the unconditional seismic-induced fire frequencies presented in Reference 38 do not yet include a scaling factor on the hazard exceedance curves to account for the plant availability factor. To do so would provide us additional margin. We therefore use 0.35g for an SSC HCLPF as the quantitative screening criterion for excluding potential seismic-induced fires.

Table 5-6 (reproduced from Reference 71) provides a list of the top 25 fire scenarios from the BVPS-1 fire PRA. Out of these 25 scenarios, CR-4 and CR-3 fire areas were the dominant contributors and those areas were chosen to have a specific seismically-induced fire walkdown.

TABLE 5-6
RISK CONTRIBUTING PLANT LOCATIONS FROM THE BEAVER VALLEY UNIT 1 FIRE PRA (REFERENCE 71)

SCENARIO ID	SCENARIO DESCRIPTION	EXPANDED DESCRIPTION
FRC134	738NW	1-RC-1 Reactor Containment, Northwest section of 738' elevation
FCR461	SSW-CMP FDS2/3/4/6/7/8/10/11/14/15	1-CR-4 Process Instrument Rm, source SSW-CMP; fire grows and affects external targets
FCR495	RK-PRI-PROC (5,8-13,19,21-22,24-29) FDS2/3/6/7/10/14	1-CR-4, source RK-PRI-PROC sections 5, 8-13,19,21-22,24-29; fire grows and affects external targets
FNS117	480VUS-1-3-E FDS4/9	1-NS-1 Normal Switchgear, source bus 480VUS-1-3E; fire grows and affects external targets
FES142	TRANS-1-8N FDS2/5	1-ES-1 1AE Emergency Switchgear, source transformer TRANS-1-8N; fire grows and affects external targets
FRC133	738N	1-RC-1, North section of 738' elevation
FRH1	3-RH-1 WHOLE ROOM	3-RH-1 Relay House in the Switchyard; whole compartment assumed burned from any of the defined sources
FCR44A	RK-REAC-PROT (A) FDS0	1-CR-4, source RK-REAC-PROT (A); fire contained within cabinet
FCR48A	RK-REAC-PROT (B) FDS0	1-CR-4, source RK-REAC-PROT (B); fire contained within cabinet
FCR499	RK-PRI-PROC-1,2,3,6,7,4,20,23,14,15,16,17,18 120VAC PRIMARY FDS2/3/6/7/10/14	1-CR-4, source RK-PRI-PROC sections 1,2,3,6,7,4,20,23,14,15,16,17,18; fire grows and affects external targets
FMCA07	1-TB-1 3-CR-1	Multi-compartment scenario; fire engulfs Turbine Building then spreads to engulf control room
FCR432	120VAC SECONDARY (BDHKJLM) FDS2/3/4/6/7/8/10/11/14/15	1-CR-4, source 120VAC Secondary Process Racks BDHKJLM; fire grows and affects external targets
FCR49A	RK-REAC-PROT (B) FDS2/3/6/7/10/14	1-CR-4, source RK-REAC-PROT (B); fire grows and affects external targets
FCR45A	RK-REAC-PROT (A) FDS2/3/6/7/10/14	1-CR-4, source RK-REAC-PROT (A); fire grows and affects external targets
FCR4A6	COMMUNICATIONS BATTERY CHARGER 48B FDS3/7/11/15	1-CR-4, source Communications Battery Charger 48B; fire grows and affects external targets
FCR427	120VAC SECONDARY (ACPEFG) FDS2/3/6/7/10/14	1-CR-4, source 120VAC Secondary Process Racks ACPEFG; fire grows and affects external targets
FER1	3-ER-1 WHOLE ROOM	3-ER-1 ERF Substation; whole compartment assumed burned from any of the defined sources
FCR301	EDG RACK (6 VERT SECT) FDS2/5	1-CR-3 Relay Room, source EDG rack; fire grows and affects external targets
FNS122	480VUS-1-3-E (HEAF) FDS3/7	1-NS-1, source bus 480VUS-1-3E High Energy Arcing Fault; fire affects external targets
FQP105	TS#3 (AF-1 ROOM)	1-QP-1 Quench Spray Pump & AFW Pump room, source transient scenario #3 (see fire modeling)

TABLE 5-6
RISK CONTRIBUTING PLANT LOCATIONS FROM THE BEAVER VALLEY UNIT 1 FIRE PRA (REFERENCE 71)
(CONTINUED)

SCENARIO ID	SCENARIO DESCRIPTION	EXPANDED DESCRIPTION
FCR494	RK-PRI-PROC (5,8-13,19, 21-22,24-29) FDS0	1-CR-4, source RK-PRI-PROC sections 5, 8-13,19,21-22,24-29; fire contained within cabinet
FCR435	COMPUTER CABINETS RK-CMP-DIN-4, TERM-2, IPC-CAB-05, RK-CMP-TERM-1 FDS2/3/6/7/10/14	1-CR-4, source computer cabinets RK-CMP-DIN-4, TERM-2, IPC-CAB-05, RK-CMP-TERM-1; fire grows and affects external targets
FCR440	DC SWBD1-5 FDS2/3/4/6/7/8/10/11/14/15	1-CR-4, source DC switchboard 1-5; fire grows and affects external targets
FNS185	TRANS-1-4G FDS5/10	1-NS-1, source transformer TRANS-1-4G; fire grows and affects external targets
FCR42A	RK-PRI-PROC-1,2,3,6,7,4,20,23,14,15,16,17,18 120VAC PRIMARY FDS4/8/12/16	1-CR-4, source RK-PRI-PROC sections 1,2,3,6,7,4,20,23,14,15,16,17,18; fire grows and affects external targets

With the quantitative screening criterion established, the potential fire sources previously screened in qualitatively for assessment, according to the arguments of Appendix A in Reference 38, were addressed.

1. Tanks, bottles, and piping (including turbine-generator, auxiliary boiler) that contain hydrogen, propane, and any other flammable gases.

   The flammable gases in the nuclear plant consists basically of hydrogen. It is used as a cover gas on the generator. The gas for the generator is in the yard well away from the plant structure itself and the generator is in the turbine building. Potential sources in the turbine building are screened because no credit is taken for SSCs within the turbine building for seismic events, with the exception for the cross-tie cables and portable generators which are actually located off the turbine deck and are separated from the turbine deck by a block wall.

   Hydrogen used for chemistry analysis was screened based on the small quantity involved and the lack of risk significant equipment in the vicinity.

   Similarly, we screened potential sources in the yard, since even if they seismically fail, they will not impact other SSCs that are credited.

2. Above-Ground tanks and piping that contain diesel fuel oil.

3. Tanks, equipment, and piping that contain lubricating oil.

- Turbine-Generator
- Turbine Lube Oil Storage Tank
- Oil-Filled Transformers
- Pumps (Especially Large Pumps)
- Compressors
- Piping

Table 5-7 below lists potential fire ignition sources at BVPS-1 not included in the SEL. These items were all part of the walkdown and evaluated for their potential to become a seismically-induced fire. The oil and grease sources on the list were part of the larger component and all screened with a HCLPF of greater than 0.3.

TABLE 5-7
HYDROGEN AND FLAMMABLE LIQUID IGNITION SOURCES

Ignition Source	Fire Compart ment	Building	Elevation	Area/ Room	Plant Area	Ignition Source ID	Hydrogen or Flammable Liquid Loadings
Unit 1	1-H-1	Yard	N/A	N/A	Bulk H2 storage tanks in BV1 yard area, above ground	Bulk Hydrogen Storage Tanks	Large
	Notes	Yard	N/A	N/A	5/8" supply line (stainless steel) enters 1-TB-1 at 683' and NE corner of TB from supply tanks in yard, goes over 1-TO-1 at 8 feet, second red-painted hydrogen line for T/G and excitor next to it; run to H2 supply manifold at 692', h2 supply line runs alongside of TB and exits to Aux building before reaching other corner on long side,	Misc. Hydrogen Piping	Large
	Notes	Yard	N/A	N/A	hydrogen line runs outside 7-10' above floor and rail car door onto service building roof outside near 3-CR-1 (SRVB,735')	Misc. Hydrogen Piping	Large
	Notes	Yard	N/A	N/A	H2 line passes into 1-PA-1A of auxiliary building at 768', then runs along wall opposite of 1-PA-1G, then goes down through floor while H2 vent line goes up and out thru ceiling	Misc. Hydrogen Piping	Large
	Notes	Yard	N/A	N/A	H2 line passes down to 752' elevation of auxiliary building, runs close to ceiling past 1-PA-1G chase until down to H2 manifold at eye level just outside VCT cubicle (at AXLB 752', walkdown notes say near 1PCV-CH-119 which is not in SEL)	Misc. Hydrogen Piping	Large
Unit 1	1-PA-1A	AXLB	768		auxiliary building, 768'7" (see above walkdown notes)	Misc. Hydrogen Piping	Large
Unit 1	1-PA-1C	AXLB	752		auxiliary building, 752'6" (see above walkdown notes)	Misc. Hydrogen Piping	Large
unit 1	1-DG-1	DGBX	735	DG Room Train A	1-EE-EG-1 (DGBX, 735', DG ROOM TRAIN A)	Diesel Generator #1-1	470 gal. lube oil, 1100 fuel oil
Unit 1	1-DG-2	DGBX	735	DG Room Train B	1-EE-EG-2 (DGBX, 735', DG ROOM TRAIN B)	Diesel Generator #1-2	470 gal. lube oil, 1100 fuel oil
Unit 1	1-IS-4	INTS	705	Cubicle 4 or D	INTAKE STRUCTURE CUBICLE 4 (or D), 705'	pumps	450 gal. FUEL OIL, 27 LUBE OIL
common	3-IS-6	INTS	705	General Area	INTAKE STRUCTURE GENERAL AREA, 705'	pumps and oxy-acetylene	165 gal. FUEL OIL, 16 LUBE OIL, OXY-ACETYLENE WELDING CART
Unit 1	1-PA-1E	AXLB	735	Charging pump cubicles	AUXILIARY BUILDING, 735' CHARGING PUMP CUBICLES PA-1F,1G,1H (SUBAREA PA-1G)	pumps	60 gal. LUBE OIL FOR EACH PUMP

Piping containing lubricating oil and hydraulic oil are mostly in the turbine building. The SSCs within the turbine building are not credited in the SPRA (one exception is for the cross tie cables and the portable generators because their location would not be affected) and so such pipes in the turbine building are screened. All pipes examined in the SPRA were found to have high capacity, and so were screened from further consideration of seismic-induced fires.

4. Equipment with electrical wire or bus bar connections at 480V and above.

Regarding switchgear, buses, and MCCs, a walkdown was performed to examine these equipment items focusing on the potential for their structural failures leading to a significant seismic-induced fire.

Both the Division 1 and Division 2 switchgear rooms were walked down due to these zones being significant contributors to CDF in the Fire PRA and because they could possibly have a high energy arcing fault.

Seismic-induced fire would require both overturning of switchgear and severing of top lines. Top conduits are rigidly braced to the wall. No potential interactions were observed that would puncture/sever top conduits, so the most likely failure mode is judged to be structural/anchorage failure resulting in switchgear overturning and severing of conduits. Preliminary calculations determined a HCLPF >0.30g for structural (anchorage) failure that would be required to initiate overturning. Those preliminary calculations conservatively do not credit the restraint added by the top conduit bracing to prevent overturning, so the actual structural capacity of the component is higher. The transformers in the area are dry type.

The high voltage switchgear in both rooms were all well supported and the potential for any differential movement between the switchgear and the conduits that enter and exit appeared to be minimal thus reducing any potential high energy arcing fault.

480V transformers are used throughout the plant to step down power to a 120vac lighting panel. These were determined to be seismically robust.

No potential seismically-induced fire sources were identified for inclusion in the SPRA. This conclusion is further supported by the review documented in Reference 72.

5.2 SPRA Plant Seismic Logic Model Technical Adequacy

The BVPS-1 SPRA seismic plant response methodology and analysis were subjected to an independent peer review against the pertinent requirements in the PRA Standard (Reference 4).

The peer review assessment, and subsequent disposition of peer review findings, is described in ***Appendix A***, and establishes that the BVPS-1 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 BVPS-1 SPRA, the following approach was used to quantify the seismic plant response model and determine seismic CDF and LERF:

The computer codes used by the BVPS-1 PRA are available from ABSG Consulting Inc. (ABS Consulting) which is the developer of the RISKMAN™ software. Technical support and quality assurance are provided by ABS Consulting. The software is classified as Category B software per the FENOC Administrative Program for Computer Related Activities (Reference 73), and has been site accepted per that program.

5.3.1.1 RISKMAN™ Software

RISKMAN 14.3 was used in the creation and maintenance of both the internal events PRA and in this SPRA. Version 14.3 was also used in the development or the Interval Events PRA. Version 14.3 was used for the SPRA and is also now used to maintain the internal events PRA models. The features and code limitations of RISKMAN are described in Reference 69 and its companion manuals for each of the main modules.

5.3.2 SPRA Model and Quantification Assumptions

The following assumptions were made as part of the seismic PRA quantification:

The quantification of CDF and LERF sequences is performed by a large, linked-event tree model in which the seismic acceleration intervals are evaluated successively and then the computed frequencies added. The seismic impacts on types of SSCs represented in the SPRA model are limited to those identified in Tables 5.2 1 of Reference 38.

1. Screening criteria for the need to include SSCs within the SPRA model were set at 0.6g HCLPF for all SSCs and up to 2.0g for SSCs related to LERF.
2. In the base-case SPRA model, the assignment of human error probabilities for each HFE is dependent on the selected component failures that impact operator response and the associated acceleration range for which the human error probability (HEP) is being evaluated (see Reference 36).
3. The base-case accident sequence quantification cutoff used was 1x10-14 per year, for both CDF and LERF. The sensitivity analyses were performed using a sequence frequency cutoff of 1x10-14 per year. See Section 4.3 of Reference 38 for a discussion of CDF and LERF convergence.

5.4 SCDF Results

The seismic PRA performed for BVPS-1 shows that the point estimate mean seismic CDF is $1.30\text{x}10^{-5}$. A discussion of the mean SCDF with uncertainty distribution reflecting the uncertainties in the hazard, fragilities, and model data is presented in ***Section 5.6***. Important contributors are discussed in the following paragraphs.

The top SCDF accident sequences are documented in the SPRA quantification (Reference 17). These are briefly summarized in ***Table 5-8***.

TABLE 5-8
SUMMARY OF TOP SCDF ACCIDENT SEQUENCES

RANK	INITIATING EVENT	IE FREQUENCY	SCDF/YR	PERCENT OF SCDF	SEQUENCE PROGRESSION DESCRIPTION
1	G09 1.0-2.0g	1.07E-06	1.52E-07	1.17%	This earthquake directly causes core-damage and large early release, without potential for mitigation, due to structural failure of one or more of the Reactor Containment Building, Safeguards Building, Main Steam & Cable Vault Building, or the Steam Generators.
2	G10 2.0-4.99g	8.59E-08	7.60E-08	0.58%	This earthquake directly causes core-damage and large early release, without potential for mitigation, due to structural failure of one or more of the Reactor Containment Building, Safeguards Building, Main Steam & Cable Vault Building, or the Steam Generators.
3	G03 0.25-0.4g	3.31E-05	4.01E-08	0.31%	This seismic event causes a loss of offsite grid, and structurally destroys the RWST. A very small LOCA is caused by seismic failure of the instrument lines under the reactor vessel; the break size is not enough to depressurize the Reactor Coolant System (RCS), so inventory is expelled at high pressure, resulting in the equivalent of a small LOCA. This LOCA leads to a Containment Isolation A (CIA) signal that swaps the RCS inventory injection source from the Volume Control Tank to the RWST. However, with the RWST seismically failed during the earthquake, high-head and low-head injection pumps have no available inventory to inject. Without RCS makeup capability, the core uncovers and core damage occurs.
4	G03 0.25-0.4g	3.31E-05	3.97E-08	0.30%	This seismic event causes a loss of offsite grid and also seismically fails the emergency AC buses, inducing a station blackout. A very small LOCA is caused by seismic failure of the instrument lines under the reactor vessel; the break size is not enough to depressurize the RCS, so inventory is expelled at high pressure, resulting in the equivalent of a small LOCA. Inventory cannot be replenished since the necessary pumps are failed by the SBO. Electric power recovery is not credited since it is assumed that the seismic damage to the emergency buses cannot be repaired. The core uncovers and results in core damage.

TABLE 5-8
SUMMARY OF TOP SCDF ACCIDENT SEQUENCES
(CONTINUED)

RANK	INITIATING EVENT	IE FREQUENCY	SCDF/YR	PERCENT OF SCDF	SEQUENCE PROGRESSION DESCRIPTION
5	G03 0.25-0.4g	3.31E-05	3.69E-08	0.28%	This seismic event fails offsite grid and seismically fails the river water pump trains, which fails cooling water to the diesel generators. The diesels will either overheat and fail very early into the sequence, or operators will secure them. However, electric power recovery for this sequence is not credited since it is assumed that the earthquake damage to the river water pump trains cannot be repaired. This puts the site in a station blackout. A very small LOCA is caused by seismic failure of the instrument lines under the reactor vessel; the break size is not enough to depressurize the RCS, so inventory is expelled at high pressure, resulting in the equivalent of a small LOCA. Inventory cannot be replenished since the high-head injection pumps are failed by the SBO, and would have also failed due to the lack of river water to cool the pumps. The core uncovers and results in core damage.
6	G05 0.5-0.6g	3.99E-06	3.06E-08	0.23%	This seismic event causes a loss of offsite grid and also seismically fails the emergency AC buses, inducing a station blackout. A very small LOCA is caused by seismic failure of the instrument lines under the reactor vessel; the break size is not enough to depressurize the RCS, so inventory is expelled at high pressure, resulting in the equivalent of a small LOCA. Inventory cannot be replenished since the necessary pumps are failed by the SBO. Electric power recovery is not credited since it is assumed that the seismic damage to the emergency buses cannot be repaired. The core uncovers and results in core damage.

TABLE 5-8
SUMMARY OF TOP SCDF ACCIDENT SEQUENCES
(CONTINUED)

RANK	INITIATING EVENT	IE FREQUENCY	SCDF/YR	PERCENT OF SCDF	SEQUENCE PROGRESSION DESCRIPTION
7	G03 0.25-0.4g	3.31E-05	2.61E-08	0.20%	This seismic event causes the loss of offsite grid, and also either the Primary Plant Demineralized Water Storage Tank (PPDWST) or a correlated failure of equipment in the AFW piping. Main feedwater is assumed failed for all seismic events because its system contains non-seismic equipment in the turbine building. Operators attempt to perform bleed & feed, the final option for secondary heat removal, but fail. Once the secondary side of the steam generators boils dry, RCS pressure rises until the steam generator safety valves lift, releasing RCS inventory. At this much higher RCS pressure, the high-head injection pumps have a lower flow rate and cannot replenish the inventory being lost. The core uncovers and core damage occurs.
8	G03 0.25-0.4g	3.31E-05	2.52E-08	0.19%	This sequence is nearly identical to sequence rank #7, with an additional seismic failure of very small RCS lines that is inconsequential for this particular sequence. The break is not large enough for adequate RCS heat removal, so core damage still occurs.
9	G04 0.4-0.5g	7.91E-06	2.52E-08	0.19%	This seismic event causes a loss of offsite grid and also seismically fails the emergency AC buses, inducing a station blackout. A VSLOCA is caused by seismic failure of the instrument lines under the reactor vessel; the break size is not enough to depressurize the RCS, so inventory is expelled at high pressure, resulting in the equivalent of a small LOCA. Inventory cannot be replenished since the necessary pumps are failed by the SBO. Electric power recovery is not credited since it is assumed that the seismic damage to the emergency buses cannot be repaired. The core uncovers and results in core damage.

TABLE 5-8
SUMMARY OF TOP SCDF ACCIDENT SEQUENCES
(CONTINUED)

RANK	INITIATING EVENT	IE FREQUENCY	SCDF/YR	PERCENT OF SCDF	SEQUENCE PROGRESSION DESCRIPTION
10	G05 0.5-0.6g	3.99E-06	2.45E-08	0.19%	This seismic event fails offsite grid and seismically fails the river water pump trains, which fails cooling water to the diesel generators. The diesels will either overheat and fail very early into the sequence, or operators will secure them. However, electric power recovery for this sequence is not credited since it is assumed that the earthquake damage to the river water pump trains cannot be repaired. This puts the site in a station blackout. A very small LOCA is caused by seismic failure of the instrument lines under the reactor vessel; the break size is not enough to depressurize the RCS, so inventory is expelled at high pressure, resulting in the equivalent of a small LOCA. Inventory cannot be replenished since the high-head injection pumps are failed by the SBO, and would have also failed due to the lack of river water to cool the pumps. The core uncovers and results in core damage.

SSCs with the most significant seismic failure contributions to SCDF are listed in ***Table 5-9***, sorted by FVI. The seismic fragilities for each of the significant contributors are also provided in ***Table 5.-9,*** along with the corresponding limiting seismic failure mode and method of fragility calculation. FVI values for seismic equipment groups were calculated using RISKMAN's "Fragile Component Importance Report," for Sequence Group SEISL1 and Master Frequency File R6IMP. ***Table 5-9*** shows the top 25 seismic equipment groups, sorted by FV. It was revealed that setting various operator actions to guaranteed failure, with a value of 1.0 (common in the SPRA), was not allowing success sequences to be quantified, and thus there were FV values that were not being calculated appropriately. Sensitivity Case 38 was devised, in which the human actions in the model that had been set to 1.0 were reset to 0.999, and the model was quantified. The importance displayed in the following tables use the results from Sensitivity Case 38.

The fragilities reflect the outcome of the refinement process outlined in ***Section 4.4.2.16***. Among the top SCDF contributors are: Loss of Offsite Grid, Seismic-Induced Very Small LOCA, the PPDWST, RWST, and 4KV-480V Transformers. Loss of Offsite Grid is associated with brittle failure of the ceramic insulators on transformers. This is assigned a 0.1g HCLPF which is conservative, but is the recommended HCLPF based on EPRI SPRAIG Report 3002000709 (Reference 15), NUREG-1738 (Reference 59), and NUREG-CR-3558 (Reference 60) reports. The Seismic-Induced Very Small LOCA is associated with the failure of NSSS Piping assumed to occur at the bottom of the reactor pressure vessel. This failure mode is assigned a 0.125g HCLPF based on the BVPS-1 SSE PGA, which is aligned with Option 3 of section 5.4.4.2 in the EPRI SPRAIG report 3002000709 (Reference 15). These two contributors are important to CDF because together they provide a challenge to the plant of providing makeup to the reactor after a LOCA occurs, but both are identified as using industry accepted methodology to obtain the HCLPF values. The PPDWST is another top contributor to SCDF because it is the primary source of auxiliary feedwater to provide feedwater to the steam generators after a station blackout. The failure mode associated with this is an anchor bolt chair failure. This has a calculated HCLPF of 0.29g. This calculation was refined after the peer review to remove conservatives and is judged to be realistic. Similarly the RWST which is also a top contributor was refined to achieve a HCLPF of 0.33g which is also judged to be realistic. The RWST is important because it is the primary source of providing makeup to the reactor coolant system. Seismic failure of this tank is also associated with a tank shell rupture near anchor bolt chairs at base. The 4160V/480V transformers that supply power the emergency buses are another top contributor. Failure of these transformers will take out the emergency AC power buses. The failure mode associated with excess sliding displacement of coils within housing.

TABLE 5-9
SCDF IMPORTANCE MEASURES RANKED BY FV

RANK	GROUP	TOP EVENT	DESCRIPTION	FVI	HCLPF (G)	A_m	β_R	β_u	FAILURE MODE	FRAGILITY METHOD
1	EQ07	ZOG	Offsite Grid-Transformers	1.63E-01	0.1	0.25	0.24	0.32	Failure of Ceramic Insulators	Assigned
2	EQ55	ZVS	VSLOCA	1.04E-01	0.125	0.31	0.24	0.32	See Note (1)	See Note (1)
3	EQ14	ZAF	PPDWST (WT-TK-10)	7.78E-02	0.29	0.65	0.24	0.26	Anchor Bolt Chair Failure	CDFM
4	EQ13	ZRW	RWST (QS-TK-1)	6.34E-02	0.33	0.74	0.24	0.26	Tank shell rupture near anchor bolt chairs at base	CDFM
5	EQ08	ZAC	4KV-480V XFMR	5.66E-02	0.34	0.86	0.24	0.32	Excess sliding displacement of coils within housing	CDFM
6	EQ37	ZWC	All River Water – Pumps	5.09E-02	0.34	0.86	0.24	0.32	Anchor bolt shear-tension interaction	CDFM
7	EQ81	ZBW	Block Walls in SRVB	3.10E-02	0.38	0.96	0.24	0.32	Structural failure	CDFM
8	EQ57	ZRS	Recirc Spray HX 1B&1D	2.83E-02	0.28	0.71	0.24	0.32	Concrete breakout anchor bolt failure	CDFM
9	EQ96	ZTX	Turbine Building	2.80E-02	0.21	0.47	0.15	0.31	Closure of the seismic gap between the Turbine Building and the adjacent Service Building	SOV
10	EQ111	ZDW	Unit 2 DWST	1.85E-02	0.17	0.43	0.24	0.32	Tank overturning	CDFM
11	EQ113	ZWC	Intake Structure Contactors	9.99E-03	0.45	0.91	0.24	0.18	Contactor chatter	CDFM
12	EQ43	ZDG	Fuel Oil Tanks TK-1A/1B	9.59E-03	0.45	1.01	0.24	0.26	Tank shell rupture near nozzle	CDFM
13	EQ102	ZM6	MCC-1-E10	6.75E-03	0.20	0.50	0.24	0.32	Interaction with adjacent reinforced concrete wall	See Note (4)
14	EQ79	ZDG	EDG Air Start Receivers	6.69E-03	0.47	1.18	0.24	0.32	Tank sliding	CDFM

TABLE 5-9
SCDF IMPORTANCE MEASURES RANKED BY FV
(CONTINUED)

Rank	Group	Top Event	Description	FVI	HCLPF (G)	A_m	β_R	β_u	Failure Mode	Fragility Method
15	EQ56	ZLK	Small LOCA	5.58E-03	0.32	1	0.3	0.4	See Note (2)	See Note (2)
16	EQ97	ZOB	PZR PORV Pressure Reducing Valves (PCV-1GN-108,109,117)	5.43E-03	0.37	1.06	0.24	0.38	Shaft binding	CDFM
17	EQ66	ZBV	Emergency Switchgear (SWGR) HVAC Ducting	4.95E-03	0.5	1.26	0.24	0.32	Structural failure	CDFM
18	EQ75	ZBV	Emergency SWGR Dampers	4.95E-03	0.5	1.26	0.24	0.32	Fragility assigned based on inherent seismic ruggedness	Assigned Screening Threshold – See Note (3)
19	EQ76	ZWC	River Water Dampers	4.79E-03	0.5	1.26	0.24	0.32	Fragility assigned based on inherent seismic ruggedness	Assigned Screening Threshold – See Note (3)
20	EQ82	ZWC	ALL RW – Underground Piping	4.79E-03	0.5	1.26	0.24	0.32	Structural failure	CDFM
21	EQ64	ZWC	Valve Pits	4.67E-03	0.5	1.13	0.24	0.26	Structural failure	Fragility assigned based on inherent seismic ruggedness
22	EQ117	ZL1	MLOCA	4.66E-03	0.5	2	0.35	0.45	See Note (2)	See Note (2)
23	EQ36	ZWC	All River Water – REJs	4.53E-03	0.5	1.27	0.24	0.32	Fragility assigned based on inherent seismic ruggedness	Assigned Screening Threshold – See Note (3)

TABLE 5-9
SCDF IMPORTANCE MEASURES RANKED BY FV
(CONTINUED)

RANK	GROUP	TOP EVENT	DESCRIPTION	FVI	HCLPF (G)	A_m	β_R	β_u	FAILURE MODE	FRAGILITY METHOD
24	EQ67	ZWC	River Water HVAC Ducting	4.53E-03	0.5	1.27	0.24	0.32	Structural failure	CDFM
25	EQ44	ZDG	Fuel Oil Pumps	4.30E-03	0.5	1.26	0.24	0.32	Fragility assigned based on inherent seismic ruggedness	Assigned Screening Threshold – See Note (3)

Notes:

(1) The fragility for VSLOCA is assumed to have a HCLPF equal to the BV1 Site SSE based on Section 5.4.4 of the EPRI SPRA Implementation Guide.

