Ameren
MISSOURI

Callaway Plant

July 10, 2020

ULNRC-06591

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

10 CFR 50.54(f)

Ladies and Gentlemen:

DOCKET NUMBER 50-483
CALLAWAY PLANT UNIT 1
UNION ELECTRIC CO.
RENEWED FACILITY OPERATING LICENSE NPF-30
SUPPLEMENTAL INFORMATION FOR RESPONSE TO MARCH 12, 2012 INFORMATION REQUEST, SEISMIC PROBABILISTIC RISK ASSESSMENT FOR RECOMMENDATION 2.1

References:

1. Ameren Missouri letter ULNRC-06524, "Response to March 12, 2012 Information Request, Seismic Probabilistic Risk Assessment For Recommendation 2.1." dated August 12, 2019 (ML 19225D321)
2. Ameren Missouri letter ULNRC-06551, "Supplemental Information For Response to March 12, 2012 Information Request, Seismic Probabilistic Risk Assessment For Recommendation 2.1," dated November 21, 2019 (ML 19325D662)

On August 12, 2019, Ameren Missouri provided the Callaway Plant Seismic Probability Risk Assessment (PRA) Summary Report to the NRC staff per Reference 1 above. The provided report was based on the fully quantified Seismic PRA model completed in March of 2019 for the Callaway Plant.

Subsequent to submittal of the report, Ameren Missouri continued to maintain and improve the Seismic PRA. A summary of the refinements/changes and their impact to the Seismic PRA results was provided to the NRC staff in Reference 2 above.

As discussed in Reference 2, the software platform for the Internal Events PRA model for the Callaway Plant was converted from WinNupra to CAFTA. This conversion also included a number of updates and refinements which included resolution of the open Facts and Observations (F&Os) on the Internal Events model. The Seismic PRA was then updated and "trued-up" to align with the Internal Events model. A full quantification was then performed on the Seismic PRA.

The Attachment to this letter is the revised Seismic PRA summary report reflecting the changes described above. The updates in this Attachment include additional refinements completed on the SPRA. For example, plant inspections were performed during Callaway's most recent refueling outage, and as a result, the contribution to Core Damage Frequency (CDF) from a Very Small Loss of Coolant Accident (VSLOCA) has been significantly reduced.

For any questions regarding this letter or its attachment, please contact Justin Hiller at 314-225-1141 or Bruce Huhmann at 573-694-6741.

I declare under penalty of perjury that the foregoing is true and correct.

Sincerely,

Executed on: 7/10/20

Stephanie Banker

Stephanie Banker
Vice President, Nuclear Engineering and Support

Attachment:

1. Callaway Energy Center Seismic Probabilistic Risk Assessment in Response to 50.54(f) Letter with Regard to NTTF 2.1 Seismic

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7/10/2020

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Callaway Energy Center Seismic Probabilistic Risk Assessment in Response to 50.54(F) Letter with Regard to NTTF 2.1 Seismic

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Executive Summary

In response to 10 CFR 50.54(f) letter issued by the NRC on March 12, 2012, a seismic probabilistic risk assessment (S-PRA) was performed for Callaway Energy Center. The S-PRA effort included performing a probabilistic seismic hazard analysis (PSHA) to develop seismic hazard and response spectra at the plant using state-of-the-art seismic source models and attenuation equations; seismic response analysis of structures, fragility analysis of structures, systems and components (SSCs); developing a logic model and performing risk quantification. The S-PRA effort underwent a final peer review by a team of experts. The comments of the reviewers were addressed and incorporated into the S-PRA as applicable.

The S-PRA identified risk-significant sequences and SSCs with their risk ranking and showed that the point estimate seismic core damage frequency (SCDF) is 5.59E-05 per year, and the seismic large early release frequency (SLERF) is 2.90E-06 per year.

Sensitivity studies were performed to identify critical assumptions, test the sensitivity to quantification parameters and the seismic hazard, and identify potential areas to consider for further reducing seismic risk. These sensitivity studies demonstrated that the model results were robust to the modeling and assumptions used.

1.0 Purpose and Objective

Following the accident at the Fukushima Dai-ichi nuclear power plant resulting from the March 11, 2011, Great Tōhoku Earthquake and subsequent tsunami, the Nuclear Regulatory Commission (NRC) established the Near-Term Task Force (NTTF) tasked with conducting a systematic and methodical review of NRC processes and regulations and to determine if the agency should make additional improvements to its regulatory system. The NTTF developed a set of recommendations intended to clarify and strengthen the regulatory framework for protection against natural phenomena. Subsequently, the NRC issued a 50.54(f) letter on March 12, 2012 [1], requesting information to assure that these recommendations are addressed by all U.S. nuclear power plants. The 50.54(f) letter requests that licensees and holders of construction permits under 10 CFR Part 50 re-evaluate the seismic hazards at their sites against present-day NRC requirements and guidance.

A comparison between the re-evaluated seismic hazard and the design basis for Callaway Energy Center (CEC) has been performed, in accordance with the guidance in EPRI 1025287, "Screening Prioritization and Implementation Details (SPID) For the Resolution of Fukushima Near Term Task Force Recommendation 2.1: Seismic" [10], and previously submitted to NRC [3]. That comparison concluded that the ground motion response spectrum (GMRS), which was developed based on the re-evaluated seismic hazard, exceeds the design basis seismic response spectrum in the 1 to 10 Hz range, and a seismic risk assessment is required. A seismic PRA (S-PRA) has been developed to perform the seismic risk assessment for CEC in response to the 50.54(f) letter, specifically item (8) in Enclosure 1 of the 50.54(f) letter.

This report describes the S-PRA for Callaway Energy Center (CEC), and provides the information requested in item (8)(B) of Enclosure 1 of the 50.54(f) letter [1]. The S-PRA 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 CEC, identifying which structures, systems, and components (SSCs) are important to seismic risk.

This report provides a summary regarding the S-PRA 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 because of the insights gained from the CEC S-PRA.

2.0 Information Provided in this Report

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

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

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 content of this report is organized as follows:

- Section 3.0 provides information related to the CEC seismic hazard analysis.
- Section 4.0 provides information related to the determination of seismic fragilities for CEC 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 S-PRA, including identified plant seismic issues and actions taken or planned.
- Section 7.0 provides references.
- Section 8.0 provides a list of acronyms used.
- Appendix A provides an assessment of S-PRA technical adequacy for response to NTTF 2.1 Seismic 50.54(f) Letter [1], including a summary of the CEC S-PRA peer review.

The SPID defines the principal parts of an S-PRA, and the CEC S-PRA has been developed and documented in accordance with the SPID. The main elements of the S-PRA performed for CEC 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.

Table 2-1: Cross-Reference for 50.54(f) Enclosure 1 S-PRA 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 SCDF and SLERF.	Sections 3.0, 4.0 and 5.0
2(I)	Methodologies used to quantify the seismic fragilities of SSCs, together with key assumptions.	Section 4.0
2(ii)	SSC fragility values with reference to the method of seismic qualification, the dominant failure mode(s), and the source of information.	Table 5-3
2(iii)	Seismic fragility parameters.	Table 5-3 provides fragilities for the top risk significant SSCs based on standard importance measures such as F-V or RRW.
2(iv)	Important findings from plant walkdowns and any corrective actions taken.	Section 4.2
2(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 S-PRA model and their motivation.	Sections 5.1 and 5.3
2(vi)	Assumptions about containment performance.	Sections 4.3 and 5.5
3	Description of the process used to ensure that the S-PRA is technically adequate, including the dates and findings of any peer reviews.	Appendix A
4	Identified plant-specific vulnerabilities and actions that are planned or taken.	Section 6.0

Table 2-2: Cross-Reference for Additional SPID Section 6.8 S-PRA Reporting

SPID Section 6.8 Item [(1)] Description	Location in this Report
A report should be submitted to the NRC summarizing the S-PRA 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. This report identifies key methods of analysis and referenced codes and standards.
The level of detail needed in the submittal should be sufficient to assess the sensitivity of the results to all key aspects of the analysis	Entirety of the submittal addresses this. Results sensitivities are discussed in the following sections: 5.7 (S-PRA model sensitivities) 4.4 Fragility screening (sensitivity)
The level of detail needed in the submittal should be sufficient to make necessary regulatory decisions as a part of NTTF Phase 2 activities.	Entirety of the submittal report addresses this.
It is not necessary to submit all of the S-PRA 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 S-PRA, with detailed information in lower tier documentation.
Documentation criteria for a S-PRA are identified throughout the ASME/ANS Standard [4]. Utilities are expected to retain that documentation consistent with the Standard.	This is an expectation relative to documentation of the S-PRA that the utility retains to support application of the S-PRA 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 CEC Seismic Hazard and Plant Response

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

The Callaway Energy Center (CEC), Unit 1 plant is a single Westinghouse 4-loop pressurized water reactor located in central Missouri approximately 10 miles southeast of Fulton and 25 miles east-northeast of Jefferson City. The CEC Unit 1 plant used the Standardized Nuclear Unit Power Plant Systems standard design, including seismic design. The regional and site (local) geology is described in additional detail in the Probabilistic Seismic Hazard Analysis (PSHA) completed to support the CEC Unit 1 S-PRA [2]. The CEC Unit 1 site ground surface is at elevation (EL) 840 feet (ft), which represents the control point for development of the Ground Motion Response Spectra (GMRS).

The geologic column underlying the CEC Unit 1 site consists of 30.5 ft of engineered backfill overlying a thick sequence of sedimentary rocks. The foundation material and foundation elevation for the CEC Unit 1 plant structures are described in Table 3-1. The geotechnical profiles developed to support the PSHA are developed using the extensive borehole and geophysical data gathered at the CEC Unit 1 site, and at the nearby site investigated to support a second unit at Callaway [2].

Table 3-1: Category 1 Structures and Geotechnical Foundation Material

Category 1 Structure	Geotechnical Foundation Material	Applicable Elevation (ft)
Reactor Building, Diesel Generator Building, Ultimate Heat Sink Cooling Tower	Engineered Backfill	829
Auxiliary/Control Building, Emergency Service Water Pumphouse	Sedimentary Rock	808.5
Alternate Emergency Power System (located ~7,500ft NW of nuclear island)	Accretion-Gley and Glacial Till Soils	832

3.1 Seismic Hazard Analysis

This section discusses the seismic hazard methodology, presents the final seismic hazard results used in the S-PRA and discusses 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 models, characterization of site response (e.g., soil and sedimentary rock column), and accounts for the uncertainties and randomness of these parameters to arrive at the site seismic hazard. The initial set of information regarding the CEC Unit 1 site hazard was provided to the Nuclear Regulatory Commission (NRC) in the seismic hazard information submitted in response to the NTTF 2.1 Seismic information request [3]. As further discussed below, an updated PSHA and site response analysis (SRA) were performed for the CEC Unit 1 site [2].

3.2 Seismic Hazard Analysis Methodology

The following method was used to perform the seismic hazard analysis.

The hard-rock PSHA is based on the Central and Eastern United States Seismic Source Characterization (CEUS-SSC) Project [4], and the Electric Power Research Institute (EPRI) Ground Motion Model (GMM) Review Project [5]. Both the CEUS-SSC and the EPRI GMM Update Projects were executed according to the latest seismic hazard guidelines as published in [6] and [7]. The CEUS-SSC Project was a Senior Seismic Hazard Analysis Committee (SSHAC) Level 3 project consistent with the guidance from [6] and [7]. The EPRI GMM Update Project was a SSHAC Level 2 update of a previous SSHAC Level 3 GMM.

Both projects considered available data, models and methods proposed by the larger technical community that were relevant to the hazard analysis. Both projects represent the center, body and range of technically defensible interpretations informed by the assessment of existing data, models and methods.

The uncertainties in source geometry, recurrence parameters, maximum magnitude and ground motion prediction equations are propagated in the hazard estimates through the logic tree formalism [2]. A review of published references and updated earthquake catalogs since the CEUS-SSC and EPRI GMM models were published found that those models continue to be appropriate for PSHA [2]. The CEUS-SSC Project included the development and use of a comprehensive earthquake catalog through 2008. An assessment of seismicity since 2008 determined that recurrence rates and maximum magnitudes from the CEUS-SSC model continue to be appropriate [2]. An evaluation of induced seismicity determined that the seismic hazard at the CEC Unit 1 site is not impacted by such events [2]. All credible seismic sources that may contribute significantly to the frequency of exceedance of vibratory ground motion at the CEC Unit 1 site are taken into account. The PSHA accounts for: (1) CEUS-SSC distributed seismicity source zones out to a distance of at least 640 kilometers (km) and (2) Repeated Large Magnitude Earthquake seismic sources within or near 1,000 km [2].

The effect of geologic deposits and geotechnical properties on ground motions at the CEC Unit 1 site are accounted for by performing a site-specific SRA. The site profiles used for the SRA were based on an updated review of available geologic and geophysical data characterized at the CEC Unit 1 site and the nearby site investigated for a potential second unit. Given these two sets of data, the CEC Unit 1 site is considered well characterized. The site profiles used for the SRA represent an update to the site profiles used for the NTTF 2.1 submittal [2] and [3]. In the NTTF 2.1 submittal [3] the upper ~30 feet of soil were modeled as glacial and post-glacial deposits. However, these materials were removed under the Nuclear Island at the CEC Unit 1 site and replaced by engineered backfill.

Thus, the Nuclear Island at the CEC Unit 1 site has 30.5 ft of engineered backfill and approximately 2,400 ft of sedimentary rocks over hard rock. Relatively flat-lying strata underlie the CEC Unit 1 site; therefore, a one-dimensional site response analysis was determined to be appropriate. Guidance for performing SRA is provided by [8], [9] and [10]; Reference [10] is referred to as the Screening, Prioritization and Implementation Details (SPID). The general approach outlined in the SPID is to define the uncertainty and variability in key site response input parameters as part of deriving the site amplification factor (AF) and its variability. Appendix B of the SPID provides direction for the assessment of both epistemic and aleatory uncertainties in key site response input parameters [10].

The SRA uses a logic tree approach to represent epistemic uncertainty in the shear-wave velocity (VS) profile, the geologic layer dynamic properties, the ground motion inputs and the upper crustal damping associated with the parameter kappa [2]. Three (3) base-case VS profiles were used for the SRA. The VS values, layer elevations and depths are shown in Table 3-2 and the VS profiles are shown on Figure 3-1 for the upper 350 ft. The SRA included two sets of non-linear dynamic properties modeled two ways accounting for non-linear and linear behavior [2]. For each base-case profile and dynamic property model, kappa values were quantified [2] to ensure that the range of kappa is consistent with the guidance from the SPID [10].

Table 3-2: Base-Case Median V_s Profiles Used for the CEC Unit 1 Site Response Analysis						
Top of Layer Elevation (ft)	Profile P1		Profile P2		Profile P3	
	Vs (fps)(1)	Depth To Top (ft)	Vs (fps)(1)	Depth To Top (ft)	Vs (fps)(1)	Depth To Top (ft)
840	1350	0	1100	0	1350	0
809.5	2337	30.5	2337	30.5	2337	30.5
781.7	4052	58.3	4052	58.3	4052	58.3
777.6	6045	62.4	6045	62.4	6045	62.4
767.2	3714	72.8	3714	72.8	3714	72.8
747.6	8358	92.4	8358	92.4	8358	92.4
707.8	6950	132.2	6950	132.2	6950	132.2
675	8319	165.0	7234	165.0	Hard Rock	165.0
	Hard Rock	2168.5	Hard Rock	2168.5		

Notes: (1) fps is feet per second.

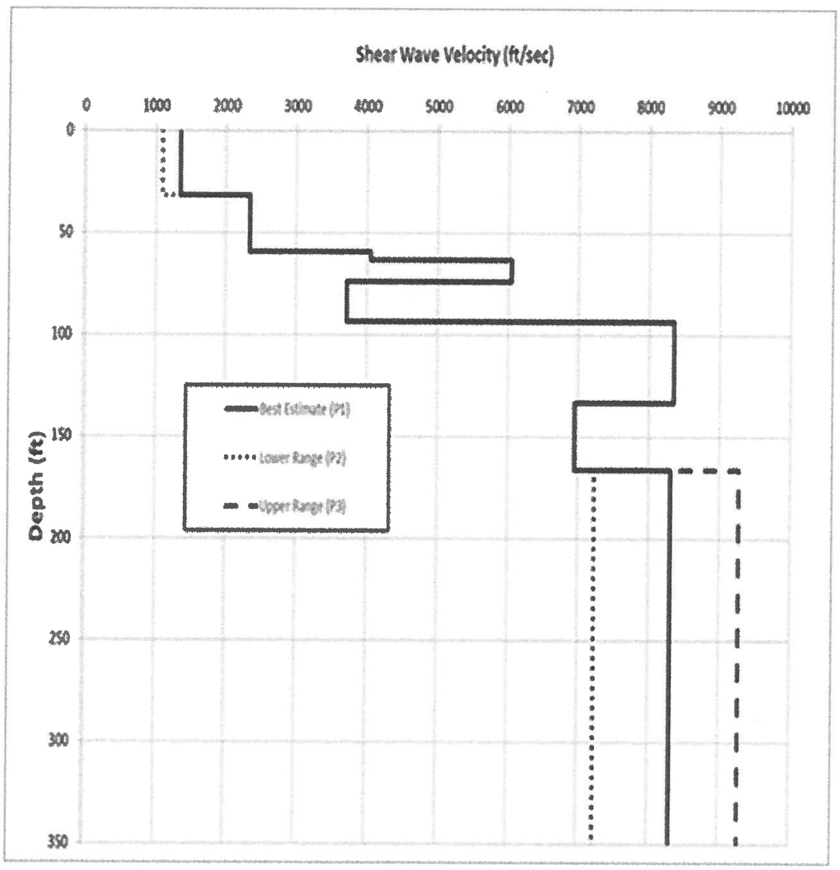

Figure 3-1: Base-Case Median Vs Profiles Down to a Depth of 350 ft Used for the CEC Unit 1 Site Response Analysis

Consistent with guidance from the SPID [10] the SRA is based on a random vibration theory approach using input ground motions that span a wide range of input amplitudes. The input ground motions used for the SRA were based on scaling three (3) deaggregation earthquakes (represented by magnitude and distance) defined at each of 11 mean annual frequencies of exceedance, for each of the seven spectral frequencies assessed as part of the hard rock PSHA. Consistent with the SPID [10], epistemic uncertainty in source spectra was modeled as either a single-corner or double-corner spectral shape. Appropriate weights are applied to each branch of the SRA logic tree.

Because the various structures at the CEC Unit 1 site are embedded to different depths, it is important that the SRA provide ground motions that are appropriate for use in SSI analysis and account for the potential influence of soil confinement above the foundation elevations. Guidance provided by [9] and [11] account for these conditions.

The SRA for shallow-embedment structures (EL 829 ft) was based on the full soil column (EL 840 ft), where the full set of strain-iterated properties is retained for each of the layers modeled. The soil column is then truncated at the foundation elevation (829 ft), and the SRA is repeated, using the strain-iterated properties from the full column; no further strain iteration is permitted. The AFs from the truncated column are used to derive the Foundation Input Response Spectra (FIRS) at EL 829 ft. In the PSHA report [2] these are designated as FIRS-1. In essence, the shallow-embedment structures are analyzed as surface structures.

The SRA for deeper embedded structures (EL 808.5 ft) is also based on the full soil column (EL 840 ft), where soil column outcrop motions at the foundation elevation (808.5 ft) are used to derive the FIRS. In the PSHA report [2] these are designated as FIRS-2. In essence, the deeper embedded structures are analyzed as embedded structures.

The results of the SRA consist of AFs that describe the amplification (or de-amplification) of reference rock motion as a function of spectral frequency and input reference rock ground motion amplitude. The AFs are represented in terms of a mean amplification value and an associated standard deviation for each spectral frequency and input rock ground-motion amplitude. For subsequent use in deriving hazard curves the AF values are retained at each end branch of the SRA logic tree. Consistent with the SPID [9], a minimum AF of 0.5 was employed when combining the SRA results with the hard-rock reference hazard results.

Figure 3-2 shows the AF results at EL 840 ft considering the overall weighted AF given the weights assigned to each branch of the SRA logic tree. Figure 3-3 shows similar results at EL 829 ft and Figure 3-4 shows the results at EL 808.5 ft. Several AF sensitivity comparisons are shown in the PSHA report [2] including the impact of each base-case profile, each of the two ground motion point-source models, and each of base-case sets of dynamic property curves. In general, the AF is most sensitive to the base-case profile, with profile P3 being different than P1 and P2 because the overall profile thickness is significantly less for P3 compared to either P1 or P2. A SRA sensitivity assessment was completed to address a peer review Fact & Observation in which additional V_S epistemic uncertainty was added to profiles P2 and P3 between depths of 30.5 and 165 ft. This sensitivity assessment demonstrated that the AF distribution is not impacted by adding in additional median V_S epistemic uncertainty over this depth range.