(2) The fragility for SLOCA and MLOCA is assigned based on following Table H-2 of the EPRI SPRAIG (EPRI 3002000709). The fragilities in Table H-2 are considered to be representative fragilities based on a survey of available industry information. The failure mode specified is the RCS boundary failure.

(3) Assigned Screening Threshold means that the SSCs were determined to be sufficiently seismically rugged as determined from plant walkdown to conservatively assign a screening Level HCLPF which initially was 0.5g.

(4) The closure of the gap calculation is carried out as a median-centered analysis which directly provides A_m. Generic betas are then adopted to calculate a HCLPF.

The most significant non-seismic SSC failures (e.g., random failures of modeled components during the SPRA mission time) are listed in ***Table 5-10***.

Reference 17 contains the FV and RAW values for each component modeled in the SPRA, for both CDF and LERF sequences. Components were determined to be significant if the component's RAW is greater than 2 or its FV is greater than 0.005 for either CDF or LERF sequences, per the definition from the PRA Standard (Reference 4). RISKMAN report "Component Importance, With Common Cause and Maximum BE RAW" was used for FV, and "Component Importance, Without Common Cause and Maximum BE RAW" was used for RAW, created using the SEISL1 sequence group for CDF data. Judging against the above criteria, there were 43 risk significant components for CDF sequences. Note that only three components are important based on the FV criteria, and these three all cause failure of one train of the Emergency Diesel Generator system. The importances presented in ***Table 5-10*** also use the results from Sensitivity Case 38.

TABLE 5-10
NON-SEISMIC SIGNIFICANT COMPONENT LIST (SORTED BY SCDF FVI)

COMPONENT	COMPONENT DESCRIPTION	SCDF FV	SCDF RAW
BV-1EE-EG-1	NO.1 EMERGENCY DIESEL GENERATOR	2.63E-02	2.79E+00
BV-LS-1EE-201-1	EE-EG-1 DAY TANK LEVEL(PUMP CTRL) LEVEL SWITCH	1.85E-02	2.79E+00
BV-PNL-DG-SEQ-1	DIESEL GENERATOR AUTOMATIC SEQUENCE RELAY PANEL 1	5.21E-03	2.79E+00
BV-1VS-F-22A	DIESEL GENERATOR BUILDING DIRECT DRIVE FAN	1.57E-03	2.79E+00
BV-4KVS-1AE-1E9	INCOMING SUPPLY FROM DIESEL GEN. #1	1.29E-03	2.79E+00
BV-1VS-D-22-1A	DIESEL GENERATOR BLDG O.A. EXHAUST DAMPER	1.05E-03	2.79E+00
BV-1WR-P-1B	REACTOR PLANT RIVER WATER PUMP 1B	5.59E-04	6.41E+00
BV-1WR-P-1C	REACTOR PLANT RIVER WATER PUMP	1.23E-04	6.41E+00
BV-1FO-35	1A/1B TRANS PUMP SUCT CHECK	1.19E-04	2.79E+00
BV-1EE-S-1A	D.G. FUEL OIL PUMP BASKET STRAINER	6.18E-05	2.79E+00
BV-1EE-FL-1A	D.G. FUEL OIL PUMP DISCHARGE FILTER	5.54E-05	2.79E+00
BV-MCC-1-E7	480V MOTOR CONTROL CENTER FED FROM 480V SUBSTA 1-8 BUS 1N(8N14)	4.75E-05	2.79E+00
BV-480VUS-1-8N1	INCOMING SUPPLY FROM 4KVS-1AE-1E12	3.30E-05	2.92E+00
BV-480VUS-1-8N16	SUPPLY TO STUB BUS 8N1	3.20E-05	2.87E+00
BV-1RW-58	RP RW PP (1WR-P-1B) DISCH CHECK	3.04E-05	5.32E+00
BV-MOV-1RW-103D	1B HDR RP RW TO RECIRC SPRAY HXS ISOL	2.62E-05	6.68E+00
BV-1EE-E-1A	DIESEL COOLING WATER HEAT EXCHANGER	2.06E-05	2.79E+00
BV-1RW-57	RP RW PP (1WR-P-1A) DISCH CHECK	2.05E-05	5.32E+00
BV-TRANS-1-8N	4160V BUS 1AE TO EMERGENCY BUS 1N TRAN	2.03E-05	2.85E+00
BV-MOV-1RW-103C	1B HDR RP RW TO RECIRC SPRAY HXS ISOL	1.91E-05	6.68E+00

TABLE 5-10
NON-SEISMIC SIGNIFICANT COMPONENT LIST (SORTED BY SCDF FVI) (CONTINUED)

COMPONENT	COMPONENT DESCRIPTION	SCDF FV	SCDF RAW
BV-480VUS-1-8-N	480V SUBSTATION 1-8 EMERG BUS 1N	1.81E-05	2.93E+00
BV-4KVS-1AE	4160 EMERG BUS 1AE	1.81E-05	2.93E+00
BV-MOV-1RW-103A	1A HDR RP RW TO RECIRC SPRAY HXS ISOL	1.79E-05	6.68E+00
BV-480VUS-1-8-N1	480V SUBSTATION 1-8 AUX BUS 1N1	1.75E-05	2.87E+00
BV-4KVS-1AE	4160 EMERG BUS 1AE	1.81E-05	2.93E+00
BV-MOV-1RW-103A	1A HDR RP RW TO RECIRC SPRAY HXS ISOL	1.79E-05	6.68E+00
BV-480VUS-1-8-N1	480V SUBSTATION 1-8 AUX BUS 1N1	1.75E-05	2.87E+00
BV-TRANS-1-8N1	4160V BUS 1AE TO 480V SUBSTATION 1-8 B	1.42E-05	2.87E+00
BV-1RW-59	RP RW PP (1WR-P-1C) DISCH CHECK	1.26E-05	5.32E+00
BV-MOV-1RW-103B	1A HDR RP RW TO RECIRC SPRAY HXS ISOL	1.02E-05	6.68E+00
BV-1RW-222	AUX RW PP (1WR-P-9B) DISCH CHECK VLV	9.86E-06	5.32E+00
BV-1RW-221	AUX RW PP (1WR-P-9A) DISCH CHECK VLV	9.05E-06	5.32E+00
BV-1FW-59	(1WT-TK-10) CHEM ADD TK CONTROL	8.04E-06	1.29E+01
BV-1WT-TK-10	PRIMARY PLANT DEMIN WTR STORAGE TANK	4.85E-06	1.29E+01
BV-4KVS-1AE-1E12	FEED TO EMERGENCY 480V SUBSTATION 1-8 BUS 1N	3.71E-06	2.92E+00
BV-1RW-200	DISCH TO 1B MAIN CONDENSER OUTLET ISOL	2.43E-06	6.86E+00
RW-PIPE	RIVER WATER SYSTEM COMMON PIPE	7.96E-07	6.86E+00
BV-1FO-1	1A STOR TK SUPPLY ISOL	7.44E-07	2.79E+00
BV-1FO-18	NO. 1 DG FILTER INLET ISOL	7.44E-07	2.79E+00
BV-1FO-22	NO. 1 DG FILTER OUTLET ISOL	7.44E-07	2.79E+00
BV-1FO-28	NO. 1 DG DAY TANK ISOL	7.44E-07	2.79E+00
BV-1RW-114	DIESEL GEN HX (1EE-E-1A) OUTLET ISOL	7.44E-07	2.79E+00
BV-1RW-815	1A EMERG DG HX RW SUPPLY ISOL VLV	7.44E-07	2.79E+00

TABLE 5-10
NON-SEISMIC SIGNIFICANT COMPONENT LIST (SORTED BY SCDF FVI) (CONTINUED)

COMPONENT	COMPONENT DESCRIPTION	SCDF FV	SCDF RAW
BV-1EE-TK-10A	ENGINE (EE-EG-1) MOUNTED FUEL OIL TANK	7.28E-07	2.79E+00
BV-1EE-TK-1A	DIESEL GENERATOR FUEL OIL STORAGE TANK	7.28E-07	2.79E+00
BV-1EE-TK-2A	DIESEL GENERATOR #1 FUEL OIL DAY TANK	7.28E-07	2.79E+00

The contribution of each category of initiating events to the total CDF was calculated and is summarized in ***Table 5-11*** below. The table is sorted by the hazard range of the initiators. Initiating event category contribution was determined by using RISKMAN's "Contribution of Initiating Events to One Sequence Group" report, using the Master Frequency File REV6MFF with Sequence Group SEISL1, at a report cutoff of 1E-14, and a quantification truncation of 1E-14.

TABLE 5-11
INITIATING EVENT CONTRIBUTION TO SCDF

INITIATOR	HAZARD RANGE (G)	INTERVAL FREQUENCY	INTERVAL SCDF	% CONTRIBUTION	CUMULATIVE CDF
G01	0.06-0.15	5.33E-04	1.65E-08	0.13%	1.65E-08
G02	0.15-0.25	1.09E-04	9.71E-08	0.74%	1.14E-07
G03	0.25-0.4	3.31E-05	1.34E-06	10.28%	1.45E-06
G04	0.4-0.5	7.91E-06	2.70E-06	20.71%	4.15E-06
G05	0.5-0.6	3.99E-06	2.97E-06	22.78%	7.12E-06
G06	0.6-0.7	2.19E-06	2.08E-06	15.95%	9.20E-06
G07	0.7-0.8	1.32E-06	1.29E-06	9.89%	1.05E-05
G08	0.8-1.0	1.40E-06	1.39E-06	10.66%	1.19E-05
G09	1.0-2.0	1.07E-06	1.07E-06	8.21%	1.30E-05
G10	2.0-4.99	8.59E-08	8.58E-08	0.66%	1.30E-05
Total	**0.06-4.99**	**6.93E-04**	**1.30E-05**	**100%**	-

The major initiating events contributing to core damage from seismic are G04, G05, and G06. This range of hazards accounts for about 60 percent of CDF. The DBE for BVPS is 0.125g; which is within the G01 initiator range of accelerations. By contrast, such seismic events contribute much less than 1 percent of the total.

In addition to examining the sequences that contribute to CDF, it can be useful to identify the systems that are most important. One measure of importance can be determined by evaluating the effect on CDF if the system is assumed to have perfect reliability. This allows the systems to be ranked according to their contributions to overall CDF; i.e., the larger the impact on CDF if the system were perfect, the larger the contribution to the base-case CDF due to the failure of that system. This is a common importance measure, and is referred to as FV Importance (FVI).

System FV values were calculated using the data from RISKMAN's "Component Importance, With Common Cause and Maximum BE RAW" report, created using the SEISL1 sequence group and Master Frequency File R6IMP. Each component is then grouped into its Maintenance Rule system, and the component FVs for each separate system are added together to determine overall system FV values. The systems modeled in the PRA with a FV greater than or equal to 1E-05 are listed in ***Table 5-12***, sorted by largest FV value. The importance displayed in ***Table 5-12*** also uses the results from Sensitivity Case 38.

TABLE 5-12
SYSTEM IMPORTANCE BY FUSSELL-VESELY

Rank	System #	Description	FV
1	36A	Emergency Diesel Generators & Support Systems	5.08E-02
2	24B	Auxiliary Feedwater System	9.02E-03
3	36B	4KV Station Service System	6.97E-03
4	11	Safety Injection System	4.88E-03
5	44F	Area Ventilation Systems - Miscellaneous	3.89E-03
6	37	480 Volt Station Service System	2.06E-03
7	30	River Water System	2.02E-03
8	13	Containment Depressurization System	4.29E-04
9	39	125 VDC Distribution System	2.63E-04
10	07	Chemical and Volume Control System	2.14E-04
11	06	Reactor Coolant system	5.86E-05

The most important system is the EDGs and its Support Systems. The EDG would be called upon following a LOOP which is probable after a seismic event.

Reference 17 summarizes the contribution to seismic CDF from the most significant post-initiator human actions. Per Reference 4, significant post-initiator operator actions are defined as those operator action basic events that have a FVI value greater than 0.005 or a RAW greater than 2. The importance measures were calculated in RISKMAN and generated through the Basic Event Importance Report for Sequence Group Report in the Event Tree Module. Reports were generated for the Sequence Group SEISL1 (seismic CDF) and the Operator Action

Events were pulled out to make the summary table in Appendix J of Reference 17. Appendix J of Reference 17 also uses importances from Sensitivity Case 38; however operator actions that are guaranteed failed in the seismic model are excluded. There were only two operator actions that meet the risk significance criteria listed above. The top action is for operators failing to initiate feed and bleed after not restoring main feedwater for a seismic event greater than the plant SSE in which control room indication is not lost and the control ceiling is intact. This action is important because in many seismic scenarios feed and bleed is the only available source of primary cooling due to seismic failures. The other risk significant action is for operators failing to initiate cooldown and depressurization also for a seismic event greater than the plant SSE in which control room indication is not lost and the control ceiling is intact.

5.5 SLERF RESULTS

The seismic PRA performed for BVPS-1 shows that the point estimate mean seismic LERF is 6.14E-07. A discussion of the mean SLERF with uncertainty distribution reflecting the uncertainties in the hazard, fragilities, and model data is presented in ***Section 5.6***. Important contributors are discussed in the following paragraphs.

The top SLERF accident sequences are documented in the SPRA quantification report (Reference 17). These are briefly summarized in ***Table 5-13***.

TABLE 5-13
SUMMARY OF TOP SLERF ACCIDENT SEQUENCES

RANK	INITIATING EVENT	IE FREQUENCY	SLERF/YR	PERCENT OF SLERF	SEQUENCE PROGRESSION DESCRIPTION
1	G09 1.0-2.0g	1.07E-06	1.52E-07	24.74%	This earthquake directly causes core-damage and large early release, without potential for mitigation, due to structural failure of one or more of the Reactor Containment Building, Safeguards Building, Main Steam & Cable Vault Building, or the Steam Generators. See Section 4.5.1 of Reference 38 for a discussion of modeling of high-impact SSCs.
2	G10 2.0-4.99g	8.59E-08	7.60E-08	12.37%	This earthquake directly causes core-damage and large early release, without potential for mitigation, due to structural failure of one or more of the Reactor Containment Building, Safeguards Building, Main Steam & Cable Vault Building, or the Steam Generators. See Section 4.5.1 of Reference 38 for a discussion of modeling of high-impact SSCs.
3	G08 0.8-1.0g	1.40E-06	1.51E-08	2.45%	This earthquake directly causes core-damage and large early release, without potential for mitigation, due to structural failure of one or more of the Reactor Containment Building, Safeguards Building, Main Steam & Cable Vault Building, or the Steam Generators. See Section 4.5.1 of Reference 38 for a discussion of modeling of high-impact SSCs.
4	G07 0.7-0.8g	1.32E-06	3.66E-09	0.59%	This earthquake directly causes core-damage and large early release, without potential for mitigation, due to structural failure of one or more of the Reactor Containment Building, Safeguards Building, Main Steam & Cable Vault Building, or the Steam Generators. See Section 4.5.1 of Reference 38 for a discussion of modeling of high-impact SSCs.
5	G09 1.0-2.0g	1.07E-06	2.32E-09	0.38%	In this seismic event, the propane tanks across the Ohio River from the site are damaged and release a vapor cloud. This cloud ignites in the vicinity of the site, detonating/deflagrating and creating a shockwave that destroys critical structures and components, leading to direct core-damage and large early release. See Section 5.9 of Reference 38 for a discussion of modeling of the propane tank farm.
6	G06 0.6-0.7g	2.19E-06	1.84E-09	0.30%	This earthquake directly causes core-damage and large early release, without potential for mitigation, due to structural failure of one or more of the Reactor Containment Building, Safeguards Building, Main Steam & Cable Vault Building, or the Steam Generators. See Section 4.5.1 of Reference 38 for a discussion of modeling of high-impact SSCs.

TABLE 5-13
SUMMARY OF TOP SLERF ACCIDENT SEQUENCES
(CONTINUED)

RANK	INITIATING EVENT	IE FREQUENCY	SLERF/YR	PERCENT OF SLERF	SEQUENCE PROGRESSION DESCRIPTION
7	G08 0.8-1.0g	1.40E-06	1.68E-09	0.27%	In this seismic event, the propane tanks across the Ohio River from the site are damaged and release a vapor cloud. This cloud ignites in the vicinity of the site, detonating/deflagrating and creating a shockwave that destroys critical structures and components, leading to direct core-damage and large early release. See Section 5.9 of Reference 38 for a discussion of modeling of the propane tank farm.
8	G10 2.0-4.99g	8.59E-08	1.43E-09	0.23%	This seismic event is of the highest postulated magnitude. It causes the seismic failure of a high-impact SSC (see Section 4.5.1 of Reference 38), guaranteeing core damage. Regardless, the earthquake also causes the seismic failure of all components necessary to mitigate any accident sequence. It fails offsite power as well as the block walls surrounding the emergency batteries, which prevents the emergency diesel generators from starting, inducing a station blackout. All pumps that rely on AC power to operate are failed. Ultimately, RCS inventory is lost through a small-sized seismic-induced break in RCS piping (ZLK10 split fraction), and there is no means of injecting new inventory due to the SBO. Electric power recovery is not credited since operator actions to restart equipment are assumed failed for an earthquake of this magnitude. A number of other seismic failures occur, but ultimately it is the loss of inventory that leads to the core uncovering and resulting in core damage. During the seismic event, a large containment penetration (one or more of the personnel or equipment hatches or large electrical penetrations) was failed (ZCP10 split fraction), providing a large and early pathway for radiological material, thus binning this sequence to large early release.
9	G10 2.0-4.99g	8.59E-08	1.43E-09	0.23%	This sequence is identical to sequence rank #8, except that the RCS is at a different pressure (intermediate, instead of low) when the reactor vessel ruptures due to core meltthrough. The pressure is inconsequential due to the large containment penetration already failed for a large early pathway.

TABLE 5-13
SUMMARY OF TOP SLERF ACCIDENT SEQUENCES
(CONTINUED)

RANK	INITIATING EVENT	IE FREQUENCY	SLERF/YR	PERCENT OF SLERF	SEQUENCE PROGRESSION DESCRIPTION
10	G07 0.7-0.8g	1.32E-06	8.64E-10	0.14%	In this seismic event, the propane tanks across the Ohio River from the site are damaged and release a vapor cloud. This cloud ignites in the vicinity of the site, detonating/deflagrating and creating a shockwave that destroys critical structures and components, leading to direct core-damage and large early release. See Section 5.9 of Reference 38 for a discussion of modeling of the propane tank farm.

SSCs with the most significant seismic failure contribution to SLERF are listed in ***Table 5-14***, sorted by FVI. The seismic fragilities for each of the significant contributors are also provided in ***Table 5-14,*** along with the corresponding limiting seismic failure mode and method of fragility calculation.

Among the top SLERF contributors are a containment isolation valve in the MSCV building at elevation 722 ft, Offsite Grid failure, failure of the turbine building, and VSLOCA.

TABLE 5-14
IMPORTANCE MEASURES FOR SEISMIC COMPONENT FAILURES TO SLERF RANKED BY FUSSEL-VESELY IMPORTANCE

RANK	GROUP	TOP EVENT	COMPONENT DESCRIPTION	FVI	HCLPF (G)	AM	BR	BU	FAILURE MODE	FRAGILITY METHOD
1	EQ93	ZCI	MSCV 722 SOV CNMT ISO	3.34E-01	0.74	1.88	0.24	0.32	Functional failure of solenoid	CDFM
2	EQ07	ZOG	Offsite Grid	2.61E-01	0.1	0.25	0.24	0.32	Failure of Ceramic Insulators	Assigned
3	EQ96	ZTX	Turbine Building	1.82E-01	0.21	0.47	0.15	0.31	Closure of the seismic gap between the Turbine Building and the adjacent Service Building	SOV
4	EQ55	ZVS	VSLOCA	1.46E-01	0.125	0.31	0.24	0.32	See Note (1)	See Note (1)
5	EQ111	ZDW	Unit 2 DWST	1.44E-01	0.17	0.43	0.24	0.32	Tank overturning	CDFM
6	EQ02	ZL2	Steam Generators	1.32E-01	0.91	2.3	0.24	0.32	Exceeding allowable stress in support framing brace	CDFM
7	EQ92	ZCI	MSCV 722 Diaphragm POV CNMT ISO	9.33E-02	0.96	2.44	0.24	0.32	Shaft binding	CDFM
8	EQ102	ZM6	MCC-1-E10	5.63E-02	0.2	0.5	0.24	0.32	Interaction with adjacent reinforced concrete wall	See Note (2)
9	EQ14	ZAF	PPDWST (WT-TK-10)	3.94E-02	0.29	0.65	0.24	0.26	Anchor Bolt Chair Failure	CDFM
10	EQ03	ZL2	MS&CV Bldg	3.54E-02	1.23	2.63	0.16	0.3	Shear wall failure	SOV
11	EQ04	ZL2	Safeguards Bldg	2.96E-02	1.26	2.75	0.16	0.31	Shear wall failure	SOV
12	EQ74	ZOP	MSCV 722 Diaphragm POVs Outside CNMT ISO	2.44E-02	0.96	2.44	0.24	0.32	Shaft binding	CDFM
13	EQ13	ZRW	RWST (QS-TK-1)	2.34E-02	0.33	0.74	0.24	0.26	Tank shell rupture near anchor bolt chairs at base	CDFM
14	EQ71	ZIP	RCBX 718 Diaphragm POVs Inside CNTM ISO	2.26E-02	0.6	1.52	0.24	0.32	Shaft binding	CDFM
15	EQ81	ZBW	Block Walls in SRVB	1.61E-02	0.38	0.96	0.24	0.32	Structural failure	CDFM
16	EQ103	ZCP	EQ Hatches & Personnel Escape Airlock	1.52E-02	1.33	3.37	0.24	0.32	Structural failure	CDFM

TABLE 5-14
IMPORTANCE MEASURES FOR SEISMIC COMPONENT FAILURES TO SLERF RANKED BY FUSSEL-VESELY IMPORTANCE
(CONTINUED)

RANK	GROUP	TOP EVENT	COMPONENT DESCRIPTION	FVI	HCLPF (G)	A_M	B_R	B_U	FAILURE MODE	FRAGILITY METHOD
17	EQ104	ZCP	Equipment Hatch Crane Hoists	1.52E-02	1.33	3.37	0.24	0.32	Structural failure	CDFM
18	EQ105	ZCP	Electrical Penetration	1.52E-02	1.33	3.37	0.24	0.32	Structural failure	CDFM
19	EQ90	ZCP	Personnel Airlock	1.52E-02	1.33	3.37	0.24	0.32	Structural failure	CDFM
20	EQ101	ZPT	Propane Tank Farm	8.95E-03	0.45	1.03	0.24	0.26	Pier flexure	CDFM

Notes:

(1) The fragility for VSLOCA is assumed to have a HCLPF equal to the BV1 Site SSE based on Section 5.4.4 of the EPRI SPRA Implementation Guide.

(2) The closure of the gap calculation is carried out as a median-centered analysis which directly provides A_m. Generic betas are then adopted to calculate a HCLPF.

The most significant non-seismic SSC SLERF contributors (e.g., random failures of modeled components during the SPRA mission time) are listed in ***Table 5-15***.

Reference 17 contains the FV and RAW values for each component modeled in the SPRA, for both CDF and LERF sequences. Components were determined to be significant if the component's RAW is greater than 2 or its FV is greater than 0.005 for either CDF or LERF sequences, per the definition from the PRA Standard (Reference 4). RISKMAN report "Component Importance, With Common Cause and Maximum BE RAW" was used for FV, and "Component Importance, Without Common Cause and Maximum BE RAW" was used for RAW, created using the SEIS sequence group for CDF data. Judging against the above criteria, there were no risk significant components for LERF sequences; however, the top 10 components by FV for seismic LERF are presented below. Note that the top five components are related to the emergency diesel generators. The importances presented in ***Table 5-15*** also use the results from Sensitivity Case 38.

TABLE 5-15
NON-SEISMIC SIGNIFICANT COMPONENT LIST (SORTED BY SLERF FVI)

Component	Component Description	SLERF FV
BV-1EE-EG-1	No. 1 Emergency Diesel Generator	2.10E-04
BV-LS-1EE-201-1	EE-EG-1 Day Tank Level(Pump Ctrl) Level Switch	1.48E-04
BV-1EE-EG-2	No. 2 Emergency Diesel Generator	3.17E-05
BV-PNL-DG-SEQ-1	Diesel Generator Automatic Sequence Relay Panel 1	4.17E-05
BV-LS-1EE-202-1	EE-EG-1 Day Tank Level (Alarm) Level Switch	2.96E-05
BV-1FW-P-3A	No. 3A Motor Driven Auxiliary Feedwater Pump	2.10E-05
BV-1FW-P-3B	No. 3B Motor Driven Auxiliary Feedwater Pump	2.10E-05
BV-1VS-F-22A	Diesel Generator Building Direct Drive Fan	1.25E-05
BV-1SI-23	Loop 1 Cold Leg SI Sup Check	9.01E-06
BV-1SI-24	Loop 2 Cold Leg SI Sup Check	9.01E-06

A summary of the SLERF results for each seismic hazard interval is presented in ***Table 5-16***. The table is sorted by the hazard range of the initiators. Initiating event category contribution was determined by using RISKMAN's "Contribution of Initiating Events to One Sequence Group" report, using the Master Frequency File REV6MFF with Sequence Group LERFS, at a report cutoff of 1E-14, after quantification truncation of 1E-14

TABLE 5-16
INITIATING EVENT CONTRIBUTIONS TO LERF

Initiator	Hazard Range (g)	Interval Frequency	Interval LERF	% Contribution	Cumulative LERF
G01	0.06-0.15	5.33E-04	3.55E-12	< 0.01%	3.55E-12
G02	0.15-0.25	1.09E-04	7.94E-12	< 0.01%	1.15E-11
G03	0.25-0.4	3.31E-05	2.01E-10	0.03%	2.12E-10
G04	0.4-0.5	7.91E-06	9.92E-10	0.16%	1.20E-09
G05	0.5-0.6	3.99E-06	4.22E-09	0.69%	5.42E-09
G06	0.6-0.7	2.19E-06	1.04E-08	1.69%	1.58E-08
G07	0.7-0.8	1.32E-06	1.83E-08	2.98%	3.41E-08
G08	0.8-1.0	1.40E-06	6.75E-08	10.98%	1.02E-07
G09	1.0-2.0	1.07E-06	4.27E-07	69.51%	5.29E-07
G10	2.0-4.99	8.59E-08	8.57E-08	13.95%	6.14E-07
Total	**0.06-4.99**	**6.93E-04**	**6.14E-07**	**100%**	-

As shown in ***Table 5-16***, seismic LERF is dominated by acceleration intervals G08 through G10 which account for almost 95 percent of the LERF contribution. At these accelerations many of the buildings are collapsing causing large openings in the containment through penetrations or failure of the containment itself.

Appendix J in Reference 17 summarizes the contribution to seismic LERF from the most significant post-initiator human actions. Per Reference 4, significant post-initiator operator actions are defined as those operator action basic events that have a FV Importance value greater than 0.005 or a RAW greater than 2. The importance measures were calculated in RISKMAN and generated through the Basic Event Importance Report for Sequence Group Report in the Event Tree Module. Reports were generated for the Sequence Group LERFS (seismic LERF) and the Operator Action Events were pulled out to make the table in Appendix J in Reference 17. Appendix J in Reference 17 also uses importances from Sensitivity Case 38. Operator Actions that had a FVI of 0 and RAW of 1 for both CDF and LERF were excluded from the table as they are not important to the seismic CDF or LERF. Also operator actions that are guaranteed failed for seismic events are excluded.