The site response results are combined with the hard-rock PSHA results to obtain hazard curves at elevations of interest that are consistent with the annual frequencies of exceedance of the hard-rock hazard curves [2]. The results are combined using Approach 3 of NUREG/CR-6728 [17], consistent with the methodology described in the SPID [10]. The combination of the hard-rock hazard results with the amplification factors is performed on a hazard curve by hazard curve basis for each individual seismic source and median ground motion model branch. Given the complexity of the logic tree used for the SRA, a review of the AFs was completed and ultimately two groups of AFs were selected to derive hazard curves at the surface. A hazard curve fractile sensitivity study was performed to assess the impact of the AF grouping process, in which a portion of the epistemic uncertainty is transferred to aleatory uncertainty. The sensitivity study showed that the AF grouping approach has essentially no impact on the mean hazard or on any of the hazard fractiles above the mean for all levels of ground motion. Also, there is no impact on fractiles below the mean hazard at ground motions less than approximately 1g [2].

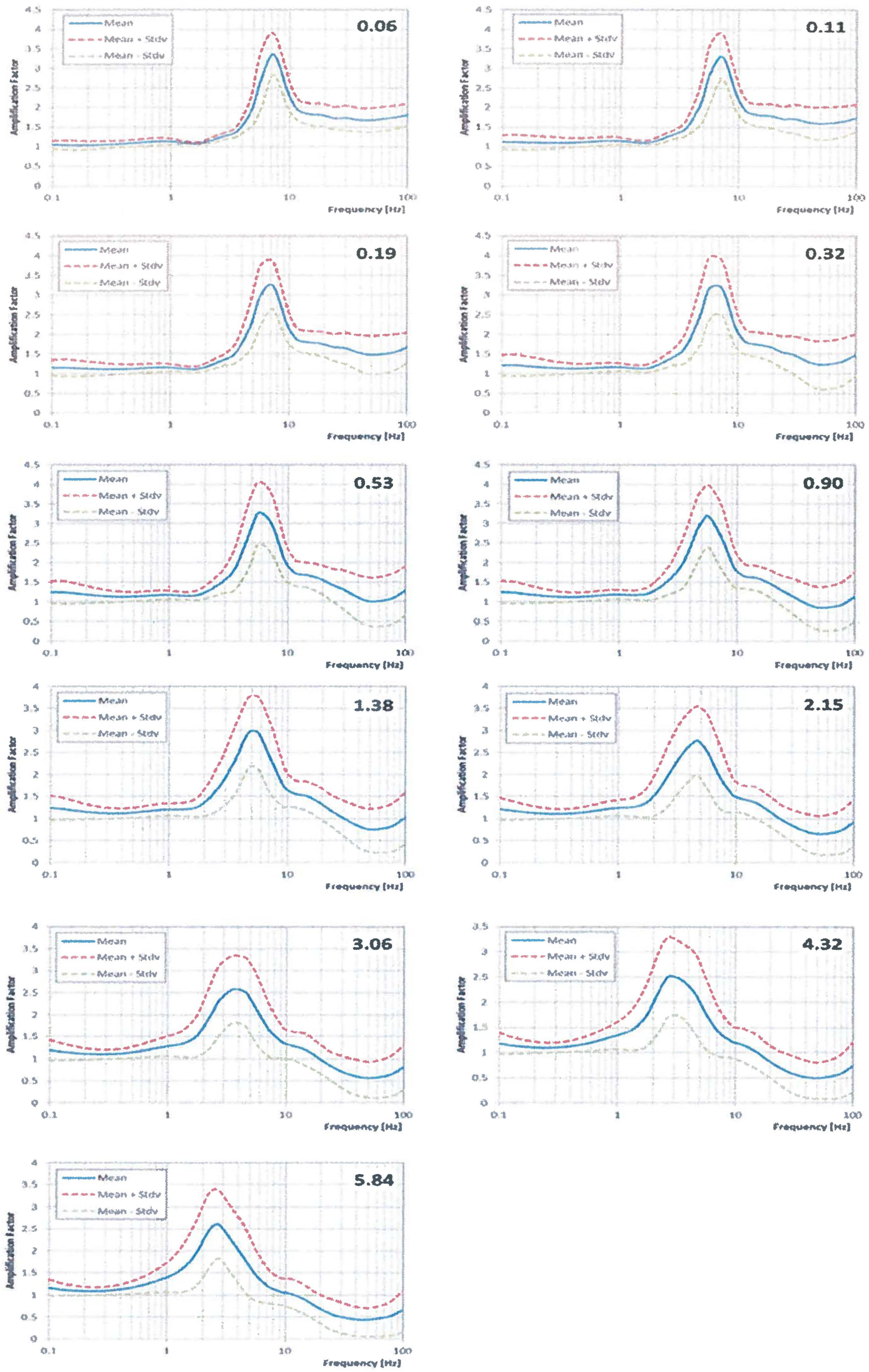

Figure 3-2: CEC Unit 1 Site Total Amplification Factors EL 840 ft

(Note: Quantities in the upper-right-hand corner represent the hard-rock input 100 Hz S_A in g's.)

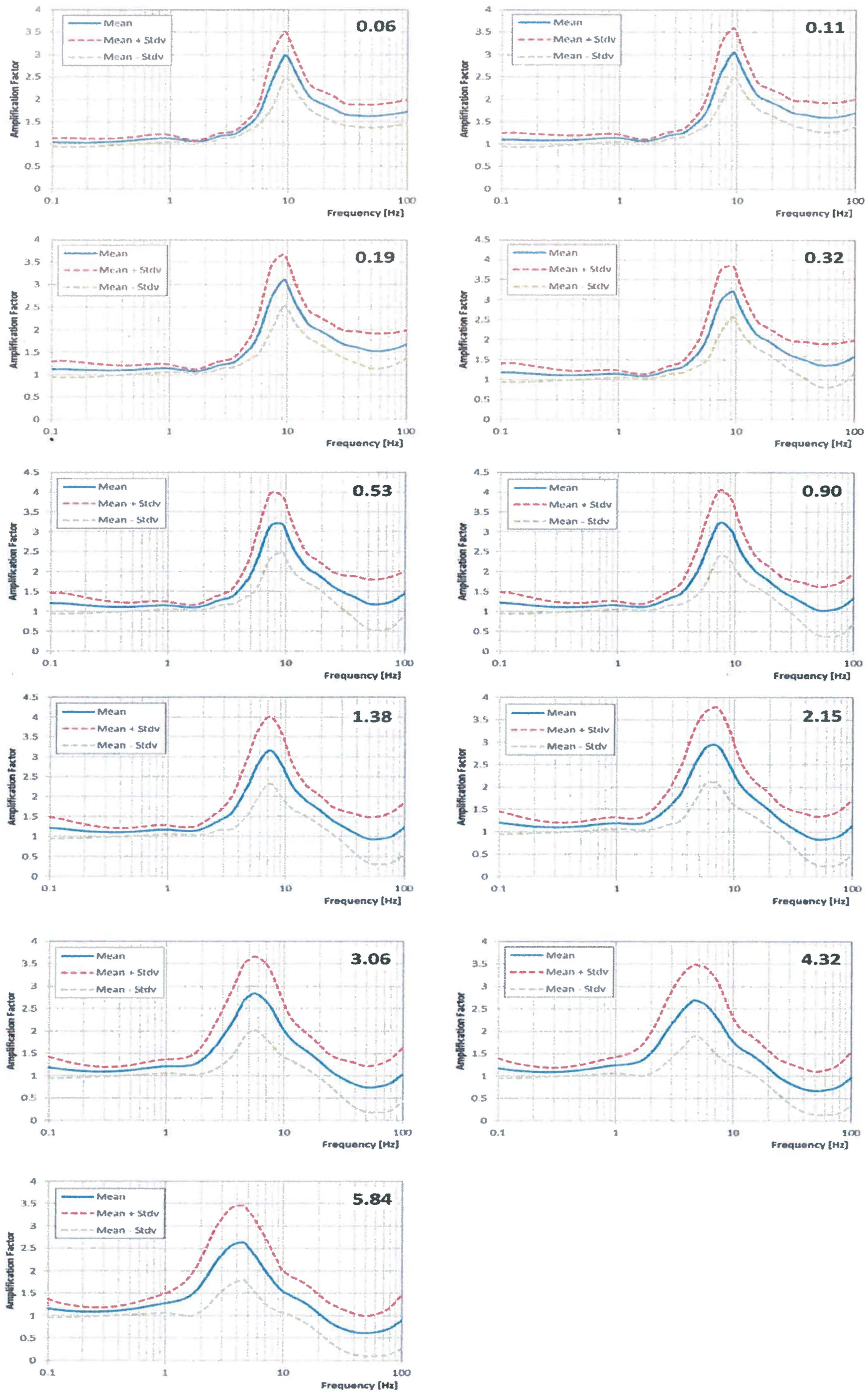

Figure 3-3: CEC Unit 1 Site Total Amplification Factors EL 829 ft

(Note: Quantities in the upper-right-hand corner represent the hard-rock input 100 Hz S_A in g's.)