Although no operator actions meet the risk significant criteria listed above, the top operator action to LERFS (seismic LERF) is the same as the most important action to seismic CDF. That is operators fail to initiate feed and bleed after not restoring main feedwater for a seismic event greater than the plant SSE in which control room indication is not lost and the control ceiling is intact.

5.6 SPRA Quantification Uncertainty Analysis

Parameter uncertainty relates to the uncertainty in the computation of the parameter values for initiating event frequencies, component failure probabilities, and HEP that are used in the quantification process of the PRA model. These uncertainties can be characterized by probability distributions that relate the analysts' degree of belief in the values that these parameters could take. To make a risk-informed decision, the numerical results of the PRA, including their associated uncertainty, must be compared with the appropriate decision criteria.

The RISKMAN software has the capability to correlate selected input distributions, propagate these uncertainties via a Monte Carlo quantification, and calculate the probability distributions for the risk metrics of the SPRA. These distributions and main uncertainty parameters (mean, 5th percentile, 50th percentile, and 95th percentile) are provided below for the seismically initiated CDF and LERF.

The parametric uncertainty results present an estimation of the uncertainty introduced by the data used to quantify the PRA model. Such data uncertainty typically shows a relatively tight distribution for internal events in a commercial nuclear plant PRA as a result of the types of distributions used (largely lognormal) and the relatively large amount of operational experience for most modeled components. For seismically initiated accident sequences this is not the case. The uncertainties in the family of seismic hazard exceedance curves, and the SSC fragility curves can be large, and with a much large impact than the data distributions applicable to internal events.

For the propagation of parameter uncertainties to seismic CDF and LERF the Uncertainty Analysis feature of RISKMAN was used. This feature requantifies the sequences using distributions for the input variables (initiators and split fractions) utilizing a Monte Carlo simulation. This method accounts for the uncertainty from all the input data parameters.

This parameter uncertainty estimation does not, however, reflect possible effects on the results from other sources of uncertainty. Such sources may include such things as: optimism or pessimism in definitions of sequence, component, or Human-Action success criteria; limitations in sequence models due to simplifications (for example, not modeling available systems or equipment) made to facilitate quantification; uncertainty in defining human response within the emergency procedures (for example, if there are choices that can be made); degree of completeness in selection of initiating events; assumptions regarding phenomenology or SSCs behavior under accident conditions (for example, RCP seal LOCA modeling assumptions). While it is difficult to quantify the effects of such sources of uncertainty, it is important to recognize and evaluate them because there may be specific PRA applications where their effects may have a significant influence on the results.

The results of the base-case seismic model parameter uncertainty analysis are shown in ***Table 5-17*** and ***Figure 5-2 and Figure 5-3.***

TABLE 5-17
PARAMETER UNCERTAINTY ANALYSIS RESULTS

	MEAN	5%	50%	95%
CDF (/Year), 10,000 Samples	1.30E-05	1.16E-06	7.29E-06	4.39E-05
LERF (/Year), 10,000 Samples	6.14E-07	2.25E-08	2.80E-07	2.32E-06

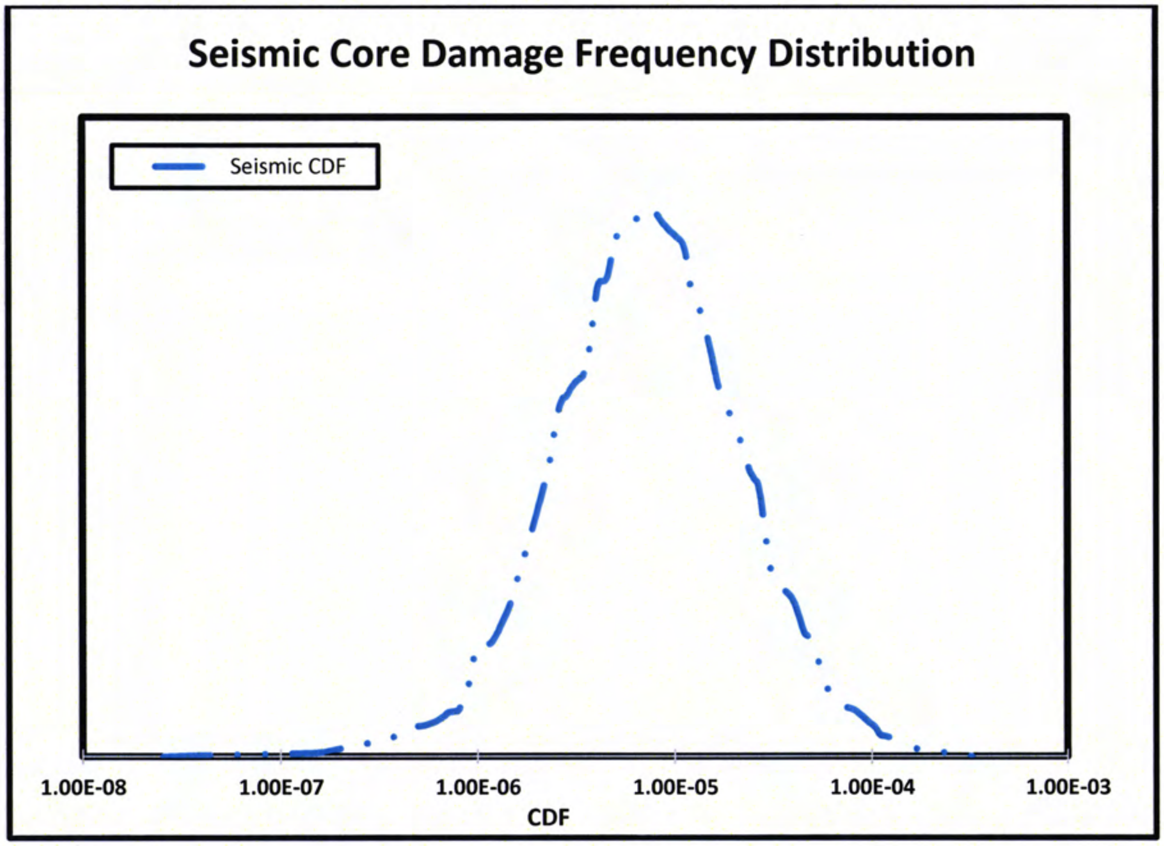


FIGURE 5-2
SPRA CDF UNCERTAINTY DISTRIBUTION (10,000 SAMPLES)

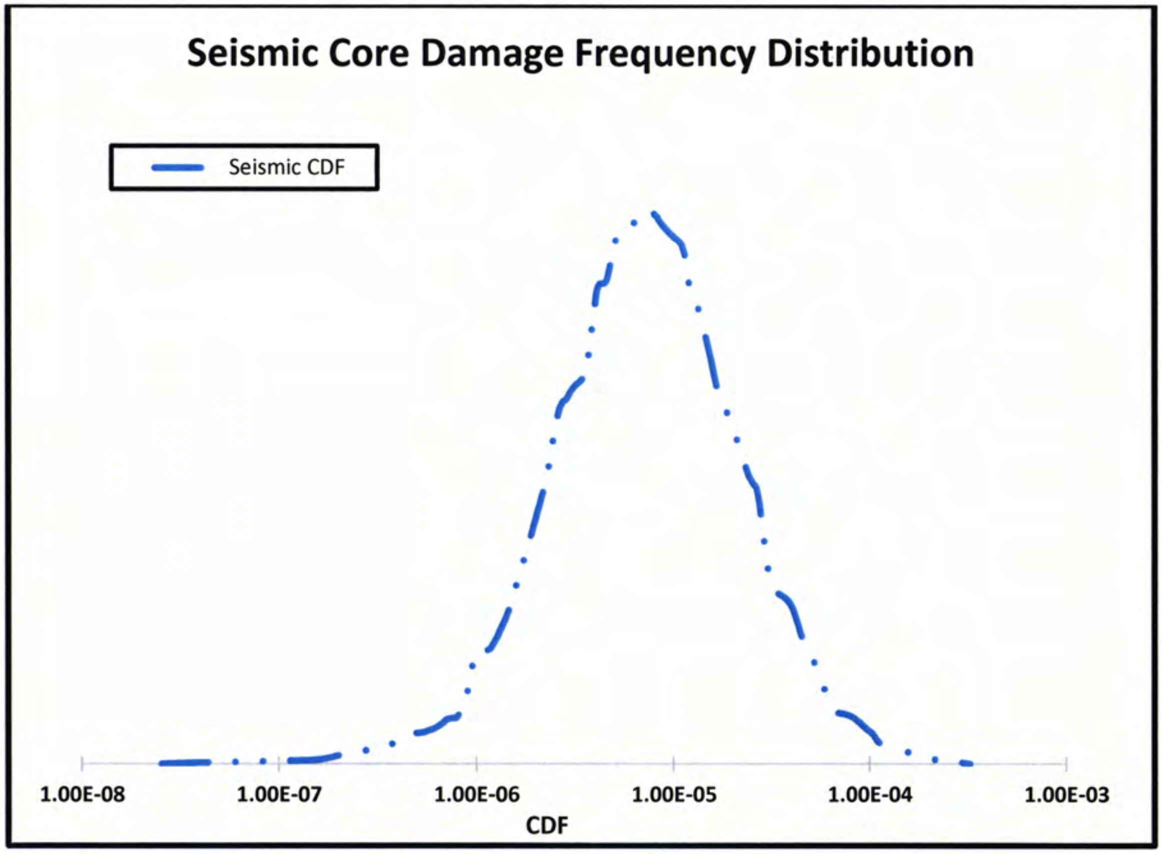


FIGURE 5-3
SPRA LERF UNCERTAINTY DISTRIBUTION (1,500 SAMPLES)

5.6.1 Model Uncertainty

Model uncertainty arises because different approaches exist to represent plant response. A source of model uncertainty is one related to an issue in which no consensus approach or model exists, and where the choice of approach or model is known to have an effect on the SPRA. These uncertainties are typically dealt with by making assumptions; e.g., the approach to address common-cause failure, how a RCP would fail following a loss of seal cooling, the approach to identify and quantify HFEs. In general, model uncertainties are addressed through sensitivity studies using different models or assumptions.

The guidance provided in EPRI 1016737, *Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments* (Reference 74), was used to address sources of model uncertainty and related assumptions. It provides a framework for the pragmatic treatment of uncertainty characterization to support risk-informed applications and decision-making. The process includes identification and characterization of sources of model uncertainty and related assumptions; the following sections summarize the sources of uncertainty found in the Level 1 SPRA.

5.6.2 Understood and Accepted Generic Uncertainties

Three issues that are generally understood and accepted as potential generic sources of model uncertainty are:

1. Treatment of Pre-Initiator and Post-Initiator Human Errors; i.e., screening human error probabilities, realistic HEPs for significant HFEs, realistic HEPs for all HFEs.
2. Treatment of Potentially Dependent Post-Initiator Human Errors; i.e., no HFE dependence, some dependent HFEs, all HFEs assessed for dependence.
3. Intra-System Common Cause Events; i.e., generic common cause failure (CCF), plant-specific CCF.

Based on lessons learned, a standard set of sensitivity cases was recommended to envelope these understood and accepted generic sources of uncertainty at a high level (Reference 16).

The four sensitivity cases are:

1. All HEPs set to their 5^{th} percentile value.
2. All HEPs set to their 95^{th} percentile value.
3. All CCF probabilities set to their 5^{th} percentile value.
4. All CCF probabilities set to their 95^{th} percentile value.

The results for these four sensitivity cases are presented in ***Table 5-18*** of ***Section 5.7.1***.

5.6.3 Generic Sources of Model Uncertainty

A generic list of additional sources of model uncertainty for internal events PRA was identified based on Reference 16. This list includes those having the highest potential to change risk metrics and decisions, and includes: phenomena or nature of the event or failure mode not completely understood; models based on significant interpretations; and issues with general agreement. Table I-1 in Appendix I of Reference 17 includes the list of generic pressurized water reactor (PWR) sources of model uncertainty and a characterization assessment for the BVPS-1 Level 1 SPRA.

5.6.4 Plant-Specific Sources of Model Uncertainty

An examination of plant-specific features and modeling approaches was also performed to identify any uncertainties not identified on the generic list. This assessment focused on identifying plant-specific features, modeling approaches and assumptions that were not included in the generic uncertainties. Table I-2 in Appendix I of Reference 17 includes the list of plant-specific sources of model uncertainty and a SPRA characterization assessment for the BVPS-1 Level 1 PRA; exceptions include generic sources of model uncertainty, alignments, and boundary systems that are not modeled because they have no impact on the PRA function system modeled.

Table I-3 of Reference 17 identifies sources of uncertainties from the assumptions listed in Section 2 of Reference 38. These assumptions are specifically related to the plant-specific SPRA for BVPS-1. The table describes the impact of the assumption on the SPRA modeling and then characterizes whether the uncertainty in the current assessment could potentially impact plant risk-based applications.

5.6.5 Completeness Uncertainty

Completeness uncertainty relates to risk contributors that are not in the SPRA model, nor were they considered in the development of the model. These include known types such as the scope of the PRA, which does not include some classes of initiating events, hazards, and operating modes; and the level of analysis, which may have omitted phenomena, failure mechanisms, or other factors because their relative contribution is believed to be negligible. They also include ones that are not known such as the effects on risk from aging or organizational changes; and omitted phenomena and failure mechanisms that are unknown. Both can have a significant impact on risk.

No completeness uncertainties were identified for the BVPS-1 Level 1 SPRA, based on the ASME/ANS PRA Standard (Reference 4).

5.7 SPRA Quantification Sensitivity Analysis

As presented in ***Section 5.7.1***, four standard sensitivity studies were selected for analysis:

- All HEP probabilities set to their 5th percentile value.
- All HEP probabilities set to their 95th percentile value.
- All CCF probabilities set to their 5th percentile value.
- All CCF probabilities set to their 95th percentile value.

The HEPs and CCF probabilities were changed to the 5th or 95th percentiles by importing distributions in the data module using the import distribution parameters function. The import file was created by exporting the parameters using the export distribution parameters function and the mean values were adjusted to the 5th or 95th percentile. The percentile values were taken from the RISKMAN titles listing report in the data module. The distributions affected were all Human-Action and beta, gamma, and delta factors used in the Multiple Greek Letter common-cause method in the model. Both CDF and LERF were requantified at the 5th or 95th percentiles for HEPs and CCF probabilities in separate cases.

The resulting 5th and 95th percentile values represent the CCF sensitivity cases listed above. The results of these sensitivity cases are discussed here and compared to the RG 1.174 CDF limit of $1x10^{-4}$/year for CDF and $1x10^{-5}$/year for LERF to obtain insights into the sensitivity of the base PRA model results to these generic high level sources of modeling uncertainty. This approach is followed rather than trying to identify all potential sources of model uncertainty associated with these issues since they are generally understood and accepted as areas of uncertainty that can be significant contributors to CDF. The results of the studies are shown in ***Table 5-18.***

The results indicate that CDF is more sensitive to these uncertainties than LERF, and each of the models are more sensitive to operator action uncertainty than they are to common-cause uncertainty. However, overall the model does not produce drastic changes for these sensitivity studies.

TABLE 5-18
CCF AND HEP SENSITIVITY CASES

CASE	5%	5% Δ FROM BASELINE	95%	95% Δ FROM BASELINE
HEP-CDF (/year)	1.25E-05	-3.93%	1.40E-05	7.56%
HEP-LERF (/year)	6.14E-07	-0.02%	6.15E-07	0.06%
CCF-CDF (/year)	1.30E-05	-0.15%	1.31E-05	0.28%
CCF-LERF (/year)	6.15E-07	0.00%	6.14E-07	0.00%

5.7.1 Seismic-Related Sensitivity Cases

This section presents the sensitivity results for selected cases defined specifically for the modeling of seismic events.

The uncertainties in the assessment of the seismic hazard curve, and of SSC fragilities are captured in the parameters that define these intermediate results; i.e., by the family of seismic hazard exceedance curves, and the parameters for each of the SSC fragilities; A_m, β_r, and β_u.

The results of the uncertainty analysis presented in ***Section 5.6*** illustrate the impact of uncertainties in the hazard exceedance curves and fragility curves on CDF and LERF. Therefore no further sensitivities were performed to assess these parameter uncertainties.

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

Each of the assumptions listed previously in Section 2 of the Quantification Notebook (Reference 17) and in other notebooks was examined to determine if a sensitivity case was feasible and instructive. The following areas were investigated:

1. Modeling of Seismic Impacts
2. Correlation of Fragilities
3. Relay Chatter
4. Human Reliability Analysis
5. Quantification Methods
6. Fragility Refinement Impacts

The results for each of the seismic-related sensitivity cases are provided in ***Table 5-19***. All sensitivities were performed using the Level 2 model, which can calculate Level 1 results, but is slightly lower than the actual Level 1 results because sequences that are close to the 1E-14 cutoff for core damage will drop below the 1E-14 cutoff after progressing through the CET tree for LERF. This is deemed acceptable for these sensitivities because the insights will be the same.

TABLE 5-19
SEISMIC-RELATED SENSITIVITY RESULTS

GROUP	CASE	SENSITIVITY STUDY (SEE NOTES BELOW AS WELL)	CDF	% CHANGE IN CDF	LERF	% CHANGE IN LERF
N/A	0	BASE CASE	1.29E-05	--	6.15E-07	--
1	1	LOOP ALWAYS TRUE	1.34E-05	3.60%	6.15E-07	0.04%
1	2	NO TURBINE BUILDING IMPACTS (FOR UNIT 1 - CREDIT STATION AIR, MFW, DAFW)	1.11E-05	-13.91%	6.00E-07	-2.29%
1	3	CREDIT ERFS BLACK DIESEL GENERATOR (FOR UNIT 1 - CREDIT DAFW)	1.14E-05	-11.26%	6.04E-07	-1.77%
1	4	DG 48 HOURS (CHANGE @T24 TO @T48 LOCAL VARIABLES FOR ALL SYSTEM TOPS)	1.28E-05	-0.62%	6.14E-07	-0.01%
1	5	EXTEND LERF EVACUATION TIME TO 48 HRS (REBIN OF LATE TO LERF DUE TO EXTENDED TIME)	1.29E-05	0.00%	6.18E-07	0.51%
1	6	NO VERY SMALL LOCA	1.20E-05	-7.06%	6.14E-07	-0.04%
1	7	ELIMINATE IMPACTS OF BLOCK WALL FAILURES	1.25E-05	-2.93%	6.14E-07	-0.12%
1	8	CREDIT FOR NO LOSS OF FIRE PROTECTION WATER	1.29E-05	0.01%	6.15E-07	0.00%
1	10	NO CREDIT PORTABLE GENERATORS IN TURBINE BUILDING	1.28E-05	-0.42%	6.11E-07	-0.54%
1	11**	ADDED INTERVALS - EXPAND TO MODEL.05G DELTA IN RANGE OF CHANGE; 0.25G TO 0.5G	1.28E-05	-0.49%	6.09E-07	-0.82%
1	13A	REMOVE IMPACT OF SFGB AND MSCV BUILDINGS ON LERF	1.29E-05	0.00%	5.74E-07	-6.55%
1	13b	REMOVE IMPACTS OF SG SAFETY VALVES AND ATMOSPHERIC RELIEF VALVES AND RHR VALVES ON AFW	1.29E-05	0.19%	6.16E-07	0.19%
1	13c	ASSUME SG SAFETY VALVES AND ATMOSPHERIC RELIEF VALVES AND RHR VALVES FAIL SEISMICALLY OPEN INSTEAD OF CLOSED	1.29E-05	-0.04%	6.14E-07	0.00%
1	15a	ELIMINATE SEISMIC FAILURE OF PPDWST	1.19E-05	-7.42%	6.10E-07	-0.81%
1	15b	ELIMINATE SEISMIC FAILURE OF RWST	1.21E-05	-5.86%	6.17E-07	0.47%

TABLE 5-19
SEISMIC-RELATED SENSITIVITY RESULTS
(CONTINUED)

GROUP	CASE	SENSITIVITY STUDY (SEE NOTES BELOW AS WELL)	CDF	% CHANGE IN CDF	LERF	% CHANGE IN LERF
1	15c	CASE 9B (NO RELAY FAILURES) PLUS CASE 13A (REMOVE MS&CV AND SFGB BLDGS (EQ03 & EQ04) FROM ZL2)	1.29E-05	0.01%	5.74E-07	-6.54%
1	15f	CREDIT FIRE HDR (CASE 8) PLUS CREDIT PORTABLE EMER SWGR FANS (ZBV=S)	1.27E-05	-1.27%	6.14E-07	0.00%
1	15g	NO RELAY FAILURES (CASE 9B) + CREDIT LPGP & FIRE HDR (CASE 8) + ELIMINATE SEISMIC FAILURE OF BAT 5/CHGR (EQ20) + ZBV=S	1.27E-05	-1.19%	6.15E-07	0.08%
1	15h	NO RELAY FAILURES (CASE 9B) + CREDIT LPGP & FIRE HDR (CASE 8) + ELIMINATE SEISMIC FAILURE OF BAT 5/CHGR (EQ20) + ZBV=S PLUS REMOVE MS&CV AND SFGB BLDGS (EQ03 & EQ04) FROM ZL2	1.27E-05	-1.23%	5.74E-07	-6.65%
1	15i	REMOVE FAILURE OF PROPANE TANK FARM	1.29E-05	0.27%	6.13E-07	-0.22%
1	15j*	NO CREDIT FOR FLEX	1.31E-05	1.74%	6.15E-07	0.02%
1	22	GUARANTEE FAIL CROSS-TIE; NO CORRELATION OF UNIT 1 AND 2 DGS	1.29E-05	0.11%	6.15E-07	0.00%
2	14b	CORRELATE THE SEISMIC FAILURE OF SFGB AND MSCV BUILDINGS	1.29E-05	0.00%	5.96E-07	-2.95%
3	9b	REMOVE ALL RELAY CHATTER IMPACTS	1.29E-05	0.01%	6.15E-07	0.01%
3	9c	REMOVE RELAY CHATTER AND REACTOR INTERNALS	1.29E-05	0.01%	6.15E-07	0.01%
4	17*	HRA 5TH %	1.25E-05	-3.93%	6.14E-07	-0.02%
4	18*	HRA 95TH %	1.40E-05	7.56%	6.15E-07	0.06%
4	19	SEIS3 TIMING SENSITIVITY 1 (SENS 1 TDELAY +30 MIN, TEXE X1 CR, TEXE X4 OUTSIDE MCR)	1.29E-05	0.02%	6.15E-07	0.00%
4	20	SEIS3 TIMING SENSITIVITY 2 (SENS 2 TDELAY +30 MIN, TEXE X2 CR, TEXE X4 OUTSIDE MCR)	SAME AS CASE 19			
4	21	SEIS3 TIMING SENSITIVITY 3 (SENS 3 TDELAY +15 MIN, TEXE X1 CR, TEXE X4 OUTSIDE MCR (MAX 30 MINUTES))	1.29E-05	0.01%	6.15E-07	0.00%
4	23	0.1 MINIMUM SEIS3 HEP	1.25E-05	-2.90%	6.14E-07	0.00%
4	24	REMOVE ZO3 AND ZO4 FROM SEIS MACROS	1.40E-05	8.72%	6.15E-07	0.11%
4	25	REMOVE PROPANE TANK FROM SEIS LEVELS	1.29E-05	0.02%	6.14E-07	0.00%

TABLE 5-19
SEISMIC-RELATED SENSITIVITY RESULTS
(CONTINUED)

GROUP	CASE	SENSITIVITY STUDY (SEE NOTES BELOW AS WELL)	CDF	% CHANGE IN CDF	LERF	% CHANGE IN LERF
4	26	REMOVE ZO3 AND ZO4 FROM SEIS MACROS (CASE 24) AND REMOVE PROPANE TANK FROM SEIS MACROS (CASE 25)	1.29E-05	0.02%	6.15E-07	0.11%
4	38	CHANGE 1.0 POST-TRIP HEPS TO 0.99	1.29E-05	0.01%	6.15E-07	0.00%
5	27*	CCF 5TH %	1.30E-05	-0.15%	6.14E-07	0.00%
5	28*	CCF 95TH %	1.31E-05	0.28%	6.14E-07	0.00%
5	29*	TRUNCATION SENSITIVITY (TRUNC = 1E-08)	1.29E-05	-0.01%	2.43E-07	-60.44%
5	30*	TRUNCATION SENSITIVITY (TRUNC = 1E-09)	1.30E-05	0.93%	2.56E-07	-58.42%
5	31*	TRUNCATION SENSITIVITY (TRUNC = 1E-10)	1.31E-05	1.61%	2.89E-07	-52.97%
5	32*	TRUNCATION SENSITIVITY (TRUNC = 1E-11)	1.33E-06	-89.68%	4.16E-07	-32.34%
5	33*	TRUNCATION SENSITIVITY (TRUNC = 1E-12)	1.22E-05	-5.74%	5.26E-07	-14.37%
5	34*	TRUNCATION SENSITIVITY (TRUNC = 1E-13)	1.28E-05	-0.78%	5.87E-07	-4.43%
5	35*	TRUNCATION SENSITIVITY (TRUNC = 1E-14)	1.30E-05	0.77%	6.15E-07	0.00%
5	36*	ZERO MAINTENANCE	1.30E-05	1.00%	6.15E-07	0.01%
6	EQ55	ZVS – VERY SMALL LOCA 2*AM	1.22E-05	-5.54%	6.12E-07	-0.35%
6	EQ14	ZAF - PPDWST (WT-TK-10) WITH 2*AM	1.196E-05	-7.33%	6.11E-07	-0.66%
6	EQ13	ZRW- RWST WITH 2*AM	1.213E-05	-5.97%	6.15E-07	0.03%
6	EQ08	ZAC - 4KV-480V XFMR WITH 2*AM	1.217E-05	-5.67%	6.12E-07	-0.46%
6	EQ37	ZWC - ALL RIVER WATER PUMPS WITH 2*AM	1.224E-05	-5.12%	6.14E-07	-0.12%
6	EQ81	ZBW - BLOCK WALLS IN SRVB WITH 2*AM	1.251E-05	-3.03%	6.12E-07	-0.37%
6	EQ93	ZCI - MSCV 722 SOV CNMT ISOLATION WITH 2*AM	1.291E-05	0.04%	4.13E-07	-32.71%
6	EQ96	ZTX - TURBINE BUILDING WITH 2*AM	1.283E-05	-0.53%	6.09E-07	-0.86%
6	EQ111	ZWD – U2 – DWST 2*AM	1.29E-05	0.00%	6.14E-07	0.00%
6	EQ02	ZL2 - STEAM GENERATORS WITH 2*AM	1.290E-05	-0.02%	5.35E-07	-12.94%
6	EQ92	MSCV 722 DIAPHRAGM POV CNMT ISOLATION WITH 2*AM	1.290E-05	0.00%	5.57E-07	-9.32%

TABLE 5-19
SEISMIC-RELATED SENSITIVITY RESULTS
(CONTINUED)

GROUP	CASE	SENSITIVITY STUDY (SEE NOTES BELOW AS WELL)	CDF	% CHANGE IN CDF	LERF	% CHANGE IN LERF
6	EQ102	ZM6 – MCC-1-E10	1.29E-05	0.05%	6.13E-07	-0.27%
6	EQ03	ZL2 - MS&CV BLDG WITH 2*AM	1.290E-05	0.00%	5.93E-07	-3.50%

Notes:

* These cases were quantified with the Level 1 and Level 2 models separately and the CDF results are compared with the 1.30E-5 seismic CDF instead of the CDF bin in the Level 2 model which truncates some CDF sequences and has a value of 1.29e-05.

** Case 11 was not performed using the updated model, instead results from the previous revision are referenced and its insights are judged applicable to the current revision.

5.7.1.1 Group 6: Fragility Refinement Impacts

The preceding seismic sensitivity cases reflect those sensitivities defined to determine the impacts of selected modeling assumptions on the CDF and LERF calculations. The cases described below are defined to examine the sensitivity of CDF and LERF to assumed improvements in the seismic capacities of the most important equipment fragility groups. One can use the FVI rankings directly for this purpose, but the FVI measure is a bounding measure assuming the SSCs in the equipment fragility groups are made perfect. For these added cases a seismic capacity improvement equal to twice the base case evaluated capacities is assumed, one equipment group at a time. Further fragility analysis is unlikely to achieve such an assessed improvement because much effort has already been dedicated to making the SSC seismic capacity assessments as realistic as possible. These cases are incorporated into the model by replacing the base median acceleration capacity, Am, by twice the Am. The Beta-r and Beta-u values are held the same so that the HCLPF accelerations are also twice the base-case values.

The FVI measures computed from Sensitivity Case 38 were used to identify fragility groups for these sensitivities as results from this case give more accurate importances as identified earlier in this submittal. The fragility component groups with FVI less than 0.03 were deleted from further consideration. They were deleted because even if they could be made perfect, the maximum reduction in CDF or LERF would be 0.03. Also deleted from further consideration was the fragility group for failures of the offsite grid (EQ07). This fragility group was assessed using generic data that is not specific to BV Unit 1 and is an industry accepted value and should not change in the near future.

All sensitivities were performed using the Level 2 model, which can also calculate Level 1 results, although its Level 1 results are slightly lower than the actual Level 1 results because sequences that are close to the 1E-14 cutoff for core damage will drop below the 1E-14 cutoff after progressing through the CET tree for LERF. This is deemed acceptable for these sensitivities because the insights will be the same.

Table 5-20 below identifies the fragility groups evaluated for the twice Am sensitivities. The top half of ***Table 5-20*** is for CDF contributors and the bottom of the table for LERF contributors. The CDF and LERF changes are nevertheless presented for all cases. The FVI measures from Sensitivity Case 38 are presented in the table as well as the revised CDF and LERF and changes in CDF and LERF are presented. All CDF frequency changes were less than 1E-6 per year. All LERF frequency changes were less than 2E-7 per year. The percent changes in CDF or LERF were, as expected, found to be less than the FVI of the fragility group to that risk measure and in some cases the change in CDF and LERF were negligible.

The largest potential decrease in CDF would come from increasing the PPDWST fragility. This fragility was already refined to remove conservatisms identified by the peer review team. Any further improvement would have to come a plant modification. The remaining fragility groups identified for CDF with the exception of VSLOCA have also been refined to remove conservatisms. Similarly, to achieve the risk reduction identified in the table below a plant modification to the identified SSCs would be needed. The VSLOCA fragility is based off of industry accepted methodology and although conservative is an accepted value. The low seismic CDF of 1.30E-05 justifies the acceptance of the conservativisms in the VSLOCA fragility as well as eliminates the need for any modifications. Additionally the delta CDFs in the mid to low 1E-7 range is further justification for accepting the conservativisms in the VSLOCA fragility and further justifies the basis for no plant modifications.

The largest potential decrease in LERF would come from increasing the capacity of the containment isolation valves in correlation group EQ93. This sensitivity case identifies a modeling conservatism taken in response to a peer review suggestion to treat multiple small containment penetrations as large. These are likely to still be small however the decision was made to treat them conservatively. Furthermore with the 0.75g HCLPF doubling the Am would give a much larger fragility than reasonably achievable. The seismic LERF value of 6.14E-07 is sufficiently low to accept this conservativism as well as the conservatism in the VSLOCA fragility. Similar to the identified CDF components the remaining identified LERF components have also been refined to remove conservatisms. It is judged to achieve the risk reductions identified in the table below a plant modification would be needed for the remaining SSCs. The low seismic LERF of 6.14E-07 eliminates the need for any modifications.

It is concluded that all other fragility groups, not evaluated here, if evaluated with twice the current capacities would lead to a reduction in CDF or LERF of less than 3%, and more precisely to less fractional reduction than their current FVI measures suggest. These SSCs are not important enough to justify refining the fragility because possible conservatisms in the fragility calculations are not driving the model results or masking insights.

TABLE 5-20
SENSITIVITY OF CDF AND LERF TO ASSUMED IMPROVEMENTS IN SEISMIC CAPACITIES

CASE ID	DESCRIPTION	BASE-CASE HCLPF	CDF FVI	CDF	CDF DIFFERENCE FROM	PERCENT CHANGE IN CDF	LERF FVI	LERF	LERF DIFFERENCE FROM*	PERCENT CHANGE IN LERF
Sensitivities at 1E-14/year; Set Am => 2*Am; all SSCs with FVI >3E-2 to CDF (exclude LOOP and VSLOCA)					**1.290E-05**				**6.145E-07**	
EQ55	ZVS – Very Small LOCA	0.125g	1.04E-01	1.22E-05	-7.15E-07	-5.54%	1.46E-01	6.12E-07	-2.15E-09	-0.35%
EQ14	ZAF - PPDWST (WT-TK-10)	0.29g	7.78E-02	1.20E-05	-9.45E-07	-7.33%	3.94E-02	6.11E-07	-4.04E-09	-0.66%
EQ13	ZRW- RWST	0.32g	6.34E-02	1.21E-05	-7.70E-07	-5.97%	2.34E-02	6.15E-07	1.60E-10	0.03%
EQ08	ZAC - 4kv-480V XFMR	0.34g	5.66E-02	1.22E-05	-7.32E-07	-5.67%	6.77E-03	6.12E-07	-2.82E-09	-0.46%
EQ37	ZWC - All river water pumps	0.34g	5.09E-02	1.22E-05	-6.60E-07	-5.12%	1.59E-03	6.14E-07	-7.40E-10	-0.12%
EQ81	ZBW - Block walls in SRVB	0.38g	3.10E-02	1.25E-05	-3.91E-07	-3.03%	1.61E-02	6.12E-07	-2.28E-09	-0.37%
Sensitivities at 1E-14/year; Set Am => 2*Am, for all SSCs with FVI >3E-2 to LERF (exclude LOOP and VSLOCA)										
EQ93	ZCI - MSCV 722 SOV CNMT ISOLATION	0.75g	~0.00E+00	1.29E-05	5.00E-09	0.04%	3.34E-01	4.13E-07	-2.01E-07	-32.71%
EQ96	ZTX - Turbine Building	0.22g	2.80E-02	1.28E-05	-6.90E-08	-0.53%	1.82E-01	6.09E-07	-5.28E-09	-0.86%
EQ55	ZVS – Very Small LOCA	0.125g	1.04E-01	1.22E-05	-7.15E-07	-5.54%	1.46E-01	6.12E-07	-2.15E-09	-0.35%
EQ111	ZWD - U2 - DWST	0.17g	1.85E-02	1.29E-05	0.00E+00	0.00%	1.44E-01	6.14E-07	-1.00E-12	0.00%
EQ02	ZL2 - STEAM GENERATORS	0.91g	1.03E-04	1.29E-05	-2.00E-09	-0.02%	1.32E-01	5.35E-07	-7.95E-08	-12.94%
EQ92	MSCV 722 DIAPHRAGM POV CNMT ISOLATION	0.97g	~0.00E+00	1.29E-05	0.00E+00	0.00%	9.34E-02	5.57E-07	-5.73E-08	-9.32%
EQ102	ZM6 - MCC-1-E10	0.2g	6.75E-03	1.29E-05	5.90E-09	0.05%	5.63E-02	6.13E-07	-1.64E-09	-0.27%
EQ14	ZAF - PPDWST (WT-TK-10)	0.29g	7.78E-02	1.20E-05	-9.45E-07	-7.33%	3.94E-02	6.11E-07	-4.04E-09	-0.66%
EQ03	ZL2 - MS&CV BLDG	1.23g	2.02E-05	1.29E-05	0.00E+00	0.00%	3.54E-02	5.93E-07	-2.15E-08	-3.50%

5.8 SPRA LOGIC MODEL AND QUANTIFICATION TECHNICAL ADEQUACY

The BVPS-1 SPRA risk quantification and results interpretation methodology were subjected to an independent peer review against the pertinent requirements in the ASME/ANS PRA Standard (Reference 4).

The peer review assessment, and subsequent disposition of peer review findings, is described in ***Appendix A***, and establishes that the BVPS-1 SPRA seismic plant response analysis is suitable for this SPRA application.

6.0 CONCLUSIONS

A seismic PRA has been performed for BVPS-1 in accordance with the guidance in the SPID. The BVPS-1 SPRA shows that the seismic CDF is 1.30×10^{-05} and the seismic LERF is 6.14×10^{-07}.

Further, no seismic hazard vulnerabilities were identified.

The updated PRA model, which includes the seismic PRA reflects the as-built, as-operated plant as of the freeze date of October 25, 2016 and includes the FLEX mitigation strategies equipment and procedure changes already installed and implemented. The PRA model provides insights and identifies the most important equipment to responding to a seismic event, but no seismic hazard vulnerabilities were identified. The seismic CDF and LERF are sufficiently low such that possible improvements or modifications to the plant are not considered necessary. In addition, the sensitivities presented in ***Table 5-20*** of this submittal show that postulated improvements that would increase the seismic capacity of the important components would not provide a significant reduction in risk.

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8.0 LIST OF ACRONYMS AND ABBREVIATIONS

ABS	ABSG CONSULTING INC.
AC	AIR CONDITIONING
ACI	AMERICAN CONCRETE INSTITUTE
AF	AMPLIFICATION FACTOR
AFW	AUXILIARY FEEDWATER
AISC	AMERICAN INSTITUTE OF STEEL CONSTRUCTION
AISX	ALTERNATE INTAKE STRUCTURE
ANS	AMERICAN NUCLEAR SOCIETY
AOV	AIR-OPERATED VALVE
ASCE	AMERICAN SOCIETY OF CIVIL ENGINEERS
ASME	AMERICAN SOCIETY OF MECHANICAL ENGINEERS
ATWS	ANTICIPATED TRANSIENT WITHOUT SCRAM (ALSO ATWT, ANTICIPATED TRANSIENT WITHOUT TRIP)
AXLB	AUXILIARY BUILDING
BE	BEST ESTIMATE
BVPS	BEAVER VALLEY POWER STATION
BVPS-1	BEAVER VALLEY POWER STATION, UNIT 1
BVPS-2	BEAVER VALLEY POWER STATION, UNIT 2
CABX	CHEMICAL ADDITION BUILDING
CCF	COMMON-CAUSE FAILURE
CDF	CORE-DAMAGE FREQUENCY
CDFM	CONSERVATIVE DETERMINISTIC FAILURE MARGIN
CET	CONTAINMENT EVENT TREE
CEUS	CENTRAL AND EASTERN UNITED STATES
CEUS-SSC	CENTRAL AND EASTERN UNITED STATES SEISMIC SOURCE CHARACTERIZATION
CMU	CONCRETE MASONRY UNIT
CNTB	CONTROL BUILDING
COV	COEFFICIENT OF VARIATION
CP	COGNITIVE PROBABILITY
CRDM	CONTROL ROD DRIVE MECHANISM
CTMT	CONTAINMENT
DAFW	DEDICATED AUXILIARY FEEDWATER
DBE	DESIGN BASIS EARTHQUAKE
DG	DIESEL GENERATOR
DGBX	DIESEL GENERATOR BUILDING
DOE	DEPARTMENT OF ENERGY
DWST	DEMINERALIZED WATER STORAGE TANK
EDG	EMERGENCY DIESEL GENERATOR
EL	ELEVATION
EPRI	ELECTRIC POWER RESEARCH INSTITUTE
ERF	EMERGENCY RESPONSE FACILITY
ERFS	EMERGENCY RESPONSE FACILITY SUBSTATION

ESEL	EXPEDITED SEISMIC EQUIPMENT LIST
ESEP	EXPEDITED SEISMIC EVALUATION PROCESS
ESFAS	ENGINEERED SAFETY FEATURE ACTUATION SYSTEM
F&O	FACTS AND OBSERVATIONS
FE	FINITE ELEMENT
FEM	FINITE-ELEMENT MODEL
FENOC	FIRSTENERGY NUCLEAR OPERATING COMPANY
FIRS	FOUNDATION INPUT RESPONSE SPECTRA
FLEX	DIVERSE AND FLEXIBLE MITIGATION STRATEGIES
FULB	FUEL HANDLING AND DECONTAMINATION BUILDING
FV	FUSSELL-VESELY
FVI	FUSSELL-VESELY IMPORTANCE
FT	FEET
FWS	FEEDWATER SYSTEM
GERS	GENERIC EQUIPMENT RUGGEDNESS SPECTRA
GIP	GENERIC IMPLEMENTATION PROCEDURE
GMM	GROUND MOTION MODEL
GMPE	GROUND MOTION PREDICTION EQUATION
GMRS	GROUND MOTION RESPONSE SPECTRA
HCLPF	HIGH CONFIDENCE OF A LOW PROBABILITY OF FAILURE
HCSCP	HAZARD-CONSISTENT STRAIN-COMPATIBLE PROPERTIES
HEP	HUMAN ERROR PROBABILITIES
HF	HIGH FREQUENCY
HFE	HUMAN FAILURE EVENTS
HHSI	HIGH-HEAD SAFETY INJECTION
HID	HAZARD INPUTS DOCUMENT
HRA	HUMAN RELIABILITY ANALYSIS
HVAC	HEATING, VENTILATION, AND AIR CONDITIONING
HX	HEAT EXCHANGER
HZ	HERTZ
IEEE	INSTITUTE OF ELECTRICAL AND ELECTRONICS ENGINEERS
IF	INTERVAL FREQUENCY
INTS	INTAKE STRUCTURE
IPEEE	INDIVIDUAL PLANT EXAMINATION FOR EXTERNAL EVENTS
ISLOCA	INTERFACING SYSTEMS LOCA
ISRS	IN-STRUCTURE RESPONSE SPECTRA
LB	LOWER BOUND
LERF	LARGE EARLY RELEASE FREQUENCY
LHSI	LOW-HEAD SAFETY INJECTION
LMSM	LUMPED-MASS STICK MODELS
LOCA	LOSS OF COOLANT ACCIDENT
LOOP	LOSS OF OFFSITE POWER
LOSP	LOSS OF OFFSITE POWER
LR	LOWER RANGE

LS-A	LIMIT STATE A
LS-C	LIMIT STATE C
LTAFW	LONG-TERM AFW
M&E	MECHANICAL AND ELECTRICAL
MAFE	MEAN ANNUAL FREQUENCY OF EXCEEDANCE
MCC	MOTOR CONTROL CENTER
MCR	MAIN COOLING RESERVOIR
MFW	MAIN FEEDWATER
MLOCA	MEDIUM LOCA
MOV	MOTOR-OPERATED VALVE
MSCV	MAIN STEAM CABLE VAULT
NEI	NUCLEAR ENERGY INSTITUTE
NEP	NON-EXCEEDANCE PROBABILITY
NFPA	NATIONAL FIRE PROTECTION ASSOCIATION
NPTX	NORTH PIPE TRENCH
NRC	NUCLEAR REGULATORY COMMISSION
NSSS	NUCLEAR STEAM SUPPLY SYSTEM
NTTF	NEAR-TERM TASK FORCE
NUREG	US NUCLEAR REGULATORY COMMISSION REGULATION
PDWS	PRIMARY PLANT DEMINERALIZED WATER STORAGE PAD AND ENCLOSURE
PGA	PEAK GROUND ACCELERATION
PIPETUNNEL	PIPE TUNNELS
PORV	PRESSURE-OPERATED RELIEF VALVE
POV	PNEUMATIC-OPERATED VALVE
PSD	POWER SPECTRAL DENSITY
PRA	PROBABILISTIC RISK ASSESSMENT
PSHA	PROBABILISTIC SEISMIC HAZARD ANALYSIS
PWR	PRESSURIZED WATER REACTOR
PZR	PRESSURIZER
RAW	RISK ACHIEVEMENT WORTH
RCBX	REACTOR CONTAINMENT
RCP	REACTOR COOLANT PUMP
RCS	REACTOR COOLANT SYSTEM
REJ	RUBBER EXPANSION JOINT
RHR	RESIDUAL HEAT REMOVAL
RIZZO	RIZZO ASSOCIATES
RLYB	SWITCHYARD RELAY HOUSE
RPS	REACTOR PROTECTION SYSTEM
RRS	REQUIRED RESPONSE SPECTRA
RSGB	ERF DIESEL GENERATOR BUILDING
RVT	RANDOM VIBRATION THEORY
RW	RIVER WATER
RWST	REFUELING WATER STORAGE TANK

SAP	PLANT DATABASE
SASSI	SYSTEM FOR ANALYSIS FOR SOIL-STRUCTURE-INTERACTION
SBO	STATION BLACKOUT
SCDF	SEISMIC CDF
SCE	SEISMIC CAPABILITY ENGINEER
SEL	SEISMIC EQUIPMENT LIST
SEWS	SEISMIC EVALUATION WORK SHEETS
SFGB	SAFEGUARDS BUILDING
SFP	SPENT FUEL POOL
SFR	SEISMIC FRAGILITY ELEMENT WITHIN ASME/ANS PRA STANDARD
SG	STEAM GENERATOR
SGTR	STEAM GENERATOR TUBE RUPTURE
SHA	SEISMIC HAZARD ANALYSIS ELEMENT WITHIN ASME/ANS PRA STANDARD
SHS	SEISMIC HAZARD SUBMITTAL
SI	SAFETY INJECTION
SLERF	SEISMIC LARGE EARLY RELEASE FREQUENCY
SLOCA	SMALL LOSS OF COOLANT ACCIDENTS
SMA	SEISMIC MARGIN ASSESSMENT
SOV	SOLENOID-OPERATED VALVE
SPID	SCREENING, PRIORITIZATION, AND IMPLEMENTATION DETAILS
SPR	SEISMIC PRA MODELING ELEMENT WITHIN ASME/ANS PRA STANDARD
SPRA	SEISMIC PROBABILISTIC RISK ASSESSMENT
SPRAIG	SEISMIC PROBABILISTIC RISK ASSESSMENT IMPLEMENTATION GUIDANCE
SPTX	SOUTH PIPE TRENCH
SRT	SEISMIC REVIEW TEAM
STOR	STOREROOM
SQSS-TK21	SURROUNDING SHIELD WALL FOR REFUELING WATER STORAGE TANK
SQUG	SEISMIC QUALIFICATION UTILITIES GROUP
SRT	SEISMIC REVIEW TEAM
SRVB	SERVICE BUILDING
SSC	STRUCTURES, SYSTEMS, AND COMPONENTS
SSE	SAFE SHUTDOWN EARTHQUAKE
SSEL	SAFE SHUTDOWN EQUIPMENT LIST
SSI	SOIL STRUCTURE INTERACTION
SSSI	STRUCTURE SOIL STRUCTURE INTERACTION
SWBX	SOLID WASTE BUILDING
SWGR	SWITCHGEAR
TDAFW	TURBINE-DRIVEN AFW
TK	TANK
TRBB	TURBINE BUILDING

TRS	TEST RESPONSE SPECTRA
TSCR	TRUNCATED SOIL COLUMN RESPONSE
UB	UPPER BOUND
UFSAR	UPDATED FINAL SAFETY ANALYSIS REPORT
UHRS	UNIFORM HAZARD RESPONSE SPECTRA
UHS	ULTIMATE HEAT SINK
UR	UPPER RANGE
USI	UNRESOLVED SAFETY ISSUE
VAC	VOLTS (ALTERNATING CURRENT)
VCT	VOLUME CONTROL TANK
VDC	VOLTS (DIRECT CURRENT)
V/H	VERTICAL-TO-HORIZONTAL
VPA_VPB	RIVER WATER VALVE PIT TRAIN
VSLOCA	VERY SMALL LOSS OF COOLANT ACCIDENTS
WTBX	WATER TREATMENT BUILDING
WUS	WESTERN UNITED STATES

PRA Model Top Event Descriptions:

A3	AUXILIARY FEEDWATER SYSTEM (NO AC POWER)
AA	FLEX ALTERNATE AFW PUMP
AF	AUXILIARY FEEDWATER SYSTEM
AG	2/3 SUPPLY FROM ACCUMULATOR (GENTRANS)
AL	SUPPLY FROM ACCUMULATOR (LLOCA)
AM	SUPPLY FROM ACCUMULATOR (MLOCA)
AO	EMERGENCY AC ORANGE TRAIN
AP	ALPHA MODE FAILURE
AS	AMSAC SIGNAL
AT	AUXILIARY FEEDWATER SYSTEM - SGTR
AW	AUXILIARY FEEDWATER SYSTEM - ATWS
AX	BOTH TRAINS OF AC POWER - CROSS-TIE
BC	BOTH TRAINS AC POWER - ORANGE AND PURPLE
BI	BASEMAT PENETRATION
BK	BLACK DIESEL POWER
BL	LARGE CONTAINMENT BYPASS
BP	EMER. AC TRAIN PURPLE
BV	EMER. SWGR VENTILATION
BY	CONTAINMENT BYPASSED
C1	CNMT FAILS PRIOR TO VESSEL BREACH
C2	CNMT FAILS AT VESSEL BREACH
C3	LATE CONTAINMENT FAILURE DUE TO BURN
C4	LONG TERM CNMT OVERPRESSURIZATION
CC	REACTOR PLANT COMPONENT COOLING
CD	OPERATOR INITIATES COOLDOWN/ DEPRESS.

CE	CNMT FAILS DUE TO EARLY H2 BURN
CG	LEVEL 1 OR LEVEL 2 SEQUENCE GROUP
CI	CONTAINMENT ISOLATION
CP	FAILURE TO COOL DEBRIS IN VESSEL
CT	TURBINE PLANT COMPONENT COOLING
D3	DC BUS NO 3 (ORANGE)
D4	DC BUS 4 (PURPLE)
D5	125V DC SWBD 1-5 POWER (STATION)
DC	FAILURE TO COOL DEBRIS EX-VESSEL
DF	DEDICATED AUX FEEDWATER
DO	125V DC BATTERY 1-1 (ORANGE)
DP	DC BUS 2 (PURPLE)
DX	125V DC 1-1 AND 1-2 SUPPLY - DUMMY TOP
DY	125V DC 1-3 AND 1-4 SUPPLY - DUMMY TOP
FA	MAIN FEEDWATER FAILS-ATWS
GE	FLEX 480V GENERATOR
GL	PORTABLE AC GENERATOR FOR SG LEVEL INSTR
H3	LATE BURN OF COMBUSTIBLE GASES
HC	COLD LEG INJECTION FROM HHSI
HE	HYDROGEN BURN WITHIN 4 HRS OF VB
HH	HIGH HEAD SAFETY INJECTION
HL	COLD LEG INJECTION PATH (LLOCA)
HM	COLD LEG INJECTION PATH (MLOCA)
HR	LOW HEAD TO HIGH HEAD CROSS TIE FOR RECIR.
IA	STATION INSTRUMENT AIR IA
IB	VITAL BUS CHANNEL III (BLUE)
IC	CONTAINMENT INSTRUMENT AIR
IO	VITAL BUS I(RED) & III(BLUE)
IP	INDUCED RCS HOT LEG OR SURGE LINE RUPTURE
IR	VITAL BUS CHANNEL I (RED)
IS	TEMPERATURE INDUCED SG TUBE RUPTURE
IW	VITAL BUS CHANNEL II (WHITE)
IX	PURPLE VITAL BUSES II & IV
IY	VITAL BUS CHANNEL IV (YELLOW)
L1	LARGE CONTAINMENT FAILURE PRIOR TO VB
L2	LARGE CONTAINMENT FAILURE @ VB
L3	LARGE LATE CONTAINMENT FAILURE
L4	LARGE LONG TERM CNMT OVERPRESSURIZATION FAILURE
LA	LOW HEAD SAFETY INJECTION TRAIN A
LB	LOW HEAD SAFETY INJECTION TRAIN B
LC	COLD LEG INJECTION FROM LHSI
LD	LOAD SHED
LE	LARGE CNMT FAILURE FROM EARLY H2 BURN

LL	COLD LEG INJECTION PATH-LLOCA
LM	COLD LEG INJECTION PATH - MLOCA
LO	COLD LEG INJECTION FROM LHSI TRAINS A&B
LP	BOTH TRAINS LOW HEAD SAFETY INJECTION PUMPS
LQ	COLD LEG INJECTION PATHS-MLOCA (HM & LM)
LR	LOW HEAD TRANSFER TO HOT LEG RECIRC
LS	INDUCED PORV LOCA
M1	480V MCCS (ORANGE) - E03 AND E11
M2	480V MCCS (PURPLE) - E04 AND E12
M3	480V MCC (ORANGE) - E05
M4	480V MCC (PURPLE) - E06
M5	ORANGE MCCS - E9 AND E13
M6	PURPLE MCCS - E10 AND E14
MA	WATER MAKEUP TO WT-TK-10 AND AFW PUMPS
ME	HIGH PRESSURE MELT EJECTION
MF	MAIN FEEDWATER SYSTEM
MS	MAIN STEAM ISOLATION
MU	MAKEUP TO RWST
NA	NORMAL 4KV BUS 1A
ND	NORMAL 4KV BUS 1D
NM	MELT DURING INJECTION PHASE
NR	RECIRCULATION REQUIRED FOLLOWING INJECTION
NX	NORMAL 4KV BUSSES 1A & 1D
OA	EMERGENCY BORATION - ATWS
OB	BLEED AND FEED COOLING
OC	OPERATOR TRIPS RCPS DURING LOSS OF SEAL COOLING
OCL	OPERATOR TRIPS RCPS DURING SEAL LOCA (30)
OD	OPERATOR INITIATES DEPRESSURIZATION
OF	OPERATOR FAILS TO ALIGN FEEDWATER
OG	OFFSITE GRID
OL	OPERATOR RESTORES COOLING TO SCRUB FAULTED SGTR
OP	OPERATOR PREMATURELY TERMINATES SI
OR	OPERATOR'S ALIGNMENT FOR RECIRCULATION FAILS
OS	OPERATOR FAILS TO INITIATE SI
OT	OPERATOR MANUALLY TRIPS REACTOR
PA	PRESSURE RELIEF FOR ATWS FAILS
PI	PRESSURIZER PORVS FAIL TO BE ISOALTED
PK	ATWS TOP EVENT PRESSURE RELIEF
PL	REACTOR POWER LEVEL FOLLOWING ATWS
PR	PRIMARY RELIEF FAILS
PT	PROPANE TANK FARM DURING EARTHQUAKE
QA	QUENCH SPRAY TRAIN A
QB	QUENCH SPRAY TRAIN B