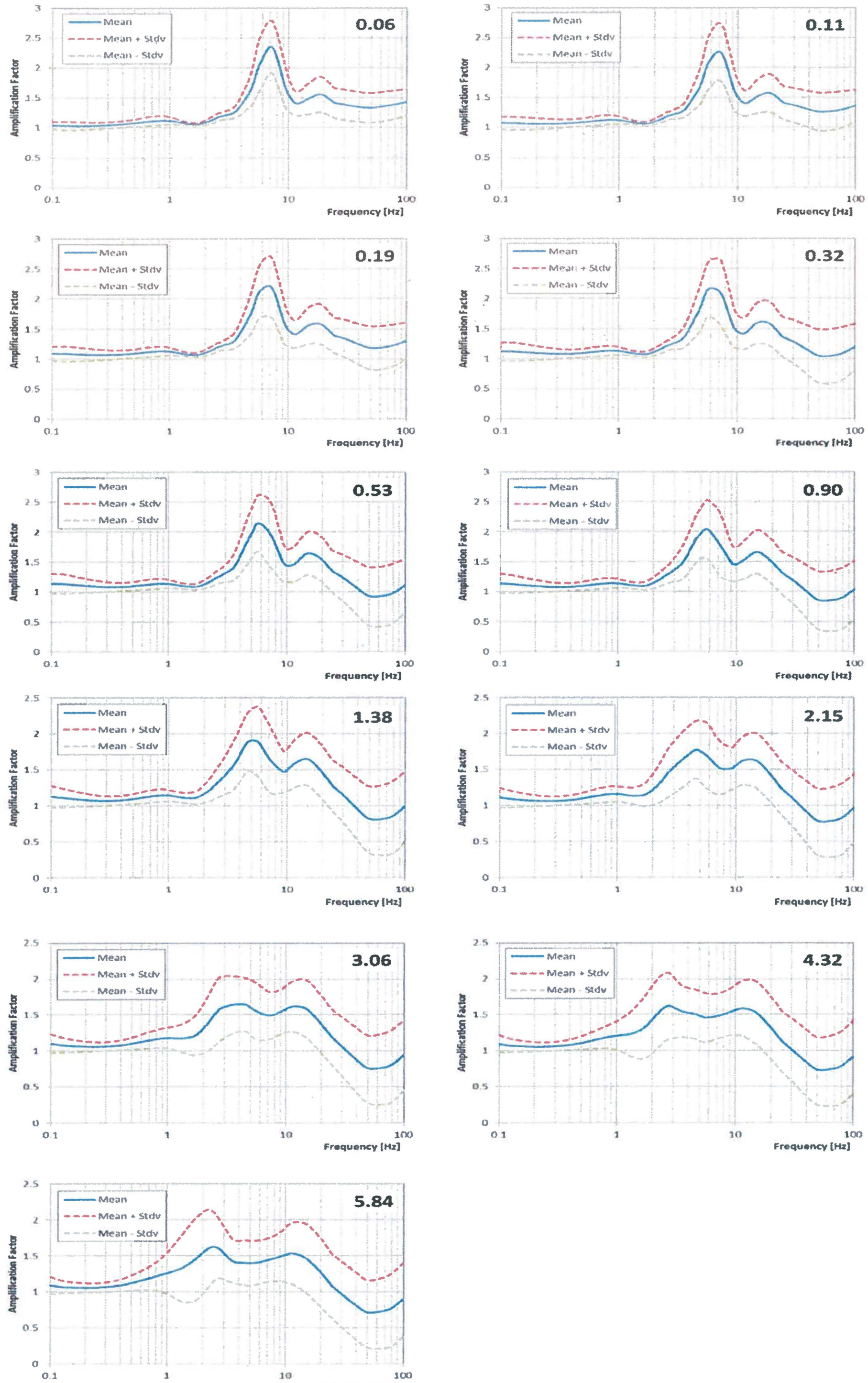

Figure 3-4: CEC Unit 1 Site Total Amplification Factors EL 808.5 ft

(Note: Quantities in the upper-right-hand corner represent the hard-rock input 100 Hz S_A in g's.)

The 100 Hz spectral acceleration (assumed to be peak ground acceleration) is the ground motion parameter used for risk quantification in the Seismic PRA. The 100 Hz spectral acceleration hazard curves (mean and fractiles) for EL 840 ft. are displayed in Table 3-3 and on Figure 3-5. The mean hazard curves for each of the seven spectral frequencies associated with the ground motion model are displayed on Figure 3-6.

Table 3-3: 100-HZ S_A Mean and Fractile Hazard Curves for the CEC Unit 1 Site at Ground Surface EL 840 ft

Spectral Acceleration [g]	Annual Frequency of Exceedance					
	MEAN	5TH	15TH	50TH	85TH	95TH
0.01	2.60E-02	9.33E-03	1.70E-02	2.53E-02	3.40E-02	5.05E-02
0.02	9.43E-03	3.40E-03	4.96E-03	8.14E-03	1.23E-02	2.52E-02
0.03	5.80E-03	1.66E-03	2.60E-03	4.76E-03	7.87E-03	1.71E-02
0.04	4.15E-03	9.33E-04	1.58E-03	3.23E-03	6.12E-03	1.23E-02
0.05	3.11E-03	5.77E-04	9.61E-04	2.31E-03	4.77E-03	9.47E-03
0.06	2.43E-03	3.69E-04	6.50E-04	1.69E-03	3.92E-03	7.80E-03
0.07	1.95E-03	2.60E-04	4.52E-04	1.24E-03	3.23E-03	6.73E-03
0.08	1.59E-03	1.73E-04	3.36E-04	9.25E-04	2.70E-03	5.81E-03
0.09	1.32E-03	1.27E-04	2.54E-04	7.20E-04	2.27E-03	4.95E-03
0.1	1.11E-03	9.06E-05	1.85E-04	5.68E-04	1.94E-03	4.35E-03
0.15	5.43E-04	3.02E-05	6.27E-05	2.16E-04	8.50E-04	2.45E-03
0.2	3.08E-04	1.43E-05	3.01E-05	1.08E-04	4.38E-04	1.47E-03
0.25	1.92E-04	7.90E-06	1.78E-05	6.45E-05	2.58E-04	9.06E-04
0.3	1.28E-04	4.96E-06	1.13E-05	4.14E-05	1.54E-04	6.12E-04
0.35	8.84E-05	3.55E-06	7.62E-06	3.01E-05	1.03E-04	4.22E-04
0.4	6.32E-05	2.48E-06	5.43E-06	2.14E-05	7.10E-05	2.94E-04
0.5	3.49E-05	1.42E-06	3.12E-06	1.23E-05	4.09E-05	1.53E-04
0.6	2.09E-05	7.99E-07	1.96E-06	7.46E-06	2.60E-05	8.53E-05
0.7	1.33E-05	5.05E-07	1.31E-06	4.88E-06	1.78E-05	5.33E-05
0.8	8.85E-06	3.48E-07	8.41E-07	3.49E-06	1.25E-05	3.67E-05
0.9	6.12E-06	2.36E-07	5.91E-07	2.40E-06	8.66E-06	2.44E-05
1	4.36E-06	1.62E-07	4.20E-07	1.76E-06	6.54E-06	1.73E-05
2	3.63E-07	7.20E-09	2.46E-08	1.32E-07	5.64E-07	1.49E-06
3	6.16E-08	6.14E-10	2.57E-09	1.77E-08	8.98E-08	2.64E-07
5	5.68E-09	1.59E-11	8.21E-11	9.33E-10	7.32E-09	2.63E-08
6	2.64E-09	4.42E-12	2.61E-11	3.53E-10	3.27E-09	1.30E-08
7	1.41E-09	1.46E-12	8.86E-12	1.46E-10	1.63E-09	6.81E-09
8	8.24E-10	5.44E-13	3.90E-12	6.66E-11	8.59E-10	4.13E-09
9	5.13E-10	2.29E-13	1.69E-12	3.46E-11	5.01E-10	2.55E-09
10	3.36E-10	9.54E-14	7.84E-13	1.77E-11	3.12E-10	1.77E-09

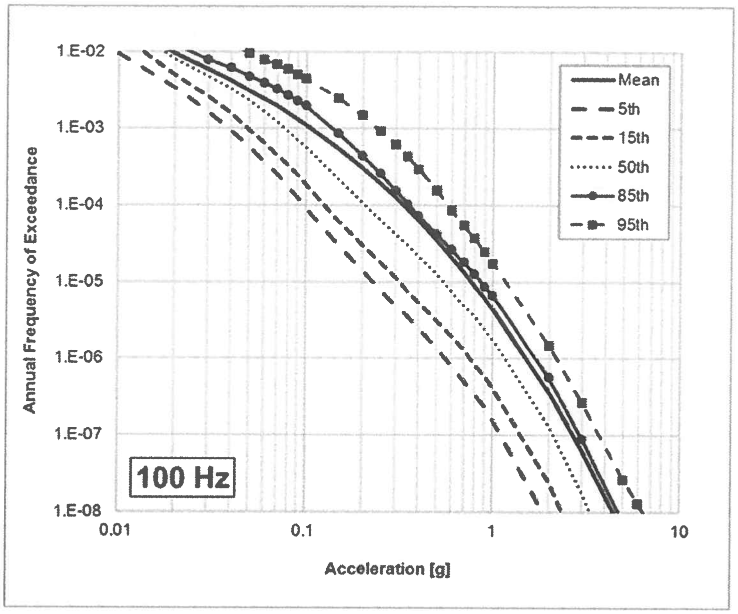

Figure 3-5: 100-HZ SA Mean and Fractile Hazard Curves for the CEC Unit 1 Site at Ground Surface EL 840 ft

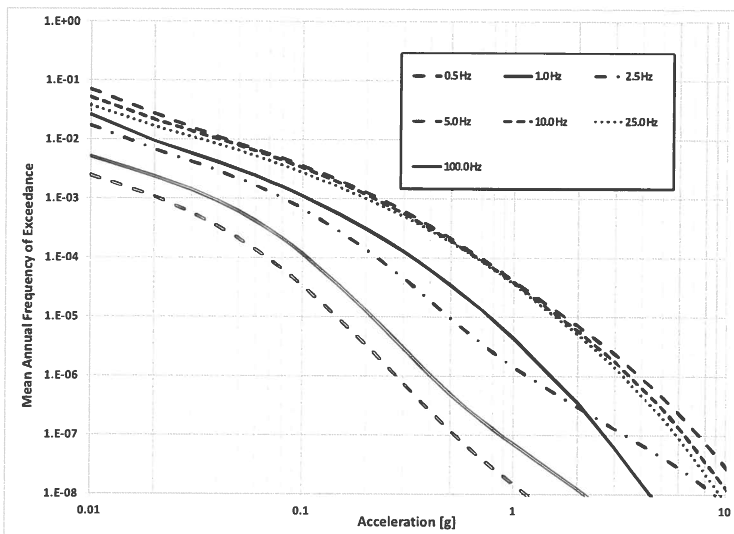

Figure 3-6: Mean Hazard Curves for the CEC Unit 1 Site at Ground Surface EL 840 ft

3.3 Seismic Hazard Comparisons and Insights

This section compares the surface control point hazard curves developed to support the NTTF 2.1 Seismic Hazard submittal [3] to that developed to support the S-PRA [2], and provides a brief summary of the PSHA technical adequacy and key uncertainties.

3.3.1 Comparison of Hazard Curves from NTTF 2.1 Seismic Hazard Submittal and PRA Site Response Analysis

As noted above, the site profiles used for the SRA were based on an updated review of the available geologic and geophysical data characterized at the CEC Unit 1 site. In the NTTF 2.1 submittal [2] the upper ~30 ft of soil was modeled as glacial and post glacial deposits. However, these materials were removed at the CEC Unit 1 site and replaced by engineered backfill which represents more competent material (increased stiffness and density). Because the engineered backfill is stiffer than the natural soils, the general trend was to result in decreases in the median AFs (lower impedance relative to the natural soils) which result in a decrease in the surface control point hazard curves.