QC	QUENCH SPRAY TRAINS A & B
R1	RIVER WATER TRAIN A TO RSS
R2	RIVER WATER TRAIN B TO RSS
R3	RIVER WATER TRAINS A AND B TO RSS
RA	OUTSIDE RECIRC SPRAY TRAIN A
RB	OUTSIDE RECIRC SPRAY
RC	OUTSIDE RECIRC SPRAY TRAINS A & B
RE	ELECTRIC POWER RECOVERY
RI	OPERATOR MANUALLY INSERTS RODS - ATWS
RL	RCP SEAL LOCA
RP	RCS PRESSURE AT VESSEL BREACH
RR	RESIDUAL HEAT REMOVAL
RS	INSIDE RECIRC SPRAY
RT	REACTOR TRIP
RW	RWST (QS-TK-1) FAILS
SA	SOLID STATE PROTECTION SYSTEM TRAIN A
SB	SOLID STATE PROTECTION SYSTEM TRAIN B
SD	SHUTDOWN SEAL ACTUATES
SE	RCP SEAL INJECTION - TOP SE
SL	SECONDARY LEAKAGE TO ATMOSPHERE
SM	WATER SUPPLY FROM CNMT SUMP & RSS COMMON CAUSE FAILURE
SP	REACTOR COOLANT PUMP SEAL LOCA
SS	NO MELT FROM LEVEL 1
SW	OPERATORS FAIL TO SWAP BATTERY TRAINS (FLEX, LOAD SHED)
SX	SOLID STATE PROTECTION SYSTEM TRAIN A & B
TB	RCP THERMAL BARRIER COOLING
TR	PRESSURE INDUCED SG TUBE RUPTURE
TT	TURBINE TRIP FAILURE
VA	LHSI TRAIN A SUCTION FROM CONTAINMENT SUMP
VB	BOTH LHSI CONTAINMENT SUMP SUCTION FAILS
VC	BOTH LHSI CNMT SUMP SUCTION FAILS
VI	VESSEL INTEGRITY, ATWS
WA	RW AND AUX RW TO HEADER A
WB	RIVER WATER HEADER B
WC	RIVER WATER SYSTEM BOTH HEADERS A & B
WM	MAKEUP TO RWST GIVEN LEAKAGE THRU SECONDARY
XL	HHSI AND LHSI PATH TO COLD LEGS - LLOCA
XT	STATION AC POWER CROSS TIE
Z2S	UNIT 2 SUPPORT FOR XT- NORMAL SWGR, U2 EDGS
ZAC	EMERG. AC - ORANGE AND PURPLE 4KV 480V
ZAF	AFW - PPDWST OR ALL 3 PUMPS
ZAI	RIVER WATER FROM ALTERNATE INTAKE STRUCTURE

ZAT	TURBINE DRIVEN AFW PUMP
ZBV	EMER. SWGR HVAC - FANS&TEMP SWITCHES&DAMPERS
ZBW	BLOCK WALLS SERVICE BLDG 713'
ZCC	PCCW - MEJ REJ PUMPS & HXS; SURGE TANK
ZCI	CONTAINMENT ISOLATION VALVES
ZCP	CONTAINMENT PENETRATIONS
ZD5	DC TRAIN 1-5 - BATTERY & CHARGER& SWBD
ZDC	EMERG. DC - SWBD BATTERIES CHARGER
ZDG	EDGS - FO HVAC MOVS; RECEIVERS
ZDW	U2 DWST
ZGL	PORTABLE GENERATOR
ZHH	HHSI - PUMPS MOVS RW STRAINERS
ZHR	HIGH PRESSURE RECIRC - MOVS
ZIM	RCBX 727 MOV INSIDE CNMT
ZIO	INSTRUM - VITAL BUS INVERTER; XMFR
ZIP	RCBX 718 DIAPHRAGM POVS INSIDE CNMT
ZIS	RCBX 718 SOV INSIDE CNMT ISO
ZL1	DIRECT CORE DAMAGE
ZL2	DIRECT CD AND LERF - RCBX; SFGD;MSCV;SGS
ZLK	SMALL RCS LOCAS
ZLP	LHSI TRAINS
ZM5	MCC E5 AND E6 CONTACTORS
ZM6	MCC-1-E10
ZMA	NORMAL MAKEUP FAILS (NO SBO)
ZMO	QSS&LHSI MOVS SFGD BLDG 747'
ZMS	MSIV FTC
ZMU	MAKEUP TO RWST - PUMPS FILTERS
ZO3	CONTROL ROOM PANELS
ZO4	CONTROL ROOM CEILING
ZOB	PZR PORVS& PSVS&BLOCK VALVES AS-IS
ZOG	OFFSITE POWER - NON-SEISMIC SWGR
ZOM	MSCV 722 MOV OUTSIDE CNMT
ZOP	MSCV 722 DIAPHRAGM POVS OUTSIDE CNMT
ZOS	MSCV 722 SOV OUTSIDE CNMT ISO
ZPN	RELAY PANELS IN SRVB 713'
ZPT	PROPANE TANK FARM
ZQS	QSS - PUMPS
ZR1	RELAY CHATTER - EDG BREAKERS DF BUS
ZR2	RELAY CHATTER - PUMPS DF BUS
ZR3	RELAY CHATTER - EDG BREAKERS AE BUS
ZR4	RELAY CHATTER - PUMPS AE BUS
ZRR	RHR - MOVS& PUMPS&HXS

ZRS	RECICULATION SPRAY - PUMPS&HXS&HEADER
ZRV	ATM & RESIDUAL HEAT RELEASE VALVES
ZRW	REFUELING WATER STORAGE TANK
ZSA	SSPS (SA AND SB)
ZSM	CONTAINMENT SUMP PASSES DEBRIS
ZSV	SG SAFETY RELIEF VALVES
ZTD	AFW - TURBINE-DRIVEN PUMP STM VALVES
ZTX	TURBINE BUILDING
ZVS	VERY SMALL LOCA
ZWC	ALL RIVER WATER - PUMPS REJS; HVAC DUCTS
ZX	NON-SEISMIC INITIATING EVENT (SEISMIC TREE)
ZY	NO SEISMIC FAILURES (SUPPORT TREE)
ZZ	NO SEISMIC FAILURES (GENTRANS TREE)

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 Beaver Valley Power Station (BVPS)-1 probabilistic risk assessment (PRA) was subjected to an independent peer review against the pertinent requirements in Part 5 of the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA Standard (Reference 4). The peer review assessment, and subsequent disposition of peer review findings, is summarized here (for the final report, see Reference 6). The scope of the review encompassed the set of technical elements and supporting requirements (SR) for the seismic hazard analysis (SHA), seismic fragilities (SFR), and seismic PRA modeling (SPR) elements for seismic core damage frequency (CDF) and large early release frequency (LERF). The peer review therefore addressed the set of SRs identified in Table 6-4 through Table 6-6 of the Screening, Prioritization, and Implementation Details (SPID) (Reference 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 (Reference 16) and the requirements in Section 1-6 of the ASME/ANS PRA Standard (Reference 4), and presents the significant results of the peer review.

The BVPS Units 1 and 2 SPRA peer review was conducted during the week of December 1, 2014, at the FirstEnergy Nuclear Operating Company (FENOC) offices in Akron, Ohio. As part of the peer review, a walkdown of portions of BVPS Units 1 and 2 was performed on December 1, 2014, by two members of the peer review team who have the appropriate Seismic Qualification Utilities Group (SQUG) training.

A.2. Summary of the Peer Review Process

The peer review was performed against the requirements in Part 5 (Seismic) of Addenda B of the PRA Standard (Reference 4), using the peer review process defined in NEI 12-13 (Reference 5). The review was conducted over a four-day period, with a summary and exit meeting on the evening of the fourth day.

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

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

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

As part of the review team's assessment of capability categories, F&Os are prepared. There are three types of F&Os defined in (Reference 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.

Section 5 of the ASME/ANS PRA Standard contains a total of 77 SRs under three technical elements. Three (3) of the supporting requirements were judged to be not applicable, and therefore the remaining 74 SRs were reviewed.

A.3. Peer Review Team Qualifications

The review was conducted by Dr. Andrea Maioli and Mr. Kenneth Kiper of Westinghouse, Dr. Martin McCann of Jack R. Benjamin & Associates, Dr. Bob Youngs of AMEC, Mr. Steve Eder of Facility Risk Consultants, Mr. Nathan Barber of Pacific Gas & Electric, Mr. Deepak Rao of Entergy, Dr. Se-Kwon Jung of Duke Energy, and Mr. Don Moore of Southern Company. Appendix D of the peer review report (Reference 6) contains the resumes for the reviewers. Reference 6 Table 2-2 shows the review assignments for each reviewer.

Dr. Andrea Maioli, the team lead, has over 10 years' experience at Westinghouse in the nuclear safety area generally and seismic PRA specifically. He has served as lead engineer for a number of seismic PRA and seismic margin studies for existing and new nuclear power plants.

Dr. Martin McCann was the lead for the SHA technical element. He has 30 years' experience in engineering seismology including site response analysis, specification of ground motion. He was assisted in the hazard review by Dr. Bob Youngs, an internationally-recognized expert in seismology and earthquake hazard assessment.

Mr. Stephen Eder was the lead for the seismic fragility analysis (SFR) technical element. Mr. Eder has more than 30 years' experience in the fields of natural hazards risk assessment, seismic fragility analysis, structural performance evaluation, and retrofit design. He was assisted by Dr. Se-Kwon Jung and Mr. Donald Moore. Mr. Moore has over 45 years of experience in specialized technical positions and supervisory positions in the field of structural engineering with specific emphasis on seismic analysis and design, seismic risk assessments, and seismic qualification of equipment and subsystems. Dr. Jung has over 10 years' experience in the field of civil and structural engineering with focus on fragility evaluation in support to seismic PRAs.

Mr. Ken Kiper was the lead for the System Response (SPR) technical element. Mr. Kiper joined Westinghouse as a Technical Manager after a 31-year career in Seabrook Station. He has experience in virtually every aspect of PRA modeling and applications, including upgrading and maintaining the RISKMAN Seabrook seismic PRA. He was assisted by Mr. Nathan Barber and Mr. Deepak Rao. Mr. Barber has more than 12 years' experience in multiple aspects of PRAs; he is the lead for the Diablo Canyon seismic PRA RISKMAN model update and maintenance. Mr. Rao has 31 years' experience in essentially every aspects of PRA.

Two working observers (Boback Torkian, Enercon and Tommy John, Dominion) supported the review of the SPR and SFR technical elements. Any observations and findings these working observers generated were given to the peer review team for their review and "ownership." As such, Mr. Torkian and Mr. John assisted with the review but were not formal members of the peer review team.

None of the peer review team members had any involvement in the development of the BVPS-1 SPRA. The peer review team members met the peer reviewer independence criteria in NEI 12-13 (Reference 5).

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 Standard, the frequency of occurrence earthquakes at the site was based on a site-specific probabilistic seismic hazard analysis (PSHA). The Senior Seismic Hazard Analysis Committee (SSHAC) process of conducting a PSHA was used to develop the seismic source characterization (SSC) and the ground motion modeling (GMM) inputs to the analysis. The SSC inputs to the PSHA are based on the recently completed central and eastern U.S. (CEUS) seismic source model. The ground motion model inputs to the PSHA are based on the CEUS ground motion update project. The requirements of the SSHAC process satisfy the requirements of the standard for data collection and use of a structured expert elicitation process. The SSHAC process describes a process and minimum technical requirements to complete a PSHA. The "SSHAC level" of a seismic hazard study ensures that data, methods, and models supporting the PSHA are fully incorporated and that uncertainties are fully considered in the process at a sufficient depth and detail necessary to satisfy scientific and regulatory needs. The level of study is not mandated in the Standard; however, both the SSC and the GMM parts of the PSHA were developed as a result of SSHAC Level 3 analyses. In the case of the GMM, a SSHAC Level 2 analysis was carried out to update a prior Level 3 study. These Level 3 studies satisfy the requirements of the Standard.

As a first step to performing a PSHA, the Standard requires an up-to-date database, including regional geological, seismological, geophysical data, and local site topography, and a compilation of surficial geologic and geotechnical site properties. These data include a catalog of relevant historical, instrumental, and paleoseismic information within 320-km of the site. This data collection effort was carried out as part of the CEUS and GMM projects that were the basis for the inputs to the Beaver Valley PSHA. To ensure that the database of information that is the basis for the PSHA is up-to-date, the PSHA analysts did not systematically conduct a review to identify and gather new geological, seismological, or geophysical data available since the completion of the CEUS-SSC study or information at a level of detail that was not considered in the CEUS-SSC regional study that would indicate there should be new seismic sources added to the SSC model or changes to existing sources.

While a systematic review and update effort was not carried out, the PSHA analysts did gather data to update the earthquake catalog to assess whether there was new information since the completion of the CEUS-SSC project that should be used to update the seismicity parameters. A subjective review of the updated catalog was conducted to conclude that an update to the seismicity parameters was not required.

As part of the CEUS-SSC model sources potentially damaging earthquakes that could occur in the CEUS were modeled. This includes all distributed seismic sources within 640 km and all Repeated Large Magnitude Earthquake (RLME) sources within 1,000 km of the BV site. In the implementation of the CEUS model for the Beaver Valley site, all seismic sources in the CEUS model were included in the PSHA. By including all the CEUS seismic sources in the analysis, the contribution of "near-" and "far-field" earthquake sources to ground motions at Beaver Valley were considered.

The Davis-Besse peer review identified the fact that the PSHA software that was used to perform the probabilistic hazard quantification did not perform the uncertainty analysis correctly. This error was not corrected for the Beaver Valley PSHA; therefore, the uncertainty results are not correct in this analysis as well. This error does not impact the estimate of the mean hazard, but it does affect the estimate of the uncertainty in the PSHA results. Consequently, the PSHA inputs to the SPRA uncertainty quantification are incorrect.

The SHA for the Beaver Valley site took into account the effects of local site response. However, the review team did not find adequate documentation to support the site-specific velocity profile used in the analysis. Also, because of the limited site-specific data, the study could not properly account for velocity uncertainties as required by the standard. The review also noted that aleatory and epistemic uncertainties in the site response were not separately combined with the uncertainty in the rock seismic hazard results. As a result, the uncertainty in the soil site hazard results is likely underestimated.

The Standard requires that spectral shapes be based on a site-specific evaluation taking into account the contributions of deaggregated magnitude-distance results of the PSHA. The PSHA fully accounted for the "near-" and "far-field" source spectral shapes.

The Standard requires that sensitivity calculations be performed to document the models and parameters that are the primary contributors to the site hazard. The PSHA documentation does provide certain information such as magnitude-distance deaggregation plots that provide insight into contributors to the site hazard. However, the PSHA documentation does not provide the results of a systematic sensitivity analysis that evaluates the importance and sensitivity of key parameters to the results. As a result this requirement was not met.

As required by the Standard, a screening analysis was performed to assess whether in addition to the vibratory ground motion, other seismic hazards, such as fault displacement, landslide, soil liquefaction, or soil settlement, need to be included in the seismic PRA. The review identified a number of areas where further information should be provided to support the conclusion that other seismic hazards can be screened out. Because of the limitations in the review and screening of other hazards, SHA-I2 is at this time identified as not MET, pending the resolution of the issues identified in SHA-I1. This SR can be non-applicable if all the other hazards are indeed confirmed as screened out, or not met if some hazard needs to be retained.

Both the aleatory and epistemic uncertainties have been addressed in characterizing the seismic sources. In addition, uncertainties in each step of the hazard analysis were propagated and displayed in the final quantification of hazard estimates for the Beaver Valley site. As noted above, the PSHA software that was used to perform the hazard calculations implements an approach for the propagation of the uncertainties in the analysis that is not correct. As a result, the uncertainty in the seismic hazard is not properly quantified.

In summary, the PSHA performed for the BVPS is based on the CEUS and GMM regional studies which are SSHAC Level 3 efforts. There are a couple of instances where the standard is not met, including a computational issue with the PSHA software that impacts the uncertainty analysis. The PSHA is well documented which supports the review process and its future use by FENOC.

SFR

The Standard requires that all the structures, systems, and components (SSC) that play a role in the seismic PRA be identified as candidates for subsequent seismic fragility evaluation. This was performed through the development of the Walkdown Seismic Equipment List (SEL). As permitted by the Standard, extremely seismically rugged and seismically insensitive items in the list were screened out; i.e., no seismic fragility evaluation is required for these items. Additional high seismic capacity screening was performed for systems and components using the Electric Power Research Institute (EPRI) seismic margins screening tables. As required by the Standard, anchorage adequacy was verified when generic functional capacity was used. Some of the items with 0.50g based generic capacity ended up being top contributors to CDF. For these cases, no additional justification for use of the generic fragilities was provided as required by the Standard.

The Standard requires that the seismic fragility evaluation be based on realistic seismic response that the SSCs experience at their failure levels. The building response spectra were developed and then subsequently utilized in the evaluation of seismic fragilities. New 3-D building models were developed for all structures and used for this purpose. However, the review team noted that the modeling methods and the performance objective for the building response analysis were suitable for the calculation of fragilities for equipment and relays (based on the Conservative Deterministic Failure Margin [CDFM] approach), but not realistic for the calculation of fragilities for the building structures (based on the separation of variables approach). The review teams also noted that simplifying assumptions used in the soil-structure interaction analyses of buildings were not fully justified and that sensitivity studies or other more detailed evaluation may be warranted.

A series of walkdowns, focusing on the anchorage, lateral seismic support, functional characteristics, and potential systems interactions were conducted and documented appropriately in support of the fragility analysis. The walkdowns also evaluated the potential for seismic-induced fires and floods, and found no hazard sources. The walkdown observations were subsequently incorporated in the seismic fragility evaluations. However the review team noted some inconsistencies between the configurations assumed for the anchorage fragility calculations and actual field conditions, which resulted in excess conservatism.

The SPRA identifies the relevant failure modes for the SSCs through a review of plant design documents, earthquake experience data, and walkdowns. Subsequently, seismic fragility evaluations were performed for the critical failure modes of the SSCs. The review team noted however that the failure modes, analytical assumptions, and associated capacities assigned to certain SSCs including relay fragilities have conservative bias and are thus not realistic.

The Standard requires that the seismic fragility parameters be based on plant-specific data supplemented as needed by earthquake experience data, fragility test data, and generic qualification test data. The review team found that this requirement was satisfied, but noted as described above that certain fragilities are not realistic and that the basis for use of generic lower bound fragilities should be revisited in certain cases.

In conclusion, seismic fragilities were developed for structures, systems, and components (SSC's) associated with the SEL. This included development of new building models and performance of site-specific response analyses for generation of in-structure response spectra. Component screening was performed using available industry guidance at 0.50g. [Note that although the peer review report says 0.5g, the final screening value for BVPS-1 was increased to 0.6g during the process of model refinements.] Thorough walkdowns were performed and documented. Many detailed calculations were performed to assess SSC fragility, and the documentation was comprehensive. However unrealistic assumptions were noted in different steps of the evaluation process, resulting in fragilities with conservative bias.

SPR

The plant-response model developed for the BVPS-1 SPRA represents a state-of-the-art model and documentation that fully meet the requirements of the Standard. The model, as reviewed, represents a final-draft version, which will need to be finalized along with the standard quantification steps and revised documentation.

The SPRA model was developed by modifying the Full Power Internal Events (FPIE) PRA model to incorporate specific aspects of seismic analysis that are different from the FPIE. The logic model appropriately includes seismic-caused initiating events and other failures including seismic-induced SSC failures, non-seismic-induced unreliability and unavailability failure modes (based on the FPIE model), and human errors.

The human reliability analysis (HRA) modeling and documentation was recognized as a best practice. This HRA used the EPRI HRA calculator and adjusts performance shaping factors (PSFs) to account for four levels of earthquake intensity. Specific adjustments were made to the delay time and execution time, to stress, and to cognitive work load. These adjustments were implemented through the HRA calculator for each action modeled in the SPRA.

The use of RISKMAN in the seismic model development and quantification fully met the challenges of integrating a seismic risk model. A significant number of sensitivities were performed to understand the impact of the various modeling and screening assumptions. In these aspects, the quantification of the BVPS-1 SPRA is judged to meet the PRA Standard.

It is apparent that the quantification process was used to inform as appropriate the fragility aspects; e.g., selection of the screening values and of the specific fragility items to be refined. The peer review team concluded that the BVPS-1 SPRA has an appropriate level of resolution for CDF evaluation, but that conservative fragilities may be masking some of the LERF contributors.

The FENOC PRA team went beyond the current state-of-practice in addressing seismic-induced fires and, especially, seismic-induced floods, leveraging the existing fire and floods PRA for a more systematic assessment of these scenarios. This was recognized as a best practice by the peer review team.

In conclusion, the seismic PRA model integrates the seismic hazard, the seismic fragilities, and the systems-analysis aspects appropriately to quantify CDF and LERF. The seismic PRA analysis was extensively documented in a manner that facilitates applying and updating the SPRA model.

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

Table A-1 presents a summary of the SRs graded as not met or not Capability Category II, and the disposition for each. ***Section A.10*** presents summary of the Finding F&Os and the disposition for each.

TABLE A-1
SUMMARY OF SRS GRADED AS NOT MET OR CAPABILITY CATEGORY I FOR SUPPORTING REQUIREMENTS COVERED BY THE BVPS-1 SPRA PEER REVIEW

SR	Assessed Capability Category	Associated Finding F&Os	Disposition to Achieve MET or Capability Category II
SHA			
SHA-F1	CC-I	2-1, 2-2	Associated F&Os have been resolved. SR is judged to now achieve CC-II.
SHA-F2	Not MET	2-3	Associated F&Os have been resolved. SR is judged to be met.
SHA-I2	Not MET	2-26, 2-27, 2-28, 2-29, 2-31	Associated F&Os have been resolved. SR is judged to be met.
SHA-J3	Not MET	2-30	Associated F&Os have been resolved. SR is judged to be met.
SFR			
SFR-A2	CC-I	4-6, 4-13, 4-16	F&Os 4-13 and 4-16 have been resolved as prescribed by the peer review team. For F&O 4-6, further justification has been provided as to why the generic fragilities described in the F&O are acceptable for use, per HLR-SFR-F, as directed in SFR-A2. This is demonstrated through the use of sensitivity studies. See the "Plant Response or Disposition" section of this F&O in Section A.10. This SR is judged to now achieve CC-II.
SPR			
[None]	N/A	N/A	N/A

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

The set of SR from the ASME/ANS PRA Standard (Reference 4) that is identified in Tables 6-4 through 6-6 of the SPID (Reference 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 BVPS-1 SPRA model meets the expectations for PRA scope and technical adequacy as presented in RG 1.200, Revision 2 (Reference 16 as clarified in the SPID (Reference 2.

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

- Summary of the SHA (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 Revision 2 is available if required to facilitate the U.S. Nuclear Regulatory Commission (NRC) staff's review of this submittal.

The BVPS-1 SPRA reflects the as-built and as-operated plant as of the cutoff date for the SPRA, October 25, 2016. This includes outage modifications, non-outage modifications, and other configuration control items. There are no permanent plant changes that have not been reflected in the SPRA model.

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

The Owners Group performed a full scope peer review of the BVPS-1 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* (Reference 4) and RG 1.200 (Reference 16) in during the week of June 6, 2011. This review documented findings for all 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 peer review of the BVPS SPRA in December 2014. 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 BVPS-1 seismic PRA model is of good quality and integrates the seismic hazard, the seismic fragilities, and the systems-analysis aspects appropriately to quantify CDF and LERF. The general conclusion of the peer review was that the BVPS-1 SPRA is judged to be suitable for use for risk-informed applications.

- ***Table A-1*** in Section A.5 provides a summary of the disposition of SRs judged by the peer review to be not met, or not meeting Capability Category II.
- ***Section A.10*** provides a summary of the disposition of the open SPRA peer review findings.
- ***Table A-2*** provides an assessment of the expected impact on the results of the BVPS-1 SPRA of those SRs and peer review Findings that have not been fully addressed.

TABLE A-2
SUMMARY OF IMPACT OF NOT MET SRS AND OPEN PEER REVIEW FINDINGS

F&O	Summary of Issue Not Fully Resolved	Impact on SPRA Results
N/A	N/A	This table is not applicable, as all F&Os listed in Table A-1 have been fully dispositioned in Section A.10. It is judged by the utility that the associated SRs now achieve at least CC-II (or MET, for SRs in which no capability category is assigned), and that no further action is needed to address any SPRA F&Os. This table is retained to maintain the numbering order from the template.

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

The PRA Standard (Reference 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 (Reference 88) and EPRI 1016737 (Reference 74) provide guidance on assessment of uncertainty for applications of a PRA. As described in NUREG-1855 (Reference 88), 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 BVPS-1 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 BVPS-1 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. These important modeling assumptions were considered when identifying sensitivity cases for quantification.
- 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. For example, the current seismic PRA only considers scenarios initiated from power operation, not from shutdown conditions. A few specific issues of PRA completeness were identified in the SPRA peer review and the associated F&Os were addressed and resolved.

A summary of potentially important sources of uncertainty in the BVPS-1 SPRA is listed in ***Table A-3***.

TABLE A-3
SUMMARY OF POTENTIALLY IMPORTANT SOURCES OF UNCERTAINTY

PRA ELEMENT	SUMMARY OF TREATMENT OF SOURCES OF UNCERTAINTY PER PEER REVIEW	POTENTIAL IMPACT ON SPRA RESULTS
Seismic Hazard	The BVPS-1 SPRA peer review team noted that both the aleatory and epistemic uncertainties have been addressed in characterizing the seismic sources. In addition, uncertainties in each step of the hazard analysis were propagated and displayed in the final quantification of hazard estimates for the Beaver Valley site. As noted above, the PSHA software that was used to perform the hazard calculations implements an approach for the propagation of the uncertainties in the analysis that is not correct.	The BVPS-1 SPRA peer review team noted that the uncertainty in the seismic hazard is not properly quantified. In response, associated F&Os were addressed and resolved. The seismic hazard reasonably reflects sources of uncertainty.
Seismic Fragilities		Section 5.7.1.6 of the main report presents sensitivities performed which adjust the HCLPFs of the top seismic SSC failures to assess the impact of assumptions and uncertainties in the fragility calculations.
Seismic PRA Model		Section 5.7.1 of the main report presents sensitivities performed that assesses the impact of assumptions and sources of uncertainties in the SPRA model.

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

The BVPS-1 SPRA reflects the as-built and as-operated plant as of the cutoff date for the SPRA, which was *October 25, 2016*. This includes outage modifications, non-outage modifications, and other configuration control items. ***Table A-4*** lists significant plant changes subsequent to this date and provides a qualitative assessment of the likely impact of those changes on the SPRA results and insights.

TABLE A-4
SUMMARY OF SIGNIFICANT PLANT CHANGES SINCE SPRA CUTOFF DATE

DESCRIPTION OF PLANT CHANGE	IMPACT ON SPRA RESULTS
N/A	This table is not applicable, as there have been no significant plant changes from the date of SPRA modeling cutoff. This table is retained to maintain the numbering order from the template.

A.10. Summary of Finding F&Os and Disposition Status

Note that some findings only pertain to Unit 2 and are noted that way in the details of the finding. The dispositions of these findings are judged to have resolved the issues identified and thus the seismic PRA meets Capability Category II or higher for all supporting requirements in Section 5 of the ASME/ANS PRA standard (Reference 4). It is believed that the standard bounds the SPID, however it has been identified that the SPID contains specific guidance that differs from the Standard or expands it in 16 different areas. These 16 topics are specifically addressed below. Based on this and the results of the peer review along with the resolutions to the findings the SPRA is judged to meet or exceed the SPID (Reference 2).

Topic 1: Seismic Hazard (SPID Section 2.1, Section 2.2, and Section 2.3)

The PSHA submitted to the NRC in response to the NTTF 2.1 50.54(f) letter in March of 2014 has been updated following the peer review for use in the final SPRA model. The guidance presented in the SPID (Reference 2) was followed for developing the PSHA update. The PSHA update is described in Section 3.1.1 of this report.

Topic 2: Site Seismic Response (SPID Section 2.4)

The site response analysis submitted to the NRC in response to the NTTF 2.1 50.54(f) letter in March of 2014 has been updated following the peer review for use in the final SPRA model. The guidance presented in the SPID (Reference 2) was followed for developing the site response analysis update. The site geotechnical model used for the site response analysis is described in Section 3.1.1.2 while the site response analysis results are described in Section 3.1.1.3 of this report.