Figure 3-7 compares the mean surface control point seismic hazard curves for spectral frequencies of 100.0, 10.0 and 1.0 Hz from the NTTF 2.1 Seismic Hazard submittal [3], assessed by the NRC [12], with the results from the Seismic PRA [2]. The relative decrease between the NTTF 2.1 submittal and the Seismic PRA is the result of lower impedance between the upper 30.5 ft of geologic material when compared to the underlying thick sequence of sedimentary rock. The median VS values for the engineered backfill, based on results from test pads of the backfill, are larger than for the natural soils that were removed at the site; median VS values of 1,100 to 1,350 feet per second (fps) for the engineered backfill [2] compared to median VS values of 400 to 625 fps for the upper 3.9 ft and 836 to 1307 fps for the remaining 26.2 ft of natural soils [3].

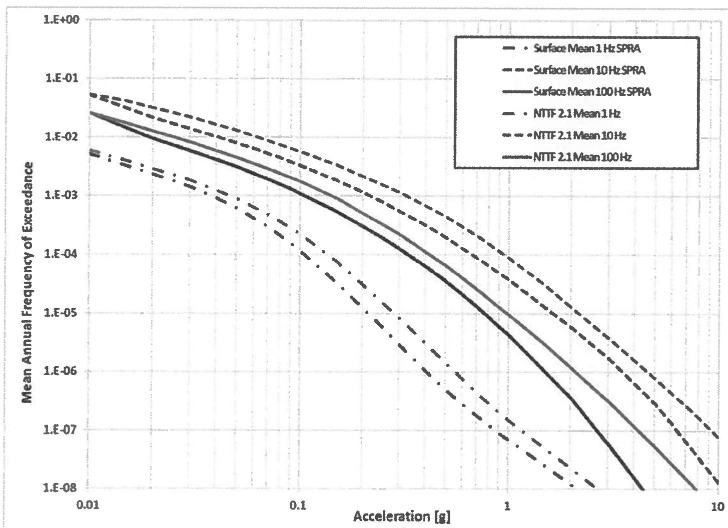

Figure 3-7: Mean Hazard Curves for the CEC NIT 1 Site at Ground Surface EL 840 ft

3.3.2 Probabilistic Seismic Hazard Analysis Technical Adequacy

The CEC S-PRA seismic hazard methodology and analysis was subject to an independent peer review against the pertinent requirements in the ASME/ANS PRA Standard [13]. The S-PRA was peer reviewed relative to Capability Category II for the full set of requirements in the Standard. After completion of the subsequent independent assessment, the full set of supporting requirements was met. The seismic hazard analysis was determined to be acceptable for use in the S-PRA. The peer review assessment, and subsequent disposition of peer review Facts and Observations (F&O) through an independent assessment, is further described in Appendix A and References [14] and [15].

3.3.3 Uncertainties in the Seismic Hazard Results from Input Parameters and Models

The PSHA results [2] were reviewed to identify and understand the sources of uncertainties and related assumptions that are important. Hazard assessment and sensitivity studies document the relative contribution to the total hazard by seismic source, the deaggregation of hazard by magnitude and distance, the impact on AF distribution for the SRA input uncertainties and the sensitivity to adding in additional epistemic uncertainty to the median V_S for the SRA site profiles.

The PSHA [2] includes an assessment of the hazard sensitivity to epistemic uncertainty in particular PSHA input variables (i.e., GMPE, seismicity of distributed sources, maximum magnitude of distributed sources, etc.), which is measured by the variance in the hazard contributed solely from epistemic uncertainty in a specific input variable, normalized by the variance in the total hazard. The results of this process are shown on Figure 3-8 which displays the variance deaggregation for the spectral frequency of 100 Hz (PGA) at the surface control point (EL 840 ft). Consistent with experience from several other PSHAs, the dominant contributor to the total variance is the epistemic uncertainty in the ground motion model. As the mean annual frequency of exceedance gets lower, the epistemic uncertainty in both maximum magnitude and the three magnitude-range cases used for deriving recurrence rates become more significant.

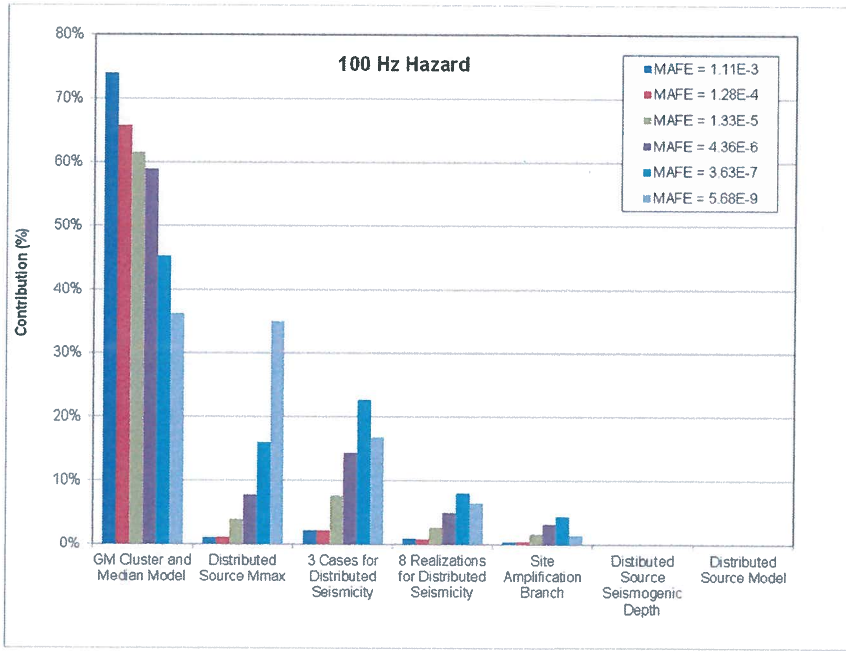

Figure 3-8: Variance Deaggregation of the CEC Unit 1 Site PSHA Logic Tree Inputs for the Spectral Frequency of 100 HZ

3.4 Horizontal and Vertical Response Spectra

This section provides the horizontal and vertical GMRS at the surface control point (EL 840 ft), the horizontal and vertical FIRS at foundation elevations 829 ft and 808.5 ft, and the horizontal response spectra at the location of the Alternate Emergency Power System (~7,500ft NW of CEC Nuclear Island).

3.4.1 Derivation of Vertical Response Spectra

The vertical response spectra were developed based on the corresponding horizontal response spectra, by scaling with appropriate vertical-to-horizontal (V/H) ratios. The derivation of V/H ratios follows the guidance developed by [16] and [17], in which generic V/H ratios are developed that can be used at nuclear power plants in the CEUS. The generic V/H ratios are adjusted to account for the CEC Unit 1 site conditions and the site-specific horizontal GMRS and FIRS. Two models were used. The first V/H model is based on interpolation between the V/H ratios for Western United States and CEUS rock site conditions from [17]. The second set of V/H values is based on empirical ground motion models. Consistent with the guidance from [16], the empirical relations from [18] and [19] were considered appropriate. The two models were used to derive a mean V/H ratio to establish a mean vertical GMRS and FIRS. The final shape and values of the recommended V/H ratios for the three elevations are consistent with the mean V/H models given by [16]. The mean V/H ratios are shown in Table 3-4.

Table 3-4: Recommended Mean V/H Ratios

Frequency (Hz)	V/H Ratio GMRS EL 840 ft	V/H Ratio FIRS-1 EL 829 ft	V/H Ratio FIRS-2 EL 808.5 ft	Frequency (Hz)	V/H Ratio GMRS EL 840 ft	V/H Ratio FIRS-1 EL 829 ft	V/H Ratio FIRS-2 EL 808.5 ft
0.10	0.694	0.697	0.697	3.56	0.694	0.697	0.697
0.13	0.694	0.697	0.697	4.52	0.694	0.697	0.697
0.16	0.694	0.697	0.697	5.00	0.694	0.697	0.697
0.20	0.694	0.697	0.697	5.74	0.694	0.697	0.697
0.26	0.694	0.697	0.697	7.28	0.694	0.697	0.697
0.33	0.694	0.697	0.697	9.24	0.694	0.697	0.697
0.42	0.694	0.697	0.697	10.00	0.694	0.697	0.697
0.50	0.694	0.697	0.697	11.72	0.694	0.697	0.697
0.53	0.694	0.697	0.697	14.87	0.694	0.697	0.697
0.67	0.694	0.697	0.697	18.87	0.694	0.697	0.697
0.85	0.694	0.697	0.697	23.95	0.735	0.729	0.733
1.00	0.694	0.697	0.697	25.00	0.747	0.741	0.744
1.08	0.694	0.697	0.697	30.39	0.807	0.799	0.798
1.37	0.694	0.697	0.697	38.57	0.870	0.860	0.853
1.74	0.694	0.697	0.697	48.94	0.899	0.889	0.880
2.21	0.694	0.697	0.697	62.10	0.895	0.887	0.883
2.50	0.694	0.697	0.697	78.80	0.858	0.852	0.855
2.81	0.694	0.697	0.697	100.00	0.806	0.804	0.813

3.4.2 Ground Motion Response Spectra at Elevation 840 FT

The horizontal and vertical GMRS at EL 840 ft are displayed on Figure 3-9 with the spectral acceleration values shown in Table 3-5.