Topic 3: Definition of the Control Point for the SSE-to-GMRS-Comparison Aspect of the Site Analysis (SPID Section 2.4.2)

The PSHA and site response analysis are used to derive Foundation Input Response Spectra (FIRS) at several foundation elevations for critical structures to support the development of fragilities. Section 3.1.1.2 summarizes the elevations for the FIRS. The SPRA does not explicitly derive a GMRS. The GMRS for the site is consistent with the SSE control point is defined in the Updated Final Safety Analysis Report (UFSAR) (Reference 29). Section 3.1.2 of this report compares the GMRS submitted to the NRC in response to the NTTF 2.1 50.54(f) letter in March of 2014 with the FIRS for the Reactor Containment Building foundation elevation. The FIRS are derived consistent with NRC Interim Staff Guidance as described in Section 3.1.1.2 of this report.

Topic 4: Adequacy of the Structural Model (SPID Section 6.3.1)

Entirely new finite element structural models were developed for the SPRA which meet the intents of Criteria 1 through 7 in the SPID (Reference 2) Section 6.3.1. Details on the structural models can be found in Section 4.3 of this submittal.

Topic 5: Use of Fixed-Based Dynamic Seismic Analysis of Structures for Sites Previously Defined as "Rock" (SPID Section 6.3.3)

Fixed-base dynamic seismic analysis of structures was not used for the SPRA since BVPS is characterized as a soil site.

Topic 6: Use of Seismic Response Scaling (SPID Section 6.3.2)

Seismic response scaling was not used for the SPRA.

Topic 7: Use of New Response Analysis for Building Response, ISRS, and Fragilities

New response analysis is not specifically addressed in the SPID for use in developing In-Structure Response Spectrum (ISRS) and fragilities. The requirements for new analysis are found in the standard under supporting requirements SFR-C2, C4, C5, and C6. The peer review team reported all four of these requirements are either met for Capability Category II or are not applicable for the BVPS-1 SPRA. Furthermore the FIRS site response is developed with appropriate building specific soil velocity profiles and captures the uncertainty and variability in material dynamic properties as described in Section 3.1.1.2 of this submittal.

Topic 8: Screening by Capacity to Select SSCs for Seismic Fragility Analysis (SPID Section 6.4.3)

The screening approach is documented in Section 4.4.1 of this document. The selection of SSCs for seismic fragility analysis used a capacity-based screening approach. This approach meets the recommendations in Section 6.4.3 of the SPID (Reference 2). All screened SSCs are retained in the PRA model. Note that analysis assessment PRA-BV1-17-004-R00 (Reference 92) documents the cumulative impact of all screened SSCs at <5% and further shows that no screened SSCs are significant based on importance measures.

Topic 9: Use of the CDFM/Hybrid Methodology for Fragility Analysis (SPID Section 6.4.1)

The CDFM methodology used for fragility analysis is documented in Section 4.4.2.2 of this submittal and meets the recommendations in section 6.4.1 of the SPID (Reference 2). Recommended variabilities in Table 6-2 of the SPID were used to develop full seismic fragility curves.

Topic 10: Capacities of SSCs Sensitive to High-Frequencies (SPID Section 6.4.2)

Contact devices identified in EPRI Phase 2 testing (Reference 90) as being sensitive to high-frequency seismic motion were included in the relay chatter evaluation documented in Section 5.1.3 of this submittal. The flow chart on Figure 6-7 of the SPID (Reference 2) can be applied to the high-frequency analysis because all high-frequency susceptible components of interest were identified through circuit analysis and if not screened from the circuit analysis had a high-frequency capacity calculated. The High Frequency Fragility Calculations were performed in accordance with EPRI's High Frequency Program - Application Guidance for Functional Confirmation and Fragility Evaluation (Reference 91). During the high-frequency fragility calculation a capacity versus demand evaluation is performed, and in all cases the capacity was greater than the demand, and therefore no components required replacement.

Topic 11: Capacities of Relays Sensitive to High-Frequencies (SPID Section 6.4.2)

The standard is acceptable for the fragility analysis, but additional guidance is presented in the SPID for circuit analysis and operator actions analysis. The BVPS-1 SPRA does not credit any specific operator action in response to any seismic-induced relay chatter. Circuit analysis was performed to identify relays that can potentially impact plant SSCs if chatter were to occur, and screen out the relay devices that do not pose a safety concern. The circuit analysis was performed in accordance with the Standard and also meets the SPID (Reference 2) and is documented in Section 5.1.3 of this submittal.

Topic 12: Selection of Dominant Risk Contributors that Require Fragility Analysis Using the Separation of Variables Methodology (SPID Section 6.4.1)

The SPRA uses the CDFM methodology for the bulk of SSCs requiring seismic fragility analysis.

Separation of Variables was not required. This is supported by the sensitivities presented in Section 5.7.1.6 of this submittal combined with a sufficiently low seismic CDF (SCDF) or 1.3E-05 and seismic large early release frequency (SLERF) of 6.14E-07. The sensitivities argue that even if the high confidence of a low probability of failure (HCLPF) of the top contributors were improved the reduction in risk is not worth the additional analysis. Furthermore with the low SCDF/SLERF values any potential conservatisms/uncertainties in the CDFM methodology are deemed acceptable.

Topic 13: Evaluation of LERF (SPID Section 6.5.1)

The evaluation LERF is judged to meet each of the elements of section 6.5.1 of the SPID (Reference 2) including Table 6-3. Section 5.1.2 of this submittal details the evaluation of LERF in the SPRA. In addition Sensitivity Case 5 in Section 5.7.1 addresses the potential impact of a seismic event extending the evacuation time.

Topic 14: Peer Review of the Seismic PRA, Accounting for NEI 12-13 (SPID Section 6.7)

The peer review of the seismic PRA performed meets the elements in Section 6.7 of the SPID (Reference 2). An in-process peer review was not performed for the SPRA. Although it is not specifically stated in the peer review report (Reference 6), the lead fragility peer reviewer and one of the two supporting fragility peer reviewers has successfully completed the SQUG training course. Additionally the fragility peer review team lead wrote most of the training course and conducted most of the original classroom lectures.

Topic 15: Documentation of the Seismic PRA (SPID Section 6.8)

This submittal is judged to meet the documentation requirements of section 6.8 of the SPID. Additionally, all documentation supporting requirements were judged met by the peer review team with the exception of SHA-J3 which is judged to be met by the response to finding 2-30.

Topic 16: Review of Plant Modifications and Licensee Actions, If Any

There are no modifications necessary to achieve the appropriate risk profile.

F&O 2-1

PRA Peer Review Fact & Observation 2-1 was identified in the Probabilistic Seismic Hazards Analysis High Level Requirement, Supporting Requirement SHA-F1 (and other affected Supporting Requirement SHA-F3).

DETAILS (Peer Review Team)

As part of the Davis-Besse peer review it was determined that the propagation of the epistemic uncertainty in the ground motion models is not correctly carried out in the estimate of the total seismic hazard at the site. The PSHA report acknowledges this finding and indicates the BVPS PSHA will be updated when an appropriate methodology is implemented in the seismic hazard software.

This issue does not impact the estimate of the mean hazard.

BASIS FOR SIGNIFICANCE (Peer Review Team)

The methodology that is implemented in the RIZZO seismic hazard software to propagate the uncertainty in the ground motion models for individual seismic sources to determine the uncertainty in the total seismic hazard at a site does not correctly implement the ground motion logic tree.

POSSIBLE RESOLUTION (Peer Review Team)

The methodology that is used in the RIZZO seismic hazard software to combine the seismic hazard for individual seismic sources to estimate the total seismic hazard (the propagation of epistemic uncertainties in the ground motion model) should be changed to properly implement the ground motion logic tree. The PSHA calculations for the BVPS should be re-run, including the estimate of the rock site hazard results and the incorporation of the uncertainty in the local site response to estimate the FIRS.

The methodology that is used in the RIZZO seismic hazard software to combine the seismic hazard for individual seismic sources to estimate the total seismic hazard (the propagation of epistemic uncertainties in the ground motion model) should be changed to properly implement the ground motion logic tree. The PSHA calculations for the BVNS should be re-run, including the estimate of the rock site hazard results and the incorporation of the uncertainty in the local site response to estimate the FIRS.

PLANT RESPONSE OR RESOLUTION (ABS Consulting, RIZZO Associates, and FENOC)

The method for combining seismic hazard curves from individual sources is revised such that when combining hazard curves for one seismic source (consistent with CEUS-SSC logic tree structure) each ground motion prediction equation (GMPE) is considered separately (consistent with the EPRI GMM logic tree). Accordingly, the post-processing scripts that implement the combination method are revised 1) to retain intermediate seismic hazard results (for each source and for each GMPE), and 2) to combine the full set of seismic hazard curves to correctly derive the total mean and fractiles. Documentation of the revised combination method is provided in more detail below.

Enhancement of the method to propagate uncertainty in local site response is described in the Disposition to F&O 2-2.

The revised control-point hazard (reactor building [RB] foundation level) due to the above revision in the hazard combination method, and incorporating enhancements to better propagate uncertainty in site response to address F&O 2-2, exhibits insignificant changes to the mean hazard and the mean uniform hazard response spectra used to determine the FIRS, while the low and high fractiles show small differences. Therefore, as discussed further below, the fragility analyses of plant SSCs, which are based on the reported FIRS, are unaffected.

Note that RIZZO-HAZARD software that calculates the hazard for each branch of the PSHA logic tree is fully verified and validated and produces correct results. The issue identified in this F&O is related to post-processing scripts that combine outputs from RIZZO-HAZARD, and not with the basic hazard computation.

<u>Revision of Method for Hazard Combination</u>

RIZZO Calculation No. 12-4735-F-120, Revision 2, develops the seismic hazard for hard-rock conditions. It describes the post-processing scripts that incorporate the GMPE correlation model, and provides details of the methodology implemented to derive the hard rock total seismic hazard curves as follows:

- Section 5.2.3: Describes the GMPE correlation model used to combine hazard curves from RLME and distributed seismicity sources.
- Section 5.4: Describes the RIZZO-HAZARD hazard curve data files per source zones (RLME and distributed seismicity sources), GMPE, and magnitude-range weighting cases used in the recurrence relationship (Cases A, B and E).
- Section 5.9.5: Describes the combination of the hazard curves from Section 5.4 to obtain total rock hazard curves. The scripts described in this section perform the following steps:
  - Uploading the hazard curves per GMPE and the three magnitude-range weighting cases used in the recurrence relationship for the distributed seismicity sources, and only by GMPE in the case of RLME source zones (files described in Section 5.4).
  - Combining hazard curves from source zones (Section 5.4) considering correlations among the magnitude-range weighting in the recurrence relationships and among GMPEs when two distributed seismicity sources are combined, and the GMPE correlation model described in Section 5.2.3 when an RLME and distributed seismicity source are combined.

RIZZO Calculation 12-4735-F-120, Revision 2, and Calculation 12-4735-F-121, Revision 2, document the resulting mean hazard and the hazard fractile curves for hard rock at the BVPS site implementing the above revisions; and Section 4.3 of *ABS Consulting/RIZZO Report 2734294-R-003, Revision 4* (updated PSHA Report), summarizes the revised hard rock hazard.

Revision of Method for Propagation of Uncertainty in Local Site Response

The "…incorporation of the uncertainty in the local site response to estimate the FIRS." in the Peer Review Suggested Resolution for Finding F&O 2-1 is addressed in the response to Finding F&O 2-2.

Assessment of Effect of Revised GMRS and FIRS on Fragility Analyses

The FIRS reported in Section 6.4 of *ABS Consulting/RIZZO Report 2734294-R-003, Revision 4,* show minor differences as compared to the FIRS reported previously in *ABS Consulting/RIZZO Report 2734294-R-003, Revision 1.* However, the differences in the spectral shapes are insignificant. Based on a comparison of the spectral shapes of the FIRS the impacts on the fragilities reported in *ABS Consulting/RIZZO Report 2734294-R-006, Revision 0,* are also insignificant. Therefore, the ground motion time histories, the building analysis, and the fragility analysis remain unaffected. This is further discussed and justified in the Section 5.5 of the revised fragility analysis reports (*ABS Consulting/RIZZO Report 2734294-R-006, Revision 1, and ABS Consulting/RIZZO Report 2734294-R-013, Revision 1*).

F&O 2-2

PRA Peer Review Fact & Observation 2-2 was identified in the Probabilistic Seismic Hazards Analysis High Level Requirement, Supporting Requirement SHA-F1 (and other affected Supporting Requirement SHA-J1).

DETAILS (Peer Review Team)

To estimate the seismic hazard at the top of the soil column (e.g., at the reactor building base elevation) the aleatory and epistemic uncertainties in the rock PSHA results and the site amplification factors are not combined to estimate the total epistemic uncertainty in the soil hazard.

This issue does not impact the estimate of the mean hazard.

BASIS FOR SIGNIFICANCE (Peer Review Team)

To estimate the seismic hazard at the top of the soil column, the rock PSHA results are combined with the probabilistic characterization of the site amplification factors. The site amplification is represented by the mean and standard deviation for the total uncertainty (combined aleatory and epistemic uncertainty) and the assumption that the amplification factors are lognormally distributed. Thus, the epistemic uncertainty in the rock site hazard is probabilistically combined with the site amplification aleatory and epistemic uncertainty. As a result, the epistemic uncertainty in the site amplification is not combined with the rock hazard uncertainty to estimate the uncertainty in the soil hazard, leading to the uncertainty in the soil hazard being underestimated.

Since the aleatory and epistemic uncertainty in the site amplification are considered, the estimate of the mean soil hazard should not be effected.

The approach that is used under-estimates the epistemic uncertainty in the soil hazard and is therefore unconservative. As a result the uncertainty in the seismic risk (CDF and LERF) will be underestimated.

As currently implemented the process for generating the input to the SPRA quantification (a series of 100 hazard curves) also does not combine the rock site hazard and the site amplification uncertainties.

POSSIBLE RESOLUTION (Peer Review Team)

As part of the site response analysis, maintain the segregation of aleatory and epistemic uncertainties and propagate these properly when combined with the rock hazard results to estimate the seismic hazard and the top of the soil column.

PLANT RESPONSE OR RESOLUTION (ABS Consulting and RIZZO Associates)

RIZZO Calculation 12-4735-F-117 is revised (Revision 2) to appropriately segregate the aleatory and epistemic uncertainties in site response such that they can be properly propagated when combining the site response with the rock hazard results to obtain control point (i.e., "soil") hazard. *RIZZO Calculation 12-4735-F-118, Revision 2 (Reactor Building Foundation), Calculation 12-4735-F-123, Revision 1 (AUX, DGB, FDB, MSVCV, SFGB, SRV, and CB Foundation), and Calculation 12-4735-F-125, Revision 1 (Intake Structure Foundation),* illustrate that the revised treatment of the uncertainties in the site response analysis, along with other changes to address F&O's, result in insignificant changes in mean horizontal control-point hazard at the top of the soil column and corresponding UHRS used to develop FIRS, while the low and high fractiles show small differences. As discussed further below, the fragility analyses of plant SSCs, which are based on the reported FIRS, are unaffected.

Revised Treatment of Aleatory and Epistemic Uncertainty in Site Response Analysis

The logic tree of input parameters for site response analysis, shown on Figure 5-1 (*Section 5.2 of ABS Consulting/RIZZO Report 2734294-R-003, Revision 4*), has 20 branches accounting for various combinations of input parameters reflecting epistemic uncertainties in the site response analysis. The aleatory variability is represented by 30 combinations of randomized Vs profiles (from hard rock to the control-point elevation at the top of the soil column), and corresponding randomized G/Gmax and damping curves. The end branches of the logic tree reflect epistemic uncertainty in the various site response inputs and take into account guidance on characterizing uncertainty provided in *Screening, Prioritization, and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic* (EPRI, 2013b). The calculation results in the mean and standard deviation of the site amplification functions (SAF) for each branch of the logic tree for each of 11 hard-rock hazard levels.

In *RIZZO Calculation 12-4735-F-117, Revision 1*, which is summarized in *ABS Consulting/RIZZO Report 2734294-R-003, Revision 1*, the approach described in EPRI (2013b, Section B-6) was followed to develop probabilistic hazard curves. Site amplification functions were determined for each combination of response frequency and hard-rock ground motion amplitude weighted sums over the 20 site response models. This effectively transfers the epistemic uncertainty in site response into aleatory uncertainty.

In *RIZZO Calculation 12-4735-F-117, Revision 2*, which is summarized in *ABS Consulting/RIZZO Report 2734294-R-003, Revision 4, and related RIZZO Calculations 12-4735-F-122 R2 and 12-4735-F-124 R2*, the site response results are summarized to maintain the general characteristics of site amplification uncertainty related to epistemic uncertainty in site response analysis inputs. Epistemic uncertainty in site response analysis inputs that does not translate into significant epistemic uncertainty in SAFs is averaged; i.e., transferred to aleatory uncertainty. Epistemic uncertainty in site response analysis that leads to relatively significant uncertainty in SAFs is retained and carried into the control-point (soil) hazard calculation.

More specifically, the control-point (RB foundation level) hazard is obtained by the convolution of hard-rock hazard with the SAF, as described in Section 6.1 of the *ABS Consulting/RIZZO Report 2734294-R-003, Revision 4*. Although in principle this process is able to segregate and propagate the aleatory and epistemic uncertainty in the site response, the previous analysis (*ABS Consulting/RIZZO Report 2734294-R-003, Revision 1*) treats epistemic uncertainty as aleatory variability, consistent with the SPID guidance (EPRI Technical Report #1025287, 2013b). However, we concur with this F&O that the propagation of epistemic uncertainty in site response into the PSHA more accurately determines the control-point (soil) hazard fractiles. In response to the F&O, *RIZZO Calculation 12-4735-F-117, Revision 2,* describes the method used to properly segregate and propagate the aleatory and epistemic uncertainties in the convolution of the SAFs with the hard-rock hazard.

Because it is computationally prohibitive to incorporate the full set of epistemic simulations (20 branches x 36 spectral frequencies x 11 HR hazard levels) into the hazard analysis, a simplified approach is utilized. This approach examines the SAFs at each end branch of the site response logic tree for all levels of input motions, and bins them into three groups of epistemic branches based on which inputs dominate the epistemic uncertainty in site response and on the similarity of the SAF. Section 5.10 of the *ABS Consulting/RIZZO Report 2734294-R-003, Revision 4,* describes the grouping and develops representative SAFs for each group. The respective group SAFs are used to convolve with the hard-rock hazard and propagate the epistemic uncertainties in developing the control-point hazard.

Calculation 12-4735-F-117, Revision 2, describes the basis for the SAF grouping (three groups), and presents Tables and Figures displaying the SAF for each group and at each of the seven spectral frequencies. This calculation is also expanded to document additional details on the derivation of the inputs used in the site response analysis. Much of this material was previously included in Section 5.0 of *ABS Consulting/RIZZO Report 2734294-R-003, Revision 1.* Further, *Calculation 12-4735-F-118, Revision 2,* provides details about how the GMPE correlation model and the epistemic uncertainty in SAF are incorporated in the process, as follows:

- Section 5.1 describes how the three groups of SAF are applied to the hard-rock hazard curve for each branch of the logic tree to obtain a new population of hazard curves at the RB foundation elevation.
- Section 5.2 describes how the scripts from calculation F-120 (hard-rock hazard curves) are modified to apply one of the three SAF groups and perform the full combination of the hazard curves considering the CEUS-SSC and EPRI GMM model logic trees. The modification to the script saves the hazard curves at the RB foundation calculated with each of the three SAF groups. Section 5.2 also describes how the three sets of hazard curves at the RB foundation obtained from the three SAF groups are combined to obtain the total RB foundation hazard curves.
- *Calculation 12-4735-F-143, Revision 2,* describes how the control-point (soil) hazard distribution for the reactor building foundation, which is determined by appropriately segregating epistemic uncertainty and aleatory variability in site response analysis and then propagating them properly when combining them with rock hazard results, is used to provide the 100 hazard curves used as input to SPRA quantification.

Note that, other than the guidance in the SPID, no other guidance is available on how site response epistemic uncertainty should be assessed as part of deriving seismic hazard curves, particularly hazard curve fractiles, while maintaining reasonable computational efforts. Given that site response epistemic uncertainty essentially impacts each GMPE used in the hazard computation, the grouping approach focuses on the critical site response epistemic uncertainty while maintaining computational viability in developing accurate mean hazard curves at the elevations where FIRS are needed for fragility calculation, and hazard fractiles at the RB foundation elevation to which the fragilities are referenced.

Assessment of Effect of Revised GMRS and FIRS on Fragility Analyses

The FIRS reported in Section 6.4 of *ABS Consulting/RIZZO Report 2734294-R-003, Revision 4,* show minor differences as compared to the FIRS reported previously in *ABS Consulting/RIZZO 2734294-R-003, Revision 1.* However, the differences in the spectral shapes are insignificant. Based on a comparison of the spectral shapes of the FIRS the impacts on the fragilities reported in *ABS Consulting/RIZZO Report 2734294-R-006, Revision 0*, are also insignificant. Therefore, the ground motion time histories, the building analysis, and the fragility analysis remain unaffected. This is further discussed and justified in the Section 5.5 of the revised fragility analysis reports (*ABS Consulting/RIZZO Report 2734294-R-006, Revision 1,* for BVPS Unit 1 and *ABS Consulting/RIZZO Report 2734294-R-013, Revision 1, for BVPS Unit 2).*

F&O 2-3

PRA Peer Review Fact & Observation 2-3 was identified in the Probabilistic Seismic Hazards Analysis High Level Requirement, Supporting Requirement SHA-F2.

DETAILS (Peer Review Team)

Sensitivity studies and intermediate results have not been systematically carried out and reported in the PSHA documentation. While some deaggregation results are reported (which can be interpreted as intermediate and sensitivity calculations), a systematic demonstration of sensitivity of the results to key parameters is not presented.

BASIS FOR SIGNIFICANCE (Peer Review Team)

The PSHA report does not present a comprehensive assessment of the sensitivity of the seismic hazard results to the different elements of the analysis; e.g., seismic source model uncertainty, ground motion model uncertainty, etc.

POSSIBLE RESOLUTION (Peer Review Team)

Perform and present sensitivity calculations that demonstrate the sensitivity of the hazard results to elements of the PSHA; ground motion attenuation models; estimates of site amplification; alternative soil profiles, estimates of kappa, etc. The sensitivity of the hazard to different factors in the PSHA could be demonstrated by adding "tornado plots" at different ground motion levels to the various branches in the logic tree. These plots show which sources of epistemic uncertainty are most important. It should include the source model uncertainty, ground motion model uncertainty, and site response uncertainty. Currently, the total uncertainty is shown by the hazard fractiles, but it is not broken down to provide understanding as to what is most important.

PLANT RESPONSE OR RESOLUTION (ABS Consulting and RIZZO Associates)

The response to this F&O improves the documentation and presentation of some of the intermediate hazard results and provides additional sensitivity calculations to provide insight into what inputs more strongly contribute to the overall distribution of hazard results. It does not change the hazard or the seismic demand on which the fragilities are based.

RIZZO Calculation F-144 Revision 1 develops the total variance deaggregation for 100 Hz surface hazard for all the logic tree branches and for different ground motion levels represented by mean annual frequency of exceedances (MAFE). The total hazard variance is deaggregated in terms of the following PSHA elements:

- Seismic source model uncertainty
  - Alternative source model approach
  - Mmax
  - Recurrence rates
  - Magnitude weighting case used to determine recurrence rates
  - Thickness of the seismogenic layer;
- Ground motion model uncertainty
- Site response uncertainty
  - Alternative SAF groupings

The deaggregated variance is a measure of relative contribution of epistemic uncertainty in each element to the total variance. These relative contributions are response frequency and annual frequency of exceedance (AFE) dependent.

Additionally, ***RIZZO Calculation F-117, Revision 2,*** develops median and standard deviations of SAFs for the 20 epistemic branches of the site response inputs logic tree. The logic tree represents the assessed uncertainty in geologic profile, seismic source spectra model, profile damping, and site kappa. These intermediate results are documented in ***Section 5.8.8*** of ***ABS Consulting/RIZZO Report 2734294-R-003, Revision 4*** (the updated PSHA Report).

RIZZO Calculation F-144, Revision 1, shows that the dominant contributor to the total variance is the epistemic uncertainty in the ground motion model; i.e., GMPEs. As the MAFE gets lower, the epistemic uncertainty in maximum magnitude, the three magnitude-range cases used for deriving recurrence rate, and the eight recurrence rate realizations become more significant. Similarly, ***RIZZO Calculation F-117, Revision 2,*** shows that the most significant factor impacting the SAFs is the uncertainty in geologic profile definition.

The above sensitivity studies were performed for additional insight of the epistemic uncertainty only, and do not affect or change any inputs to the PRA model.

Section 5.9 of ABS Consulting/RIZZO ***Report 2734294-R-003, Revision 4,*** documents the contribution of different sources of uncertainties modeled in the PSHA. It describes the wide range of sensitivity calculations and also presents an assessment of the variance contribution to the hazard for all PSHA inputs (seismic source, ground motion, and site response). The variance assessment is accomplished for a wide range of ground motion levels represented by the annual frequencies of exceedance. Figure 5-37 displays the variance contribution for each PSHA input. This is effectively similar to "tornado plots," and provides an understanding of which PSHA inputs are more significant from an epistemic uncertainty perspective.

F&O 2-26

PRA Peer Review Fact & Observation 2-26 was identified in the Probabilistic Seismic Hazards Analysis High Level Requirement, Supporting Requirement SHA-I1 (and other affected Supporting Requirement SHA-I2).

DETAILS (Peer Review Team)

A screening assessment of other seismic hazards was performed. There are a number of technical questions associated with elements of the analysis for some of the other seismic hazards.

BASIS FOR SIGNIFICANCE (Peer Review Team)

The NRC has identified two dams that are upstream of the BVPS that may pose a flood hazard. In fact there are multiple dams upstream of the plant.

The PSHA report does not address the potential for seismically-initiated dam failure that could impact the dams. A large seismic event in the region could potentially simultaneously cause high ground motions at the BVPS and at the upstream dams leading to dam failure and damage to the BVPS.

POSSIBLE RESOLUTION (Peer Review Team)

The potential seismically-initiated failure of upstream dams and their flooding consequence should be addressed as part of the seismic screening analysis.

PLANT RESPONSE OR RESOLUTION (ABS Consulting and RIZZO Associates)

Section 7.3.5 ("Seismically Induced Dam Failures") of has been added to *ABS Consulting/RIZZO Report 2734294-R-003, Revision 4,* to include an assessment for the potential seismically induced failure of upstream dams and their flooding consequences. The analysis reported in BVPS-2 UFSAR (Appendix 2.4A) concludes that the failure of the upstream Conemaugh Dam, which is the most critical with respect to flooding, raises the flood stage to EL 725.2 ft. This is less than design basis flood level of EL 730.0 ft. Therefore, this seismic-related hazard is screened out from further analysis.

F&O 2-27

PRA Peer Review Fact & Observation 2-27 was identified in the Probabilistic Seismic Hazards Analysis High Level Requirement, Supporting Requirement SHA-I1 (and other affected Supporting Requirement SHA-I2).

DETAILS (Peer Review Team)

A screening assessment of other seismic hazards was performed. There are a number of technical questions associated with elements of the analysis for some of the other seismic hazards.

BASIS FOR SIGNIFICANCE (Peer Review Team)

It is argued that the consequence of slope failure that is based on the minimum FOS slip surfaces is negligible because they do not intersect critical structures. However, analyses are not presented of slip surfaces that would have safety consequences to plant structures in order to show margins against these slope failures.

Impact of failure of slope in Cross Section 2-2 on the Intake Structure itself has not been clearly assessed. It is stated on Page 410 3rd paragraph, "In the event of a failure in Section 2-2, the material of the lower slope is expected to displace less than one half of a foot. The upper slope in Section 2-2 is expected to be retained by the retaining structure. These displacements are relatively small and do not affect the function of the Intake Structure." It is not clear that this has been clearly analyzed in the context that a HCLPF for displacements has been analyzed.