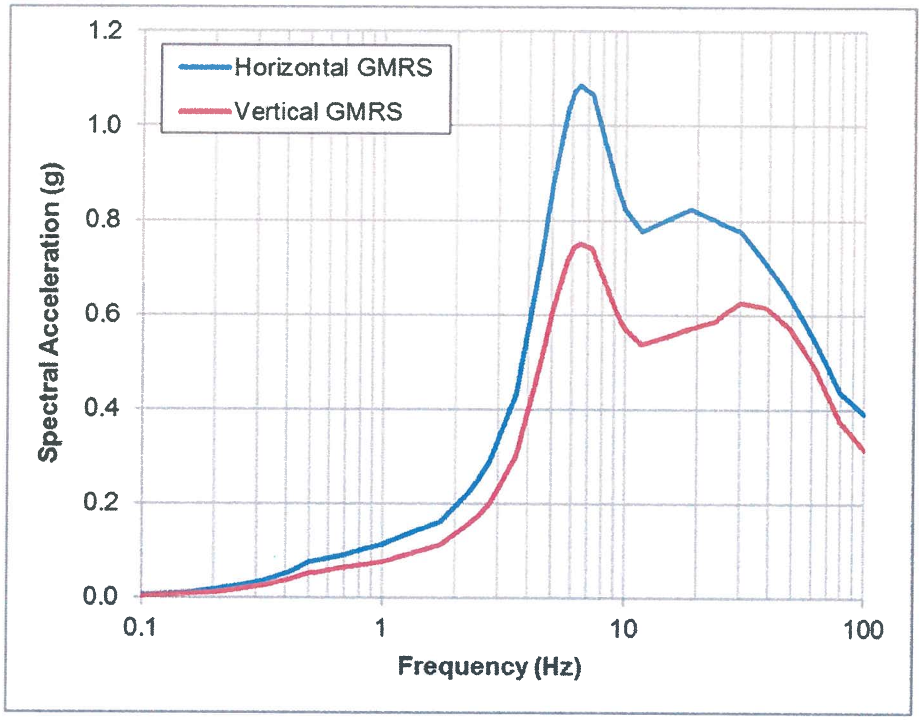

Figure 3-9: CEC Unit 1 Site Vertical and Horizontal GMRS at EL 840 ft

Table 3-5: CEC Unit 1 Site Vertical and Horizontal GMRS at EL 840 ft					
Frequency [Hz]	**Horizontal GMRS (S_A in g)**	**Vertical GMRS (S_A in g)**	**Frequency [Hz]**	**Horizontal GMRS (S_A in g)**	**Vertical GMRS (S_A in g)**
0.10	0.0058	0.0040	5.00	0.8716	0.6045
0.13	0.0086	0.0060	5.74	1.0303	0.7146
0.16	0.0126	0.0087	6.12	1.0716	0.7433
0.20	0.0182	0.0127	6.51	1.0843	0.7521
0.26	0.0265	0.0183	6.89	1.0759	0.7462
0.33	0.0385	0.0267	7.28	1.0661	0.7395
0.42	0.0563	0.0391	9.24	0.8770	0.6083
0.50	0.0754	0.0523	10.00	0.8236	0.5712
0.53	0.0782	0.0542	11.72	0.7774	0.5392
0.67	0.0901	0.0625	14.87	0.7992	0.5543
0.85	0.1042	0.0723	18.87	0.8246	0.5719
1.00	0.1115	0.0774	23.95	0.8007	0.5884
1.08	0.1192	0.0827	25.00	0.7947	0.5940
1.37	0.1402	0.0972	30.39	0.7766	0.6269
1.74	0.1623	0.1125	38.57	0.7081	0.6157
2.21	0.2173	0.1507	48.94	0.6368	0.5723
2.50	0.2530	0.1755	62.10	0.5449	0.4875
2.81	0.2912	0.2020	78.80	0.4365	0.3743
3.56	0.4305	0.2986	100.00	0.3899	0.3144
4.52	0.7234	0.5017			

3.4.3 Foundation Input Response Spectra at Elevation 829 FT

The horizontal and vertical FIRS-1 at EL 829 ft are displayed on Figure 3-10 with the spectral acceleration values shown in Table 3-6.

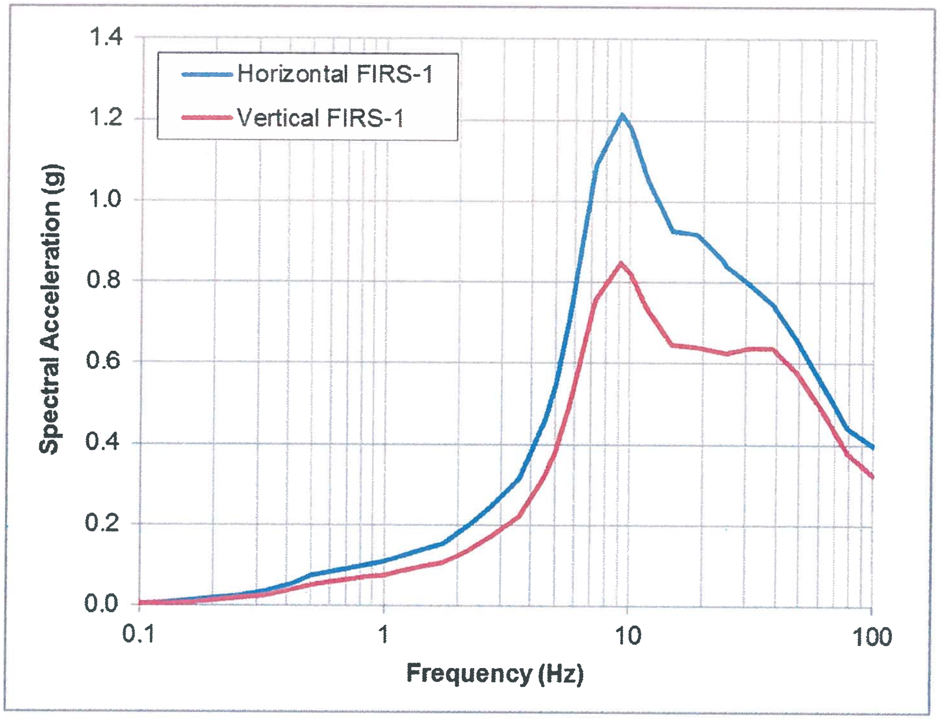

Figure 3-10: CEC Unit Site Vertical and Horizontal FIRS-1 at EL 829 ft

Table 3-6: CEC Unit 1 Site Vertical and Horizontal FIRS-1 at EL 829 ft					
Frequency [Hz]	**Horizontal firs-1 (S_A in g)**	**Vertical firs-1 (S_A in g)**	**Frequency [Hz]**	**Horizontal firs-1 (S_A in g)**	**Vertical firs-1 (S_A in g)**
0.10	0.0058	0.0040	3.56	0.3164	0.2205
0.13	0.0085	0.0059	4.52	0.4589	0.3198
0.16	0.0125	0.0087	5.00	0.5482	0.3820
0.20	0.0181	0.0126	5.74	0.7219	0.5030
0.26	0.0262	0.0183	7.28	1.0889	0.7588
0.33	0.0381	0.0266	9.24	1.2163	0.8476
0.42	0.0557	0.0388	10.00	1.1806	0.8227
0.50	0.0745	0.0519	11.72	1.0526	0.7335
0.53	0.0772	0.0538	14.87	0.9278	0.6465
0.67	0.0889	0.0620	18.87	0.9181	0.6398
0.85	0.1026	0.0715	23.95	0.8565	0.6242
1.00	0.1095	0.0763	25.00	0.8396	0.6221
1.08	0.1167	0.0813	30.39	0.7993	0.6390
1.37	0.1358	0.0947	38.57	0.7466	0.6420
1.74	0.1543	0.1075	48.94	0.6547	0.5818
2.21	0.1977	0.1378	62.10	0.5498	0.4874
2.50	0.2246	0.1565	78.80	0.4428	0.3774
2.81	0.2501	0.1743	100.00	0.3969	0.3190

3.4.4 Foundation Input Response Spectra at Elevation 808.5 ft

The horizontal and vertical FIRS-2 at EL 808.5 ft are displayed on Figure 3-11 with the spectral acceleration values shown in Table 3-7.

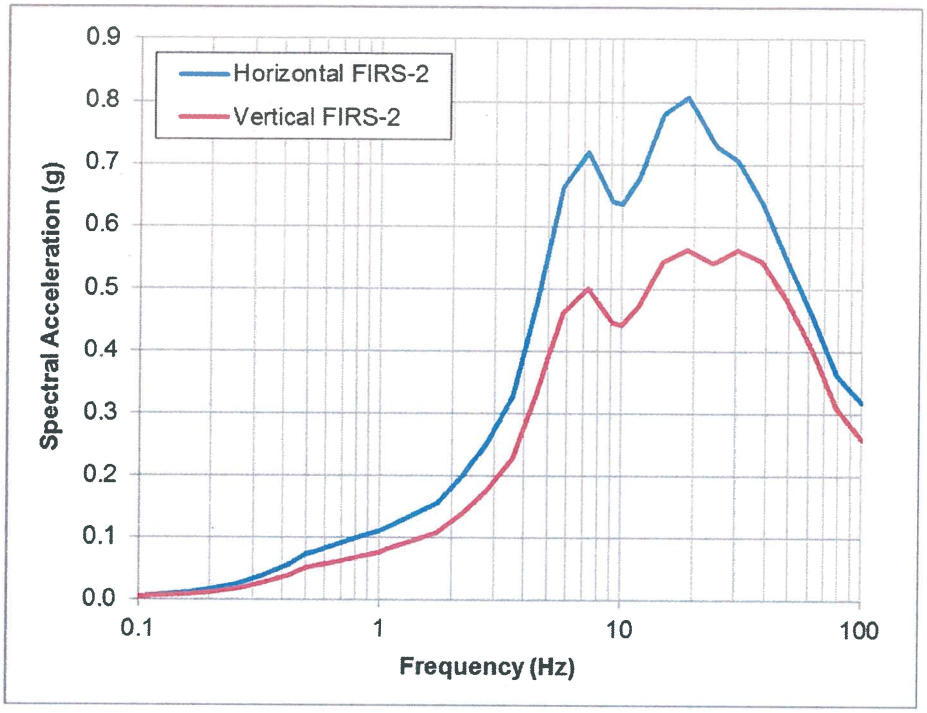

Figure 3-11: CEC Unit Site 1 Vertical and Horizontal FIRS-2 at EL 808.5 ft

Table 3-7: CEC Unit 1 Site Vertical and Horizontal FIRS-2 at EL 808.5 ft					
Frequency [Hz]	**Horizontal FIRS-2 (S_A in g)**	**Vertical FIRS-2 (S_A in g)**	**Frequency [Hz]**	**Horizontal FIRS-2 (S_A in g)**	**Vertical FIRS-2 (S_A in g)**
0.10	0.0058	0.0040	3.56	0.3262	0.2273
0.13	0.0085	0.0059	4.52	0.4834	0.3369
0.16	0.0124	0.0087	5.00	0.5601	0.3903
0.20	0.0181	0.0126	5.74	0.6620	0.4613
0.26	0.0262	0.0183	7.28	0.7192	0.5012
0.33	0.0381	0.0266	9.24	0.6390	0.4453
0.42	0.0557	0.0388	10.00	0.6356	0.4429
0.50	0.0745	0.0519	11.72	0.6763	0.4713
0.53	0.0772	0.0538	14.87	0.7794	0.5431
0.67	0.0890	0.0620	18.87	0.8077	0.5628
0.85	0.1027	0.0716	23.95	0.7390	0.5419
1.00	0.1094	0.0763	25.00	0.7293	0.5429
1.08	0.1167	0.0813	30.39	0.7050	0.5625
1.37	0.1359	0.0947	38.57	0.6384	0.5444
1.74	0.1549	0.1079	48.94	0.5460	0.4807
2.21	0.1989	0.1386	62.10	0.4582	0.4044
2.50	0.2264	0.1578	78.80	0.3628	0.3103
2.81	0.2524	0.1759	100.00	0.3185	0.2590

3.4.5 Response Spectra for the Alternate Emergency Power System

A horizontal GMRS was developed for the AEPS location at the CEC site. The AEPS is located about 7,500 ft northwest of the CEC Unit 1 Nuclear Island, and the AEPS foundation slab is at EL 832 ft, about 8 ft lower than the surface control point (EL 840 ft). The AEPS foundation overlies natural soils.

The AEPS GMRS was developed by deriving scale factors that represent adjustments needed to derive the AEPS GMRS relative to the surface control point (Nuclear Island) GMRS. These adjustments were derived by: (1) reviewing available geologic and geotechnical information, (2) developing a model for the subsurface below the AEPS location and (3) completing simplified site response which is used to derive the relative AF differences between the AEPS location and the Nuclear Island. Because the relative adjustments were based on simplified site response, the resulting AEPS GMRS was smoothed to account for site response input model uncertainties which were explicitly modeled when deriving the GMRS for the Nuclear Island surface control point. The AEPS smoothed horizontal GMRS is shown on Figure 3-8 with the spectral acceleration values shown in Table 3-8. For comparison, Table 3-8 lists the spectral acceleration values for the surface control point (EL 840 ft) and the unsmoothed and smoothed GMRS at the AEPS location (EL 832 ft).

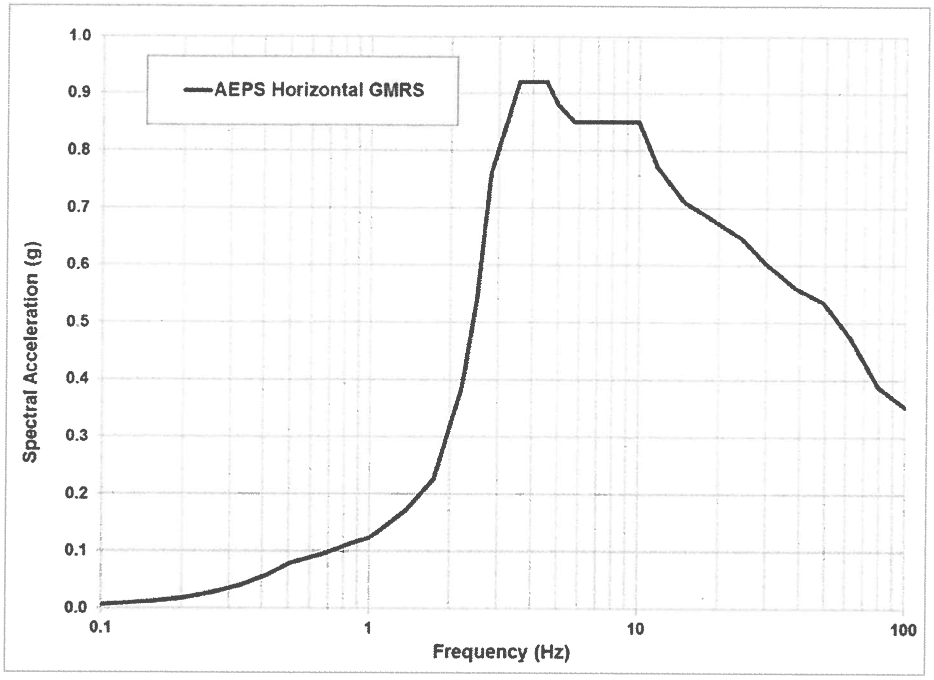

Figure 3-12: Horizontal GMRS for the AEPS Location at the CEC Unit 1 Site

Table 3-8: Horizontal GMRS at the CEC Unit 1 Site and at the AEPS Location

Frequency (Hz)	Ground Motion Response Spectra (g)		
	Nuclear Island Surface Control Point EL 840 ft	AEPS EL 832 ft Unsmoothed	AEPS EL 832 ft Smoothed
0.1000	0.0058	0.0063	0.0063
0.1269	0.0086	0.0092	0.0092
0.1610	0.0126	0.0133	0.0133
0.2043	0.0182	0.0190	0.0190
0.2593	0.0265	0.0274	0.0274
0.3290	0.0385	0.0397	0.0397
0.4175	0.0563	0.0581	0.0581
0.5000	0.0754	0.0782	0.0782
0.5298	0.0782	0.0813	0.0813
0.6723	0.0901	0.0951	0.0951
0.8532	0.1042	0.1128	0.1128
1.0000	0.1115	0.1240	0.1240
1.0826	0.1192	0.1348	0.1348
1.3738	0.1402	0.1710	0.1710
1.7433	0.1623	0.2266	0.2266
2.2122	0.2173	0.3864	0.3864
2.5000	0.2530	0.5407	0.5407
2.8072	0.2912	0.7615	0.7615
3.5622	0.4305	1.0515	0.9200
4.5204	0.7234	0.6886	0.9200
5.0000	0.8716	0.5870	0.8800
5.7362	1.0303	0.4512	0.8500
6.1219	1.0716	0.4187	0.8500
6.5076	1.0843	0.4286	0.8500
6.8933	1.0759	0.4617	0.8500
7.2790	1.0661	0.5162	0.8500
9.2367	0.8770	0.8798	0.8500
10.0000	0.8236	0.8877	0.8500
11.7210	0.7774	0.7712	0.7712
14.8735	0.7992	0.7101	0.7101
18.8739	0.8246	0.7040	0.6800
23.9503	0.8007	0.5652	0.6500
25.0000	0.7947	0.5243	0.6400
30.3920	0.7766	0.5272	0.6000
38.5662	0.7081	0.5604	0.5604
48.9390	0.6368	0.5363	0.5363
62.1017	0.5449	0.4749	0.4749
78.8046	0.4365	0.3892	0.3892
100.0000	0.3899	0.3529	0.3529

4.0 Determination of Seismic Fragilities for the S-PRA

This section provides a summary of the process for identifying and developing fragilities for SSC that participate in the plant response to a seismic event for the Callaway Energy Center S-PRA. The subsections provide brief summaries of these elements.

4.1 Seismic Equipment List

A seismic equipment list (SEL) was developed that includes those SSC whose seismic-induced failure could either give rise to an initiating event or degrade capability to mitigate a seismically-induced initiating event. The SEL was developed for the end states of core damage and large early release. The methodology used to develop the SEL is generally consistent with the guidance provided in the EPRI S-PRA Implementation Guide [27].

4.1.1 SEL Development

The SEL includes all plant components and structures whose seismic-induced failure could either give rise to an initiating event or degrade capability to mitigate a seismic-induced initiating event. A preliminary SEL is developed based on seismic-relevant portions of the IE PRA model. This preliminarily SEL is supplemented by a series of reviews intended to identify seismically risk-significant components not modeled by the IE PRA.

The steps followed in developing the full SEL are as follows:

- The internal event logic model was reviewed to identify all physical components associated with the modeled basic events. Equipment that is captured through "rule-of-the-box" considerations, e.g., equipment contained on a skid or in a cabinet, that can be subsumed into the major skid equipment or into the cabinet, were identified. For such equipment, the seismic fragilities for the containing equipment consider all the equipment in the "box." The rule-of-the-box components were included in the SEL and tracked by identification of the parent item. The resulting SEL applicable for walkdowns includes approximately 730 items not counting rule-of-the-box components.
- The Callaway IPEEE was included in the analysis to capture additional information associated with components on the SEL such as seismic classification and component locations.
- Seismically-induced equipment failures potentially resulting in an initiating event were added to the SEL.
- The main structures retained for evaluation in the S-PRA have been included in the SEL. The structures associated with the SEL equipment are the following:
    - Auxiliary Building (AB)
    - Control Building (CB)
    - Diesel Generator Building (DGB)
    - Essential Service Water System Pumphouse (ESWS)
    - Ultimate Heat Sink (UHS)
    - Reactor Building (RB)
    - Hardened Condensate Storage Tank (HCST)
    - Alternate Emergency Power System (AEPS)
- Instrumentation is not always explicitly modeled in the IE PRA due to the high level of redundancy. This screening criterion is potentially not applicable to a S-PRA because of the possibility of a seismic event inducing a common mode failure (i.e., full correlation of seismic failure) of similar equipment that would circumvent such redundancy. For this reason the more detailed

instrumentation review performed for the Callaway F-PRA was used to supplement the SEL developed in the first bullet.

- Internal events modeling often screens flow diversion paths based on the size of the potential diversion path. The internal events analysis uses a single failure as one of the criterion for screening. This screening criterion is potentially not applicable to a S-PRA because of the possibility of a seismic event inducing a common mode failure (i.e., full correlation of seismic failure) of similar equipment that would result in multiple diversion paths being impacted. Spurious opening of passive valves is not considered a realistic seismic-induced failure mode; thus the primary concern is the spurious opening of power operated valves associated with relay chatter. This is in essence the same concern of spurious actuation due to hot shorts in the F-PRA. The Callaway F-PRA was therefore reviewed for additional equipment that was included in the Multiple Spurious Operations (MSO) evaluation. Equipment identified in this analysis was added to the SEL.

- Containment penetrations are screened from explicit modeling in the IE PRA based on a number of criteria, included the size of the penetration. In case of a seismic event, it could be envisioned that multiple small bore lines could fail to be isolated due to a common mode, seismic-induced failure. For this reason, the containment penetrations were reviewed to assess if the screening criterion was still applicable in a seismic scenario. Spurious opening of passive valves is not considered a realistic seismic-induced failure mode; thus the main focus was on power operated valves that could be spuriously opened by seismic-induced relay chatter. Additional equipment was added to the SEL if the screening criteria were not applicable.

- ISLOCA pathways are screened from explicit modeling in the IE PRA based on a number of criteria, including the size of the interfacing lines. This screening criterion is potentially not applicable to a S-PRA because of the possibility of a seismic event inducing a common mode failure (i.e., full correlation of seismic failure) of similar equipment that would result in multiple valves opening. Spurious opening of passive valves is not considered a realistic seismic-induced failure mode; thus the main focus was on valves that could be spuriously opened by seismic-induced relay chatter. These valves have been reviewed to evaluate whether any additional equipment needed to be added to the baseline SEL.