A generic procedure has been used to estimate the HCLPF for soil structures. It is not clear that the generic procedure that includes (at least implicitly) estimates of aleatory and epistemic uncertainty in soil properties, stability analyses, etc. is an appropriate basis to estimate the HCLPF and serve as a basis for screening.

POSSIBLE RESOLUTION (Peer Review Team)

The analysis should evaluate potential slope failure modes that would impact critical structures and components.

PLANT RESPONSE OR RESOLUTION (ABS Consulting and RIZZO Associates)

Section 7.3.3 of *ABS Consulting/RIZZO Report 2734294-R-003, Revision 4* evaluates three permanent slopes whose failure could affect safety-related functions, including:

- Slope north of the Unit 1 (Figure 2.6-3 of BVPS-1 UFSAR).
- Riverward slopes involving Service Water Piping (Figure 2.5.4-57 of BVPS-2 UFSAR).
- Intake Channel Slopes (Figures 2.5.4-37 and 61 of BVPS-2 UFSAR).

As reported in Section 7.3.3, the slope stability analyses for the above permanent slopes are performed using Version 7.23 of the SLOPE/W Stability Analysis Program (Geo-Slope, 2007; RIZZO, 2012b). The HCLPFs are obtained using site-specific geotechnical characteristics obtained from the FSAR. As described below, the HCLPFs for slope failures are smaller than 0.5g. However, the slope failure is screened on the basis of the consequence to the affected

SSCs. The consequences are assessed based on the expected post-failure displacements, which are significantly smaller than the distance to the affected structures.

The slope north of Unit 1 has a HCLPF value of 0.5g PGA. It is noted, however, that this failure mode does not affect the Turbine Building (TRBB) because the failure circle is expected to daylight about 150 ft from the TRBB foundation. The HCLPF value of the analyzed failure circle is taken to be a conservative lower bound affecting the TB. This is in excess of the assumed HCLPF of the TB structure. Potential failure surfaces involving the TB footprint would be characterized by larger margins and are not controlling failure modes associated with slope failure affecting the TB.

The minimum slope stability factor of safety for the Riverward Slope is 1.54. The corresponding HCLPF value is 0.33g PGA. In the event of a slope stability failure, a maximum displacement of 1 inch is predicted. Based on the acceleration required to cause 1 inch of displacement, the HCLPF capacity associated with slope displacement is 0.38g. This analysis also shows that the critical slip surface outcrops approximately 150 ft from the Intake Structure. Therefore, possible displacements due to the slope failure caused by an earthquake are not expected to affect the structural integrity of the Intake Structure. Shallower failure surfaces extending to the Intake Structure are expected to have larger factors of safety than the critical slip surface, and therefore do not represent controlling failure modes for slope failure.

The factors of safety for the upper and lower slopes at the intake are calculated to be 1.66 and 1.43., and the corresponding HCLPF values for slope failure are 0.36g and 0.31g. In the event of slope failure, the upper slope is expected to be retained by the retaining structure. The unrestrained displacements of the lower slope are less than one foot. Therefore, it will not affect the function of the Intake Structure, which is more than 90 ft from the toe of the slope.

The analyses presented conclude that potential failure of the intake slopes and the resulting displacement profiles do not affect the structural integrity of the structures or the function of the intake channel.

F&O 2-28

PRA Peer Review Fact & Observation 2-28 was identified in the Probabilistic Seismic Hazards Analysis High Level Requirement, Supporting Requirement SHA-I1 (and other affected Supporting Requirement SHA-I2).

DETAILS (Peer Review Team)

A screening assessment of other seismic hazards was performed. There are a number of technical questions associated with elements of the analysis for some of the other seismic hazards.

BASIS FOR SIGNIFICANCE (Peer Review Team)

Text in the PSHA report at the bottom of Page 405 indicates a minimum HCLPF for Unit 2 bearing capacity of 0.45g. The minimum value in Table 7-1 for Unit 2 appears to be 0.50g.

POSSIBLE RESOLUTION (Peer Review Team)

Modify the text to be consistent with the analysis results.

PLANT RESPONSE OR RESOLUTION (ABS Consulting and RIZZO Associates)

Section 7.3.2 of *ABS Consulting/RIZZO Report 2734294-R-003, Revision 4,* has been revised to be consistent with the minimum HCLPF presented in Table 7-1. This is a documentation change and does not affect PRA inputs.

F&O 2-29

PRA Peer Review Fact & Observation 2-29 was identified in the Probabilistic Seismic Hazards Analysis High Level Requirement, Supporting Requirement SHA-I1 (and other affected Supporting Requirement SHA-I2).

DETAILS (Peer Review Team)

A screening assessment of other seismic hazards was performed. There are a number of technical questions associated with elements of the analysis for some of the other seismic hazards.

BASIS FOR SIGNIFICANCE (Peer Review Team)

There is no indication that lateral spreading of the ground in the vicinity of the Intake Structure or other critical structures has been assessed.

A generic procedure has been used to estimate the HCLPF for soil structures. It is not clear that the generic procedure includes (at least implicitly) estimates of aleatory and epistemic uncertainty in soil properties, stability analyses, etc. is an appropriate basis to estimate the HCLPF and serve as a basis for screening.

POSSIBLE RESOLUTION (Peer Review Team)

The analysis should evaluate the potential for liquefaction and lateral spreading that could impact critical structures and components.

PLANT RESPONSE OR RESOLUTION (ABS Consulting and RIZZO Associates)

As described in Section 7.3.4 of *ABS Consulting/RIZZO Report 2734294-R-003, Revision 4*, the foundations for all power block structures are supported on either in-situ competent soil in the higher terrace or on engineered structural backfill. NUREG/CR-5741 concludes that the liquefaction susceptibility of terrace soils from the Pleistocene period is 'very low'. Additionally, the liquefaction potential is also 'very low' when depth of the groundwater is greater than about 50 ft (NUREG/CR-5741; NRC, 2000). All of the power block structures satisfy both conditions, and are therefore not affected by liquefaction, and this failure mode is screened out for the power block SSCs.

Section 7.3.4 presents the detailed liquefaction analysis of the yard area between the plant and the intake. The reported liquefaction analysis is based on conservative design parameters in the FSAR such as recorded SPT blow counts, the particle size distribution and fines content, and the water table elevation. These are taken to be the 84th percentile values. The calculated HCLPFs for liquefaction and its effects on affected SSCs (buried pipes) thus represent CDFMs.

Based on the calculated settlements due to liquefaction, and assuming an allowable seismic-induced settlement associated with the buried lines of 3 inches, the HCLPF value associated with seismic-induced settlement is 0.39g. Allowing for a nominal ductility ($F\mu = 2.0$), the HCLPF associated with structural integrity of the buried pipes is about 0.8g. This is significantly in excess of the CDFM HCLPF values of equipment in the Intake Structure. Therefore, the liquefaction failure mode affecting the plant SSCs is screened out. Additionally, due to the generally flat topography lateral spreading is not an issue.

F&O 2-30

PRA Peer Review Fact & Observation 2-30 was identified in the Probabilistic Seismic Hazards Analysis High Level Requirement, Supporting Requirement SHA-J3.

DETAILS (Peer Review Team)

A foundational element of PSHA as it has evolved over the past 30 years is the development and implementation of methods to identify, evaluate, and model sources of epistemic (model and parametric) uncertainty in the estimate of ground motion hazards. As such fairly rigorous analyses are carried out (SSHAC studies) to quantitatively address model uncertainties.

At the same time there is within any analysis sources of uncertainty that are not directly modeled and assumptions that are made for pragmatic or other reasons. There are also sources of model uncertainty that are embedded in the context of current practice that are 'accepted' and typically not subject to critical review. For instance, in the PSHA it is standard practice to assume that the temporal occurrence of earthquakes is defined by a Poisson process. This assumption is well accepted despite the fact that it violates certain fundamentally understanding of tectonic processes (strain accumulation). A second practice is the fact that earthquake aftershocks are not modeled in the PSHA, even though they may be significant events (depending on the size of the main event).

In the spirit of the standard it seems appropriate that sources of model uncertainty that are modeled as well as sources of uncertainty and associated assumptions as they relate to the site-specific analysis should be identified/discussed and their influence on the results discussed.

As SPRA reviews and the use of the standard have evolved, it would seem the former interpretation is reasonable, but potentially incomplete. It is reasonable from the perspective that documentation of the sources of model uncertainty and their contribution to the site-specific hazard results is a valuable product that supports the peer review process and assessments in the future as new information becomes available). Similarly, documenting assumptions provides similar support for peer reviews and future updates. The notion that model uncertainties and related assumptions that are not addressed in the PSHA is at a certain level an extreme requirement that may not be readily met and may not be particularly supportive of the analysis that is performed.

For purposes of this review, the following approach is taken with regard to this supporting requirement:

1. The documentation should present quantitative results and discussion the sources of epistemic uncertainty that are modeled and their contribution to the total uncertainty in the seismic hazard.
2. The documentation should discuss elements of the PSHA model where these may be latent sources of model uncertainty that are not modeled and assumptions that are made in performing the analysis.

BASIS FOR SIGNIFICANCE (Peer Review Team)

The documentation of the sources of model uncertainty analysis and a description of the analysis assumptions is not complete in the PSHA report in its current form such that a clear understanding of the contribution of individual sources of uncertainty to the estimate of hazard are understood. Limited information on the contribution of seismic sources to the total mean hazard is presented, but information on the contributors to the uncertainty is not provided.

With respect to addressing model uncertainties and associated assumptions there are some examples that can be identified in the Beaver Valley (BV) PSHA. These are:

1. In the site response analysis the assumption is made that the 1D equivalent-linear model (SHAKE type) to estimate the site amplification and ground motion input to plant structures is appropriate. In addition, an assumption is made that the variation in the rock topography does not significantly influence the ground motion that is input to the plant. This modeling approach and the potential model uncertainty that it represents relative to the conditions at the BV site should be addressed.
2. In the estimate of vertical ground motions, an envelope of alternative V/H ratio models was used. This approach is conservative. It is implicitly assumed this approach is reasonable and appropriate as a basis to provide input to the seismic fragility analysis. This assumption and its potential implications is a topic that should be identified and discussed in the context of addressing this requirement.

POSSIBLE RESOLUTION (Peer Review Team)

The resolution to this finding could involve:

1. Documentation and discussion of the contribution of different sources of uncertainty that are modeled in the PSHA. The documentation of the contribution of different sources of uncertainty can be shown by means of "tornado plots" that quantify the sensitivity of the hazard at different ground motion levels to the various branches in the logic tree. These plots show which sources of epistemic uncertainty are most important. It should include the source model uncertainty, ground motion model uncertainty, and site response uncertainty. Currently, the total uncertainty is shown by the hazard fractiles, but it is not broken down to provide understanding as to what is most important.
2. Identification and discussion of model assumptions that are made.

PLANT RESPONSE OR RESOLUTION (ABS Consulting and RIZZO Associates)

This F&O relates to the documentation of the sensitivity analyses addressed in response to F&O 2-3 and documentation of model assumptions. It does not affect the hazard definition or the UHRS.

As stated in the Disposition of F&O 2-3, Section 5.9 of *ABS Consulting/RIZZO Report 2734294-R-003, Revision 4* documents the contribution to hazard of different sources of uncertainties modeled in the PSHA. Additionally, Section 5.8.8 presents details of the sensitivity of the site amplification factors to various inputs to the site response analysis such as geologic profile, ground motion amplitude, seismic source spectra, profile damping assumptions, and site kappa.

Section 5.9 concludes that the dominant contributor to the total hazard variance is the epistemic uncertainty in GMPEs. As the MAFE gets lower, the epistemic uncertainty in maximum magnitude, the three magnitude-range cases used for deriving recurrence rates and the eight recurrence rate realizations become more significant. Similarly, Section 5.8.8 concludes that the most significant factor impacting the SAFs is the uncertainty in geologic profile definition.

The modeling assumptions for various elements of the PSHA are described in Section 2.0 for the seismic source models and in Section 3.0 for the ground motion models and in the references cited therein. The modeling assumptions for the site response analysis are described in Section 5.0 and cited references.

Assumptions used are those associated with current standards of practice. Examples are as follows:

- Ergodic assumption as applied to the estimation of maximum earthquake magnitude for distributed seismicity sources and to ground motion prediction
- Seismic source characterization model
  - The spatial distribution of seismicity is generally temporally stationary
  - The occurrence of independent earthquakes is a stationary Poisson process
  - The size distribution of earthquake magnitudes for distributed seismicity sources follows an doubly truncated exponential distribution
- Ground motion characterization
  - Variability in ground motion follows a lognormal distribution
- Site response analysis
  - Use of equivalent-linear analysis and vertically propagating shear waves adequately represents the important trends in site response for the levels of ground motion considered
  - A site geotechnical model consisting of homogeneous, horizontal layers adequately represents the site conditions

Conditions at the Beaver Valley sites are consistent with the standard practice use of the above assumptions.

F&O 2-31

PRA Peer Review Fact & Observation 2-31 was identified in the Probabilistic Seismic Hazards Analysis High Level Requirement, Supporting Requirement SHA-I1 (and other affected Supporting Requirements SHA-I2, SFR-D1).

DETAILS (Peer Review Team)

A screening assessment of other seismic hazards was performed. There are a several technical questions associated with elements of the analysis for some of the other seismic hazards.

BASIS FOR SIGNIFICANCE (Peer Review Team)

An analysis was performed to assess potential bearing capacity failures.
Calculation 12-4736-F-033 R1 presents the methodology for calculating the bearing capacity; however it does not discuss how the HCLPF is estimated. As such it is not clear if the HCLPF estimates, which are the basis for screening bearing capacity failures are appropriate.

Discussions with the analyst involved in the analysis suggest that the median capacity for a bearing failure may not be significantly higher than the estimated median capacity. If this is the case, additional support for screening out this failure mode is required.

POSSIBLE RESOLUTION (Peer Review Team)

Provide documentation of the methodology for estimating the bearing capacity HCLPF. If the median seismic capacity is not significantly higher than the estimated HCLPF, then additional basis for screening out this failure mode should be provided

PLANT RESPONSE OR RESOLUTION (ABS Consulting and RIZZO Associates)

ABS Consulting/RIZZO Report 2734294-R-003, Revision 4, includes revisions to enhance the discussion of bearing capacity HCLPF values. It provides additional basis to screen out bearing capacity failure. Based on available margins assuming linear behavior, the HCLPF is sufficiently large to accommodate the possibility that, due to inherent nonlinearities, the median capacity is not significantly larger than the HCLPF.

Section 7.3.2 of *ABS Consulting/RIZZO Report 2734294-R-003, Revision 4,* documents the methodology for estimating the HCLPF associated with bearing capacity failure. The factors of safety reported in the FSAR indicate relatively significant margins against bearing capacity failure under SSE conditions. To account for potential uplift at higher ground motion levels, a bounding analysis is performed. This analysis conservatively ignores that uplift reduces the demand overturning moment. On the other hand, it accounts for the fact that uplift reduces the effective bearing area and therefore increases the bearing pressure and reduces the effective bearing capacity. Table 7-1 presents the resulting conservative bounds for the HCLPF values. The minimum bounding HCLPF for the BVPS Unit 1 and Unit 2 structures is 0.53g and 0.5g, respectively.

It is noted that uplift of the foundation mat due to seismic ground motion significantly reduces overturning moments and in turn the bearing pressure. These reductions in demand, along with (1) the calculated bounding HCLPFs in Table 7-1, and (2) the significant margins under SSE conditions, are used as basis to screen out bearing capacity as the controlling failure mode for the BVPS structures.

F&O 4-6

PRA Peer Review Fact & Observation 4-6 was identified in the Seismic Fragility Analysis High Level Requirement, Supporting Requirement SFR-A2 (and other affected Supporting Requirement SFR-F2).

DETAILS (Peer Review Team)

Excess conservatism and unrealistic assumptions were noted in a number of calculations providing the fragility parameters for components identified as top contributors to CDF.

BASIS FOR SIGNIFICANCE (Peer Review Team)

(Sequential letters added by FENOC for clarity in Plant Response or Resolution section)

a) BV1 residual heat removal (RHR) pumps are evaluated in 2734294-C-106 R0 BVPS1 Seismic Fragility for Vertical Pumps, Section 5.4. EW and NS seismic accelerations are enveloped. 3% damping is used but response is dominated by the steel support frame. CDFM capacity is scaled from a conservative design calculation. The design calculation includes operational considerations and seismic nozzle loads, but it is not checked if these loads are realistic for fragility evaluation purposes. No inelastic energy absorption factor is used. Weld capacity is governed by base metal and this is not a realistic failure modes per American Institute of Steel Construction (AISC).

b) BV1 Pressurizer power operated relief valve 2RCS-PCV455C is evaluated in 2734294-C-208 R0 BVPS2 Seismic Fragility for Motor _ Solenoid Operated Valves, Section 5.2. A conservative lower bound natural frequency estimate is used in the evaluation, and conservative generic capacity is assigned. A value lower than the calculated HCLPF capacity was used in the quantification.

c) BV1 Pressurizer relief valve 2RCS-RV551A is evaluated in calculation 2734294-C-207 R0 BVPS2 Seismic Fragility for Pneumatic Operated Valves, Section 5.27. A conservative lower bound natural frequency estimate is used in the evaluation, and conservative generic capacity is assigned. A value lower than the calculated HCLPF capacity was used in the quantification.

d) BV2 battery charger 2BAT-CHG2-7 is evaluated in calculation 2734294-C-216 R0 BVPS2 Seismic Fragility for Battery Chargers and Inverters, Section 5.2. Weight is determined by Reference to generic implementation procedure (GIP) and 3 x weight of sheet metal is used. However, this is for a control cabinet, not a battery charger. A battery charger weight should be based on 45 lbs/ft3. The resulting weight by generic method is 1,485 lbs, not 1,104 lbs as used in the calculation. A conservative 0.60 knock down factor is used in the fragility calculation for anchorage capacity due to unknown anchor type, but the anchor type was clearly identified during the peer review walkdown.

e) BV2 MCC-2-E06 is evaluated in 2734294-C-201 R0 BVPS2 Seismic Fragility for Motor Control Centers, Section 5.5. Functional capacity of the MCC is based on ratio of generic equipment ruggedness spectra (GERS) to ISRS for 18 Hz response in the vertical direction. The realistic failure mode of the MCC associated with vertical motion is not described. The anchorage section of the calculation states that vertical frequency is at least 33 Hz but 18 HZ is used for functional evaluation. A plug weld detail is assumed for the base connection. Plug weld capacity is governed by base metal capacity, although AISC no longer recognizes base metal as a realistic failure mode for filet welds.

f) The BV1 Primary Plant demineralized water storage tank (DWST) is evaluated in calculation 124736 F-135. Although it is essentially axisymmetric, loads are increased by 40 percent based on 100-40-40 considerations which are not applicable, thus introducing excess conservatism. The failure mode of tank wall bending is not applicable since the anchor chairs are encased in concrete.

g) BV1 RHR heat exchangers are evaluated in calculation 2734294-C-121 R0 BVPS1 Seismic Fragility for Tanks and Heat Exchangers, Section 5.9. The 19.8 Hx frequency estimate is conservatively applied in all directions. The same input motion scape factor is used in all directions. CDFM capacity is scaled from a conservative design calculation. The design calculation includes operational considerations and seismic nozzle loads, but it is not checked if these loads are realistic for fragility evaluation purposes. No inelastic energy absorption factor is used.

h) 2FWS-FCV479 is evaluated in calculation 2734294-C-207 R0 BVPS2 Seismic Fragility for Pneumatic Operated Valves, Section 5.13. Lack of meeting SQUG caveats is not described clearly in the calculation. A lower bound frequency estimate is used in the evaluation. A value lower than the calculated HCLPF capacity was used in the quantification.

i) The functional/anchorage HCLPF capacity for the representative battery charger, BAT-CHG1-5, is conservatively assumed to be 0.1g. Since this is one of the risk significant items ranked within top ten contributors to the seismic CDF, its fragility needs to be refined to obtain a more realistic estimate of the seismic fragility.

j) For a group fans on isolators listed in Table 5.3-1 of 2734294-C-109 R0, the obtained HCLPF capacity is calculated as 2.29 g on Page 31 of 2734294-C-109 R0. When a review of top contributors to seismic CDF, it is noticed that the fragility capacity for Emergency Switchgear heating, ventilation, and air-conditioning (HVAC) fans is set to the HCLPF capacity of 0.3g. Please explain the difference.

POSSIBLE RESOLUTION (Peer Review Team)

The Standard requires that realistic fragilities are used for top contributors to CDF. More realistic fragility analysis is required for these items.

PLANT RESPONSE OR RESOLUTION (ABS Consulting and RIZZO Associates)

a) In response to the F&O, a refined fragility was calculated for the BV1 RHR pumps in Revision 1 of BVPS-1 calculation 2734294-C-106. To this end, existing computer analytical models of the pumps and support frame documented in design calculations were reproduced in a new calculation 12-4735-F-141. This facilitated the elimination of conservatisms from the design calculation scaling approach in 2734294-C-106, Revision 0. Conservative assumptions removed by: (1) performing analysis in the NS and EW directions with their respective seismic accelerations, rather than an envelope,

(2) using specific motor weights for "A" and "B" pumps instead of an envelope, (3) retaining plus or minus signs for nozzle loads instead of conservatively assuming absolute maxima, (4) including dead weight of the support frame, (5) transferring calculated pump foot reactions from "A" and "B" pump models to the support frame model instead of envelope, (6) determining seismic responses of pump/frame based on 7% damping for welded steel structures and of piping nozzle loads based on 5% damping per ASCE 43-05 instead of conservative design damping, (7) using pinned connections at the support frame to reinforced concrete pier anchorage locations instead of conservatively assuming fixed connections, (8) applying the 100-40-40 rule for combining seismic spatial components in the three orthogonal directions instead of an absolute sum, (9) using the governing thermal condition instead of an envelope of potential thermal conditions for RHR pump suction and discharge nozzle loads, and (10) scaling seismic nozzle loads based on resonant frequencies of piping reported in design evaluations. In Revision 0 of calculation 2734294-C-106, the governing failure mode was of the ductile steel anchorage and an inelastic energy absorption factor of greater than unity could have been warranted. However, Revision 1 of calculation of calculation 2734294-C-106 expanded the structural fragility section for the RHR pumps to evaluation concrete-related failure modes of the anchorage calculated in accordance with ACI 349-06. The governing structural/anchorage failure mode of the pumps is concrete breakout failure of the pump support frame to reinforced concrete piers cast-in-place anchor bolts. An inelastic energy absorption factor was not used because this failure mode is brittle; i.e., $F\mu$=1. With respect to weld capacity, the capacity used in Revision 0 of Calculation 2734294-C-106 is in accordance with ANSI/AISC 360-10 Section J2.4 which states: "the design strength of welds shall be the lower value of the base material and the weld metal strength." All of these details are addressed in the Revision 1 of Calculation 2734294-C-106 and/or new Calculation 12-4735-F-141, Revision 0.

b) It should be noted that the peer review F&O report has a typographical error in the Basis for Significance section of this F&O. The first word in the second paragraph is "BV1," but the rest of the paragraph is about the Unit 2 pressurizer power operated relief valve (PORV), 2RCS-PCV455C. The fragility for this valve was updated after the BV2 model was locked. The fragility report summary table in Revision 1 of the fragility report 2734294-R-013 reflects the updated valve HCLPF of 0.54g, which has been incorporated into the PRA model (original HCLPF was 0.32g). In addition, the Seismic PRA Quantification Notebook now includes a group of sensitivities in Section 6 which address the models sensitivity to refinement of fragilities. These new cases only look at seismic components whose Fussell-Vesely importance (FVI) is greater than 0.03; anything less is considered to not significantly change results even if the HCLPF values were improved. Those SSCs that had a FVI >0.03 had sensitivities performed in which the HCLPF was doubled, in order to bound the small changes in HCLPF values that would more realistically be expected. This was done for both CDF and LERF. In many cases, the sensitivity showed a small change in CDF/LERF, indicating that improving the fragility would have very little effect on the model. In the cases for which there is a noticeable change, the fragilities of those SSCs are deemed to be already realistic—either because they were refined following peer review, or the peer review team did not identify any lack of realism in the fragility calculations—and calculating a more robust fragility is not seen as plausible. Therefore, the PRA team concludes that there are no possible conservatisms in the fragility calculations that are driving the model results or masking insights. The fragility for the PORV identified in this F&O was refined and also has a low Fussell-Vesely (FV), signifying that any additional refinement would not have a significant impact on the CDF/LERF.

c) The fragility for this valve (BV2 pressurizer relief valve 2RCS-RV551A) was updated after the BV2 model was locked prior to peer review. The fragility report summary table in Revision 1 of the fragility Report 2734294-R-013 reflects the updated valve HCLPF of 0.55g which has been incorporated into the PRA model (original HCLPF was 0.32g). In addition, the Seismic PRA Quantification Notebook now includes a group of sensitivities in Section 6 which address the models sensitivity to refinement of fragilities. These new cases only look at seismic components whose FVI is greater than 0.03; anything less is considered to not significantly change results even if the HCLPF values were improved. Those SSCs that had a FVI >0.03 had sensitivities performed in which the HCLPF was doubled, in order to bound the small changes in HCLPF values that would more realistically be expected. This was done for both CDF and LERF. In many cases, the sensitivity showed a small change in CDF/LERF, indicating that improving the fragility would have very little effect on the model. In the cases for which there is a noticeable change, the fragilities of those SSCs are deemed to be already realistic—or because they were refined following the peer review—and calculating a more robust fragility is not seen as plausible. Therefore, the PRA team concludes that there are no possible conservatisms in the fragility calculations that are driving the model results or masking insights. Since the peer review, the fragility for the pressurizer relief valve identified in this F&O was refined, and also has a low FV, signifying that any additional refinement would not have a significant impact on the CDF/LERF.

d) To resolve this F&O, the fragility for BV2 battery charger 2BAT-CHG2-7 was evaluated with estimated weight of 1,485 lbs calculated based on 45 pounds per cubic foot for battery chargers per the SQUG GIP and anchorage capacity based on plant-specific walkdown observations by qualified personnel from ABS, RIZZO, and/or FENOC that the anchors are shell-type Philips studs. Revision 1 of BVPS-2 Calculation 2734294-C-216 documents the updated evaluation of 2BAT-CHG2-7.

e) Per EPRI TR-102180, minimum frequencies of free standing MCCs are in the range of 3-10 Hz in the horizontal direction. The minimum horizontal frequency of 7 Hz was appropriately used in this calculation as the lower bound estimate. While the vertical frequency of MCCs were considered to be at 33Hz and above for evaluation of anchorage, the minimum frequency considered in functional fragility analysis was limited to 15 Hz to account for potentially damaging local modes of the MCC and internal components (e.g., breakers, contactors, transformers) with lower resonant frequencies. For anchorage fragility calculation, these local modes will not result in significant anchor loads and the evaluation is based on only the global resonant frequency which was judged above 33 Hz. Details of the connection between the MCC and base channel were not available during preparation of Revision 0 of this fragility analysis and a worst case scenario of plug weld anchorage was assumed for MCCs. Further walkdowns performed by qualified personnel from ABS, RIZZO, and/or FENOC confirmed that MCCs are connected to their base channel sills with 3/8" diameter bolts. Therefore, calculation of HCLPF due to plug weld capacity is removed in Revision 1 of this calculation. In Revision 0 of this calculation, the plug weld capacity considered base metal capacity consistent with requirements in AISC 360-10, which considers base metal shear capacity as a potential failure mode. In Revision 0 of this calculation, plug welds governed the anchorage capacity of MCCs. Welded connections are considered brittle connections per EPRI NP-6041-SL and therefore an inelastic energy absorption factor of 1.0 was assigned. Also, in Revision 1 of this calculation the anchorage capacity is governed by headed studs in concrete, which are also considered to have brittle failure mode and an inelastic absorption capacity of 1.0 is assigned. Revision 1 of the MCC fragility Calculation 2734294-C-201 includes the previously described expanded discussion and the updated anchorage evaluation.

f) Revision 1 of calculation 2734294-C-121 was issued to calculate a refined fragility for the BV1 Primary Plant DWST. To this end, horizontal loads are no longer combined with the 100-40-40 rule in consideration of the essentially axisymmetric tank shape. The BV1 walkdown report 2734294-R-004 Revision 1 clearly shows the BV1 Primary Plant DWST anchor chairs are not encased in concrete and therefore the last part of the peer review comment is not applicable.

g) The Seismic PRA Quantification Notebook now includes a group of sensitivities in Section 6 which address the models sensitivity to refinement of fragilities. These new cases only look at seismic components whose FVI is greater than 0.03; anything less is considered to not significantly change results even if the HCLPF values were improved. Those SSCs that had a FVI >0.03 had sensitivities performed in which the HCLPF was doubled, in order to bound the small changes in HCLPF values that would more realistically be expected. This was done for both CDF and LERF. In many cases, the sensitivity showed a small change in CDF/LERF, indicating that improving the fragility would have very little effect on the model. In the cases for which there is a noticeable change, the fragilities of those SSCs are deemed to be already realistic—or because they were refined following the peer review—and calculating a more robust fragility is not seen as plausible. Therefore, the PRA team concludes that there are no possible conservatisms in the fragility calculations that are driving the model results or masking insights. The RHR HXs identified in this F&O has a low FV, signifying that any additional refinement would not have a significant impact on the CDF/LERF.

h) The HCLPF for 2FWS-FCV479 was increased from 0.28g to 0.41g after locking the BV2 model. The summary table in Revision 1 of the BVPS-2 fragility Report 2734294-R-013 was updated to match the fragility reported in Revision 1 of Calculation 2734294-C-207. Also, the Seismic PRA Quantification Notebook now includes a group of sensitivities in Section 6 which address the models sensitivity to refinement of fragilities. These new cases only look at seismic components whose FVI is greater than 0.03; anything less is considered to not significantly change results even if the HCLPF values were improved.

Those SSCs that had a FVI >0.03 had sensitivities performed in which the HCLPF was doubled, in order to bound the small changes in HCLPF values that would more realistically be expected. This was done for both CDF and LERF. In many cases, the sensitivity showed a small change in CDF/LERF, indicating that improving the fragility would have very little effect on the model. In the cases for which there is a noticeable change, the fragilities of those SSCs are deemed to be already realistic—or because they were refined following the peer review—and calculating a more robust fragility is not seen as plausible. Therefore, the PRA team concludes that there are no possible conservatisms in the fragility calculations that are driving the model results or masking insights. Since the peer review, the fragility for the Flow Control Valve identified in this F&O was refined, and also has a low FV, signifying that any additional refinement would not have a significant impact on the CDF/LERF.

i) Revision 1 of Calculation 2734294-C-116 was issued to calculate a refined fragility for BAT-CHG1-5. To this end, an experience-based approach of 1.5 x Reference Spectrum was used to establish a functional fragility. The anchorage fragility is in excess of the functional fragility based on a review of the seismic characteristics of the component and its anchorage and walkdown photographs and observations documented in the walkdown report 2734294-R-004, Revision 1. The governing HCLPF based on the refined calculation was 0.70g.

j) The correct and final HCLPF value is 2.29g. The 0.30g value was originally submitted to the PRA modeler for its initial risk quantification using conservative assumptions. This fan subsequently showed as a top contributor and a more representative fragility of 2.29g was calculated. The 0.30g HCLPF value was incorrectly left in Revision 0 of the Fragility Report (2734294-R-006). This value has been corrected in Revision 1.

F&O 4-13

PRA Peer Review Fact & Observation 4-13 was identified in the Seismic Fragility Analysis High Level Requirement, Supporting Requirement SFR-A2 (and other affected Supporting Requirements SFR-F4, SPR-E6).

DETAILS (Peer Review Team)

The LERF model appears to be conservative with regard to the structural failures modeled in Top Event ZL2 that are mapped directly to CDF and LERF.

Building structures are important to LERF. Fragilities should be realistic.

BASIS FOR SIGNIFICANCE (Peer Review Team)

Structural failures in top event ZL2 are important contributors to LERF. However, it is not clear from the documentation how these failures cause core damage and containment failure. This is especially true for the MS Cable Vault structure (where it is not clear how core damage is guaranteed) and the containment (where the dominant failure mode is an internal wall, not a functional failure of containment).

The failure mode of buildings needs to be realistic. There is no explanation of how a failure of a single internal wall leads to gross failure of the Reactor/Containment Building Report 2734294-C-133, Revision 0, states the lowest HCLPF of the Reactor/Containment Building walls is 0.61g. This is an internal wall (690-INT-W2). This HCLPF is assigned as the gross failure mode of the Reactor/Containment Building.

There is a discrepancy in structural damping. Calc. 2734294-C-133 Fragility Analysis RCBX Section 7.2 Damping Factor states seismic demand is based on 7% structural damping. But Report 2734294-R-006 Section 7.2.4 Modeling of Structural Parameters states the structural damping of 4% is assumed based on the expected damage level. In the typical building response analysis, the 4% damping is used to be consistent with the CDFM approach. However, when the building structural responses obtained from the CDFM building analysis are used with the separation of variables approach, it is stated that the converted building responses are equivalent to response analysis results corresponding to 7% structural damping. For example, on Page 54 of 2734294-C-128 R0 BVPS1 Fragility Analysis AXLB, it is stated that the seismic demand is based on 7% structural damping. The basis for this when the 4% structural damping is actually used for the CDFM approach is not described.

Forces and moments for selected major shear walls and columns are provided in Tables A.I-1 and A.I-2 of 2734294-R-005, Part A. Then, these appear to be converted to median values for use with separation of variables and presented in Section 5.2 of Calc. 2734294-C-128 R0. It is not clear how this conversion was conducted. Please provide the process for how the CDFM-calculated demands were converted to the corresponding median demands.

A review of building fragility calculations shows that the variabilities associated with the following fragility parameters were not included:

- Horizontal Direction Peak Response
- Vertical Component Response
- Time History

No fragilities are calculated for floor diaphragms.

In shear wall fragilities, axial compression forces are neglected.

Forces and moments for selected structural components are provided as follows:

- 2734294-R-005, Revision 1, Part A, Attachment A.I for Auxiliary Building
- 2734294-R-005, Revision 1, Part B, Attachment B.I for Reactor Building
- 2734294-R-005, Revision 1, Part C, Attachment C.I for Diesel Generating Building
- 2734294-R-005, Revision 1, Part D, Attachment D.I for Fuel and Decontamination Buildings
- 2734294-R-005, Revision 1, Part E, Attachment E.I for Service Building
- 2734294-R-005, Revision 1, Part F, Attachment F.I for Main Steam Valve and Cable Vault Building
- 2734294-R-005, Revision 1, Part G, Attachment G.I for Intake Structure
- 2734294-R-005, Revision 1, Part H, Attachment H.I for Safeguards Building

All these include twisting moments in the summary tables. It is not described how the twisting moments are considered as part of building fragility evaluations.

Section 6.3 in 2734294-R-005, Revision 1, Part H states the following:

> "This approach conservatively assumes that all accelerations are co-directional and ignores the effects due to mode shapes. This conservative bias could be as high as about 50 percent in individual structural components, but it is considered acceptable because the fragilities of the structural components, such as reinforced concrete walls, are generally high and; therefore, will not contribute to the CDF (fragilities of other components will control). I f subsequent calculations determine otherwise, the specific structural components will be re-evaluated to obtain more accurate estimates of forces and moments. We anticipate that this will be accomplished by integrating stresses from the SASSI analysis."

This statement acknowledges conservatisms embedded in the seismic demands for Safeguards Building and justifies them based on the assumption that the corresponding building fragilities do not play a major role in the plant risk such as CDF. However, a review of top 10 contributors to LERF reveals that Safeguards Building is one of the three buildings that are ranked within the first top three contributors to LERF, along with main steam cable vault (MSCV) and Reactor

Containment Buildings. Therefore, the building fragilities for these three buildings need to be refined by eliminating the aforementioned conservatisms.

While the documents mentioned in this finding are from BV1, this observation also extends to BV2.

POSSIBLE RESOLUTION (Peer Review Team)

Review the dominant contributors to LERF to assure they are assessed as realistically as possible. Document the assumptions that are used to map the structural failures to CDF and LERF.

Provide basis that the lowest fragility of a component of a building represents the gross failure fragility of the building.

Correct the discrepancy in the description of structural damping.

PLANT RESPONSE OR RESOLUTION (ABS Consulting and RIZZO Associates)

In Revision 0 structural fragility calculations which were reviewed by the peer review team, a fragility was calculated for the limit state of structural deformation causing failure of equipment anchorage using ASCE 43-05 inelastic energy absorption factors for Limit State C. The failure of equipment supported within these structures will lead to core damage. The capacity calculated for structural deformation causing failure of equipment anchorage was also conservatively taken as representative for collapse. Collapse of the CIS or adjacent buildings such as the MS Cable Vault structure can be assumed to guarantee containment failure. Revision 2 of structural fragility calculations include a calculation of the capacity for the limit state of incipient collapse using ASCE 43-05 inelastic energy absorption factors for Limit State A. Revision 2 of structural fragility Calculations 2734294-C-128 through –C-135 and C-228 through -235 include expanded discussion of limit states and a calculation of collapse capacity.

In Revision 0 of the reactor building structural fragility calculation which was reviewed by the peer review team, a fragility was calculated for the limit state of structural deformation causing failure of equipment anchorage using ASCE 43-05 inelastic energy absorption factors for Limit State C. The failure of equipment supported within the reactor building will lead to core damage. The capacity calculated for structural deformation causing failure of equipment anchorage was also conservatively taken as representative for collapse. Collapse of the CIS can be assumed to guarantee containment failure. Revision 2 of the reactor building structural fragility calculation includes a calculation of capacity for the limit state of incipient collapse using ASCE 43-05 inelastic energy absorption factors for Limit State A. To address this F&O, discussion was added stating that the critical structural members for which fragilities are calculated are major walls and columns for which failure poses a potential gross loss of structural stability that could lead to collapse of the structure. Yielding of minor walls is not a concern since loads in these walls will be redistributed to the major shear walls. Internal wall 690-INT-W2 is categorized as a major shear wall. Failure of internal wall 690-INT-W2 according to the limit state of structural deformation causing failure of equipment anchorage leads to core damage. Failure of internal wall 690-INT-W2 according to the limit state of incipient collapse leads to large early release. Revision 2 of the reactor building structural fragility Calculations 2734294-C-133 (Unit 1) and 2734294-C-233 (Unit 2) include expanded discussion of failure modes, limit states and a calculation of collapse capacity.

As pointed out by the peer review team, 4% structural damping based on Response Level 1 was used to obtain the seismic structural response documented in Revision 1 of the Building Seismic Analysis reports 2734294-R-005 (Unit 1) and 2734294-R-012 (Unit 2) which is appropriate for development of ISRS for use in CDFM equipment HCLPF calculations. However, for evaluating forces and moments in structural members using the separation of variables method, a higher level of structural damping is permissible per ASCE 43-05. To address the finding, structural fragility Calculations 2734294-C-128 through C-135 and C-228 through C-235 were revised as follows. For fragility evaluation of the limit state of collapse used for LERF quantification, Response Level 3 structural damping of 10% was used for evaluating seismic-induced forces and moments in structures by elastic analysis as permitted by ASCE 43-05. For fragility evaluation of the limit state of structural deformation causing failure of equipment anchorage used for CDF quantification, structural damping was limited to Response Level 2 of 7% since the structure will be at a less degraded condition at the limit state which will cause incipient failure of wall mounted anchorage. The higher damping levels and associated variabilities were incorporated in to the fragility analysis via the Damping Factor, one of the Separation of Variables Structural Response Factors. This change results in a 32% higher seismic capacity for the limit state of structural deformation causing failure of equipment anchorage and a 58% higher seismic capacity for the limit state of collapse. Revision 2 of structural fragility Calculations 2734294-C-128 through C-135 and C-228 through C-235 include the above updates.

In response to this finding, structural fragility Calculations 2734294-C-128 through C-135 and C-228 through C-235 following the separation of variables methodology were revised to convert CDFM level demands defined at the 84th percentile NEP to median demand using the following approach. The seismic demand used in the structural fragility calculations reviewed by the peer review team was developed with 1 time history and BE soil properties in accordance with ASCE 4-98, which resulted in an approximately 84th percentile NEP structural response appropriate for CDFM evaluations. In order to achieve realistic structural fragilities, the 84th percentile NEP seismic forces and moments in the walls and columns were reduced by a median demand conservatism ratio factor based on EPRI Report 1019200 in the revised calculations. The median demand conservatism ratio factor was calculated using a seismic demand logarithmic standard deviation based on probabilistic SSI studies in literature. Structural fragility calculations following the separation of variables methodology were revised to reduce seismic forces and moments in the walls and columns by the median demand conservatism ratio factor to obtain a median response. As a result, structural fragilities increased by approximately 18%. Revision 2 of structural fragility Calculations 2734294-C-128 through C-135 and C-228 through C-235 include the above updates.

A detailed breakdown of the logarithmic standard deviations associated to each of the aforementioned factors is presented in the Revision 2 of the fragility calculations for each of the structures evaluated in the BVPS. It is noted that these calculations assume that variabilities associated with the Horizontal Direction Peak Response, the Vertical Component Response and Time History simulation do not contribute significantly to the log standard deviations in the seismic demand. In response to F&O 4-13, this assumption is re-examined as follows. The variability associated with the horizontal direction peak response accounts for the fact that the PGA in any one horizontal direction may exceed the geo-mean PGA used to base the fragilities. Although the SRSS method is used in calculating the total seismic shear in a wall, much of this shear is determined by the input motion parallel to the orientation of the wall. Therefore, the corresponding log standard deviation is taken to be 0.12 consistent with the recommendations in EPRI TR-103959. Because the vertical FIRS is site specific, the variability associated with the basic variable "vertical component response" is typically represented by βr in the range of 0.22 to 0.28 and βu less than about 0.2 (EPRI 103959). However, the effect of the vertical load on the wall shear capacity is relatively small (see also response to F&O 4-13). Therefore, the associated βr in seismic margin is relatively small (on the order of 0.01). The time histories used in the analysis closely match the target FIRS at 5% damping. The peaks and valleys are less than plus or minus 10% above or below the target FIRS at range of frequencies of 2.5Hz to 8Hz, near the fundamental frequency of the building; i.e., 4Hz. Thus, it is judged that a time history simulation factor is 1.0 and an uncertainty of 0.05 is used consistent with EPRI TR 103959. Also, Recent EPRI workshops have recommended that if only one time history is used in obtaining the 84th percentile response a random variability of 0.15 should be assigned to reflect effects of random phasing of the Fourier components on the resulting peak response. Revision 1 of the fragility analysis reports 2734294-R-006 (Unit 1) and 2734294-R-013 (Unit 2) include the discussion of these fragility analysis factors and the updated structural fragility parameters.

Floor diaphragm fragilities were considered to not govern over the in-plane shear and moment capacities of vertical structural members. The primary purpose of floor diaphragms part of the lateral force resisting system is to transfer lateral forces in a given floor into the vertical members of the lateral force resisting system. Typical floor slab thickness of BVPS buildings is 2 ft and longer spans are supported by beams composite with the slabs. Given the typical thickness and configurations of the floor diaphragms, it is judged their fragilities do not govern over in-plane shear or flexure fragilities of shear walls near the base resisting lateral forces accumulated from the stories above. Revision 1 of the fragility analysis reports 2734294-R-006 (Unit 1) and 2734294-R-013 (Unit 2) include the justification for the omission of floor diaphragm fragility evaluation.

In response to this finding, a representative structural fragility calculation was revised to demonstrate the effect of the axial compressive forces on shear wall shear capacity. The effect was found to be insignificant and therefore it was concluded the assumption to omit the effect from calculations remains valid. The other structural fragility calculations were revised to reference the representative calculation for the basis for omission of axial compressive load effect on shear wall shear capacity. Revision 2 of structural fragility Calculations 2734294-C-128 through C-135 and C-228 through C-235 include the above described updates.

To address this F&O, Calculation 12-4735-F-148, Revision 0 was prepared to elaborate and demonstrate how twisting moments reported in the Building Seismic Analysis Reports 2734294-R-005 (Unit 1) and 2734294-R-012 (Unit 2) affect building seismic fragilities documented in structural fragility Calculations 2734294-C-128 through C-135 and C-228 through C-235. The calculation clarifies that the reported twisting moments are the resultant of the distribution of out-of-plane shear forces on the elements that comprise the section cuts. Also, the calculation estimates the out-of-plane shear strength factor for both with and without the effects of the twisting moment for a representative BVPS structure. The strength factors are compared to the reported strength factors from the structural fragility calculation which are based on in-plane shear. Including the effects of the resultant twisting moments, the calculation demonstrates that the maximum out-of-plane shear is well within the shear capacity of the wall, and confirms that out-of-plane shear does not govern the wall fragility.

The justification for the approach to obtain forces and moments used as inputs to structural fragility calculations was clarified and augmented. The approach implemented to obtain the response quantities on the structural members uses the maximum absolute accelerations resulting from the SSI analyses in an equipment static analysis of the fixed-base structure. The equivalent static analysis is performed using the program SAP2000. The equivalent static analysis conservatively assumes that all response accelerations are co-directional and ignores the effects due to mode shapes. However, this is justified on the basis that the dominant mode shape is typically characterized by monotonically increasing shear displacements with height. The conservative bias could be as high as 50 percent for some structural components such as columns and other elements which may be influenced by local modes. The approach is further judged to be acceptable on the following basis. Fragility refinements were performed which increased the HCLPFs of the CDF related failure mode (i.e., building deformation causing equipment failure) by a factor ranging from 1.3 to 1.8. For LERF, a refined fragility was calculated (i.e., building collapse) which increases the HCLPF used in quantification by a factor ranging from 2.2 to 2.9. Considering these increase factors, the fragilities of structural components such as reinforced concrete shear walls are high and therefore are not expected to be significant contributors to CDF or LERF. The above described basis is documented in Revision 2 of structural fragility Calculations 2734294-C-128 through C-135 and C-228 through C-235 and Revision 2 of Building Seismic Analysis reports 2734294-R-005 (Unit 1) and 2734294-R-012 (Unit 2).

F&O 4-16

PRA Peer Review Fact & Observation 4-16 was identified in the Seismic Fragility Analysis High Level Requirement, Supporting Requirement SFR-A2 (and other affected Supporting Requirement SFR-F4).

DETAILS (Peer Review Team)

Containment building analysis for BV1 and BV2 is not realistic.

BASIS FOR SIGNIFICANCE (Peer Review Team)

On Page 18 of 2734294-R-005, Part B, the second paragraph states that the steel liner is anchored to the concrete inside surface at sufficiently close intervals so that the overall deformation of the liner is essentially the same that of the concrete wall; thus, performing as additional reinforcement. Then, on Page 28 of 2734294-R-005, Part B, the third paragraph further states that the steel lining on the internal face of the reinforced walls of the RCS was modeled by defining a concrete equivalent thickness; such that the moment of inertia per unit width results is equal to 0.5 the moment of inertia of concrete (cracked stiffness) plus the moment of inertia from the transformed steel lining area. The mass and weight densities are modified accordingly, to match the actual values for steel plus concrete.

As stated above, the steel liner is not explicitly treated in the analysis model and converted to the equivalent concrete thickness. Then, 2734294-R-005, Revision 1 Part B, Attachment B.I presents resulting section forces and moments for a section cut located at EL. 690 as follows, which is slightly less than the bottom of the steel liner elevation of EL 690 ft-11inches.

It is important to note that the obtained forces and moments in Tables B.I-1 and B.I-2 are consistent with the requirements of the CDFM approach. Thus, they need to be adjusted to be median-centered values when the separation of variables approach is used for building fragility evaluations. However, when Section 5.2 of 2734294-C-133 R0 is reviewed, it is found that the values from Tables B.I-1 and B.I-2 are directly copied and used in the fragility evaluation.

POSSIBLE RESOLUTION (Peer Review Team)

Based on these findings and observations, the following should be addressed:

Explain why the CDFM-related section forces and moments from Tables B.I-1 and B.I-2 of 2734294-R-005, Part B are directly used for the separation of variables fragility evaluation in Section 5.2 of 2734294-C-133 R0.

Explain why the twisting bending moments from Tables B.I-1 and B.I-2 of 2734294-R-005, Part B are completely ignored in Section 5.2 of 2734294-C-133 R0.

It appears that the obtained section cut forces presented in Tables B.I-1 and B.I-2 of 2734294-R-005, Part B are for the combined section of the concrete and the steel liner. This approach may be reasonable when the overall section capacity of the combined section is evaluated assuming the perfect composite action at the interface between the liner and the concrete section. However, this approach does not consider another potential mode associated with the liner itself such as rupturing due to excessive strain. This failure mode should be separately evaluated.

PLANT RESPONSE OR RESOLUTION (ABS Consulting and RIZZO Associates)

With respect to CDFM forces and moments, in response to this finding, reactor building structural fragility Calculations 2734294-C-133 (Unit 1) and 2734294-C-233 (Unit 2) were revised to convert CDFM level demands defined at the 84th percentile NEP to median demand using the following approach.

The seismic demand used in the structural fragility calculations reviewed by the peer review team was developed with one time history and BE soil properties in accordance with ASCE 4-98, which resulted in an approximately 84th percentile NEP structural response appropriate for CDFM evaluations. In order to achieve realistic structural fragilities, the 84th percentile NEP seismic forces and moments in the walls and columns were reduced by a median demand conservatism ratio factor based on EPRI Report 1019200 in the revised calculations.

The median demand conservatism ratio factor was calculated using a seismic demand logarithmic standard deviation based on probabilistic SSI studies in literature. Structural fragility calculations following the separation of variables methodology were revised to reduce seismic forces and moments in the walls and columns by the median demand conservatism ratio factor to obtain a median response. As a result, structural fragilities increased by approximately 18%.

Related to twisting moments, to address this F&O, Calculation 12-4735-F-148, Revision 0 was prepared to elaborate and demonstrate how twisting moments reported in the reactor building fragility calculations affect building seismic fragilities.

The calculation clarifies that the reported twisting moments are the resultant of the distribution of out-of-plane shear forces on the elements that comprise the section cuts. Also, the calculation estimates the out-of-plane shear strength factor for both with and without the effects of the twisting moment for a representative BVPS structure. The strength factors are compared to the reported strength factors from the structural fragility calculation which are based on in-plane shear. Including the effects of the resultant twisting moments, the calculation demonstrates that the maximum out-of-plane shear is well within the shear capacity of the wall, and confirms that out-of-plane shear does not govern the wall fragility.

Pertaining to the combined concrete and steel liner section, as pointed out by the peer reviewers, the steel liner is not explicitly treated in the analysis model and converted to the equivalent concrete thickness. This approach adequately captures the dynamic response of the steel liner / concrete shield.

For cylindrical shell structures such as the containment building, local shear or bending failures will not govern the capacity under seismic loading. Instead, global failure will govern where the whole cross section is engaged in shear or flexure eliciting a composite response. Thus, local failure of the steel liner is precluded under seismic loading.

Revision 1 of the fragility analysis reports (2734294-R-006 / 2734294-013) include the rationale for not evaluating rupture fragility of the containment steel liner.

ABSG CONSULTING INC.
300 Commerce, Suite 200
Irvine, CA 92602
Telephone 714-734-4242
Fax 714-734-4252

ABS Consulting

ABS GROUP OF COMPANIES, INC.
16855 Northchase Drive
Houston, TX 77060
Telephone 281-673-2800
Fax 281-673-2801

NORTH AMERICA

2100 Space Park Drive, Suite 100
Houston, TX 77058
Telephone 713-929-6800

Energy Crossing II, E. Building
15011 Katy Freeway, Suite 100
Houston, TX 77094

1525 Wilson Boulevard, Suite 625
Arlington, VA 22209
Telephone 703-682-7373
Fax 703-682-7374

10301 Technology Drive
Knoxville, TN 37932
Telephone 865-966-5232
Fax 865-966-5287

1745 Shea Center Drive, Suite 400
Highland Ranch, CO 80129
Telephone 303-674-2990

1390 Piccard Drive, Suite 350
Rockville, MD 20850
Telephone 301-907-9100
Fax 301-990-7185

3115 East Lion Lane, Suite 160
Salt Lake City, UT 84121
Telephone 801-333-7676
Fax 801-333-7677

140 Heimer Road, Suite 300
San Antonio, TX 78232
Telephone 210-495-5195
Fax 210-495-5134

823 Congress Avenue, Suite 1510
Austin, TX 78701
Telephone 512-732-2223
Fax 512-233-2210

55 Westport Plaza, Suite 700
St. Louis, MO 63146
Telephone 314-819-1550
Fax 314-819-1551

One Chelsea Street
New London, CT 06320
Telephone 860-701-0608

100 Danbury Road, Suite 105
Ridgefield, CT 06877
Telephone 203-431-0281
Fax 203-431-3643

1360 Truxtun Avenue, Suite 103
North Charleston, SC 29405
Telephone 843-297-0690

152 Blades Lane, Suite N
Glen Burnie, MD 21060
Telephone 410-514-0450

MEXICO

Ciudad del Carmen, Mexico
Telephone 52-938-382-4530

Mexico City, Mexico
Telephone 52-55-5511-4240

Monterrey, Mexico
Telephone 52-81-8319-0290

Reynosa, Mexico
Telephone 52-899-920-2642

Veracruz, Mexico
Telephone 52-229-980-8133

SOUTH AMERICA

Macaé, Brazil
Telephone 55-22-2763-7018

Rio de Janeiro, Brazil
Telephone 55-21-3179-3182

Sao Paulo, Brazil
Telephone 55-11-3707-1055

Viña del Mar, Chile
Telephone 56-32-2381780

Bogota, Colombia
Telephone 571-2960718

Chuao, Venezuela
Telephone 58-212-959-7442

Lima, Peru
Telephone 51-1-437-7430

Manaus, Brazil
Telephone 55-92-3213-9511

Montevideo, Uruguay
Telephone 5982-2-901-55-33

UNITED KINGDOM

EQE House, The Beacons
Warrington Road
Birchwood, Warrington
Cheshire WA3 6WJ
Telephone 44-1925-287300

3 Pride Place
Pride Park
Derby DE24 8QR
Telephone 44-0-1332-254-010

Unit 3b Damery Works
Woodford, Berkley
Gloucestershire GL13 9JR
Telephone 44-0-1454-269-300

ABS House
1 Frying Pan Alley
London E1 7HR
Telephone 44-207-377-4422

Aberdeen AB25 1XQ
Telephone 44-0-1224-392100

London W1T 4TQ
Telephone 44-0-203-301-5900

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Amsterdam, The Netherlands
Telephone 31-205-207-947

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Oslo, Norway
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Stavanger, Norway
Telephone 47-51-93-92-20

Trondheim, Norway
Telephone 47-73-900-500

ASIA-PACIFIC

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Navi Mumbai, India
Telephone 91-22-757-8780

New Delhi, India
Telephone 91-11-45634738

Yokohama, Japan
Telephone 81-45-450-1250

Kuala Lumpur, Malaysia
Telephone 603-79822455

Kuala Lumpur, Malaysia
Telephone 603-2161-5755

Beijing, PR China
Telephone 86-10-58112921

Shanghai, PR China
Telephone 86-21-6876-9266

Busan, Korea
Telephone 82-51-852-4661

Seoul, Korea
Telephone 82-2-552-4661

Alexandra Point, Singapore
Telephone 65-6270-8663

Kaohsiung, Taiwan, Republic of China
Telephone 886-7-271-3463

Bangkok, Thailand
Telephone 662-399-2420

West Perth, WA 6005
Telephone 61-8-9486-9909

INTERNET

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