- Distributed systems are not normally modeled in the IE PRA because of their low failure probability but are vulnerable to seismic-induced failure. Distributed systems have been added to the SEL in a representative fashion.
    - Inclusion of piping associated with fluid systems credited in the S-PRA.
    - Inclusion of cable trays associated with electrical systems credited in the S-PRA.
    - Inclusion of ducts associated with HVAC systems credited in the S-PRA.

- A number of checks have been run to ensure that individual assumptions or modeling simplifications in the internal events model would not result in incorrectly leaving out equipment from the analysis.

- All the input generated by the previous steps has been assembled into one summary table representing the Callaway S-PRA SEL

- The internal fire and internal flooding PRAs were used to gain insights on risk significant flood and fire. These fires and flood sources are not included in the SEL but are provided to the fragility team so that they can be investigated for additional seismic related vulnerabilities.

- A qualitative comparison with previously completed Seismic Equipment Lists.

4.1.2 Relay Evaluation/Spurious Breaker Trip Evaluation

During a seismic event, vibratory ground motion can cause relays to chatter. The chattering of relays potentially can result in spurious signals to equipment. Most relay chatter is either acceptable (does not impact the associated equipment), is self-correcting, or can be recovered by operator action. An extensive relay chatter evaluation was performed for the CEC S-PRA, in accordance with EPRI 1025287, "Screening, Prioritization and Implementation Details (SPID) [28] Section 6.4.2 and ASME/ANS PRA Standard [29] Section 5-2.2. Fragility analysis was performed for relays with the potential to impact SEL component functions. Those relays identified as having a significant impact on core damage frequency (CDF) and large early release frequency (LERF) were functionally screened. The evaluation resulted in most relay chatter scenarios screened from further evaluation based on no significant impact to component function. Table 4-1 lists relays with significant contributors to risk, along with their function and disposition in the S-PRA with appropriate seismic fragility or operator action.

An evaluation of spurious trips of breakers was also performed for low and medium voltage switchgear. The functionality of breakers was analyzed through evaluation of the site-specific seismic qualification testing performed for design basis and verification of EPRI NP-5223-SL [30] generic equipment ruggedness spectra (GERS) caveats. The major types of breakers at the plant are air breakers (medium voltage), draw-out type breakers (low voltage), and molded case circuit breakers. Molded case circuit breakers inherently have high seismic capacity. The switchgear fragility which house breakers were evaluated through EPRI NP-6041-SL [31] conservative, deterministic failure margin (CDFM) criteria. The seismic capacities used in this evaluation were based on the site-specific qualification testing and EPRI GERS in EPRI NP-5223-SL [30]. This evaluation addressed fragility for high frequency sensitive components as discussed in Section 6.4.2 of the SPID [28].

Table 4-1: Summary of Disposition of Risk Significant Relays

Relay	Function	Disposition
Relay_0.18DG	Can cause DG circuit breaker closure, affecting bus operation and leading to loss of AC power.	Fragility analysis performed and incorporated explicitly into the S-PRA model.
Relay_0.33	Linked to various systems and components	Fragility analysis performed and incorporated explicitly into the S-PRA model.

4.2 Walkdown Approach

This section provides a summary of the methodology and scope of the seismic walkdowns performed for the S-PRA. Walkdowns were performed by personnel with appropriate qualifications as defined in EPRI NP-6041-SL [31] Section 2 and the requirements in the ASME/ANS PRA Standard [29] Section 5-2.2. Each seismic review team (SRT) utilized for the S-PRA included seismic engineers with extensive experience in fragility assessment and seismic walkdown training. Walkdowns of those SSC included on the seismic equipment walkdown list were performed to assess the as-installed condition of these SSC for use in determining their seismic capacity (e.g. anchorage and lateral support), and performing initial screening, to identify potential II/I spatial interactions and look for potential seismic-induced fire/flood interactions. The fragility walkdowns were performed in accordance with the criteria provided in EPRI NP-6041-SL [31]. The information obtained was used to provide input to the fragility analysis and S-PRA modeling (e.g., regarding correlation and rule-of-the-box considerations).

The seismic fragility walkdowns were conducted on all accessible SEL equipment including equipment inside the RB. The fragility walkdowns included the evaluation of seismic interactions, including the effects of seismic-induced fires and flooding.

In addition to evaluating individual components and associated systems on the SEL, the walkdown reviewed the fire protection system. The fire protection piping was found to be well supported and not susceptible to anchorage failures.

A concern for seismic-induced fires is from flammable gases and liquids. Thus, the walkdown included these sources and their proximity to components on the SEL. Potential fires due to hydrogen piping in SEL buildings and transformers in SEL buildings are examples of scenarios that were evaluated. The potential for fire initiation due to seismic failure of high-voltage non-safety electrical cabinets was also investigated.

The potential for seismically-induced flooding was also evaluated. During the walkdowns, potential spray and flooding scenarios from piping systems and SEL components were reviewed. Flood sources, including the fire-protection system, were evaluated.

Pre-action fire protection systems were also considered and were investigated for seismic-induced flooding due to potential tripping of ionized particle smoke detectors for the case of nearby block walls failing, thereby producing dust in the air and initiating false alarming.

4.2.1 Significant Walkdown Results and Insights

Consistent with the guidance from EPRI NP 6041-SL [31], no significant findings were noted during the CEC seismic walkdowns. Components on the SEL were evaluated for seismic anchorage and interaction effects in accordance with SPID [28] guidance and ASME/ANS PRA Standard [29] requirements.

The walkdowns also assessed the effects of component degradation, such as corrosion and concrete cracking, for consideration in the development of SEL fragilities. In addition, walkdowns were performed on operator pathways, and seismic-induced fire and flooding scenarios were assessed, and potential internal flood scenarios were incorporated into the CEC S-PRA model. The walkdown observations were used in developing the SSC fragilities for the S-PRA.

4.2.2 Seismic equipment List and Seismic Walkdowns Technical Adequacy

The CEC S-PRA SEL development and walkdowns were subjected to an independent peer review against the pertinent requirements (i.e., the relevant SFR and SPR requirements) in the PRA Standard [32], also followed up by an F&O closure review.

The peer review assessment, and subsequent disposition of peer review findings, is described in Appendix A, and establishes that the CEC S-PRA SEL and seismic walkdowns are suitable for this S-PRA 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. The methodologies used to develop fixed-base and soil structure interaction (SSI) models, and to perform SSI analyses are discussed.

4.3.1 Fixed-base Analyses

All major CEC structures are founded on soil. Therefore, fixed-base analyses were not applicable. It is noted however that intermediate fixed-base modal analyses were performed to check the modeling fidelity when Lumped Mass Stick Models (LMSM) were recreated from design models, as well as for validation purposes.

4.3.2 Soil Structure Interaction (SSI) Analyses

All major structures are founded on or embedded in soil and therefore an SSI analysis was required for a realistic estimate of response. In addition to the structures, an SSI analysis was performed for the Refueling Water Storage Tank (RWST).

The ACS SASSI [33] program was used to perform the SSI analyses for the 3-D finite element model of the AB/CB, while the EKSSI program [34] was used to perform the SSI analyses on the LMSM of the RB, the DGB, the ESWS, the UHS, and RWST. Frequency-dependent impedance functions for layered media were used in all analyses.

Soil layers and properties are primarily based on the profiles developed in Probabilistic Seismic Hazard Analysis Seismic Probabilistic Risk Assessment Project, Callaway Energy Center, Unit 1 [35]. The SSI analyses used these soil layers, along with the Ground Motion Response Spectrum (GMRS) seismic hazard and the corresponding hazard-consistent strain-compatible properties provided in [35], to (a) generate time histories and soil column data for the Foundation Input Response Spectra FIRS-1 and FIRS-2 compatible to the GMRS, and (b) convert the generated time histories to in-structure response spectra (ISRS) for use as seismic input to the building models.

Modeling uncertainty was addressed by developing best-estimate (BE), upper bound (UB), and lower bound (LB) soil stiffness models. For all buildings, the general approach followed was that the soil stiffness uncertainty dominates all other uncertainties. An SSI response analysis was performed for each of the BE, UB and LB cases for the AB/CB, RB, DGB, ESWS, and UHS.

Ground motion input for the SSI analyses was based on the site-specific GMRS which is anchored to a 0.39g horizontal peak ground acceleration (PGA). A single set of artificial time histories (one time history per direction) was generated based on the criteria presented in ASCE 43-05 [36] Section 2.4 and NUREG-0800 Section 3.7.1 [37] Acceptance Criterion 1B, Option 1, Approach 2. Additional criteria used, include checks for statistical independence, strong-motion duration, power spectral density and Arias intensity, to ensure that the resulting time histories are suitable without any deficiencies of power across the frequency range of interest. The adequacy of the set of artificial time histories was further verified by comparing their ISRS to the average ISRS from SSI analyses using 5 sets of time histories generated from real earthquake seeds and spectrally matched to the site-specific GMRS.

A series of sensitivity studies were performed to check the validity of stiffness variability and cracking assumptions.

The generated ISRS have a 84% non-exceedance probability and are suitable for CDFM analysis per the criteria of EPRI NP-6041-SL [31] Section 2 and EPRI 1019200 [38] Appendix A. ISRS with amplified narrow frequency content were clipped for comparison to broad-banded test response spectra, typical of most nuclear power plant components. The guidance in EPRI TR-103959 [39] Section 3 was followed for the peak clipping process.

4.3.3 Structure Response Models

The ANSYS [40] finite element program was used to develop the AB/CB 3-D finite element fixed-base model, while the GTSTRUDL software [41] was used to develop the fixed-base LMSM of the remaining building structures (RB, DGB, ESWS, and UHS). The modeling effort was guided by these primary goals:

- Models are to provide the capability to estimate realistic seismic demand on in-scope SSC.
- Models are to satisfy review criteria of EPRI 1025287, "Screening, Prioritization and Implementation Details" [28], Section 6.3.1.

The Auxiliary Building and the Control Building share a common foundation; therefore, they were modeled and analyzed together. All other buildings are on independent foundations; therefore, they were modeled and analyzed as independent structures.

A detailed 3-D finite element fixed-base model was developed for the AB/CB to capture its irregular geometry with respect to lateral load paths, as well as the global lateral load path, the vertical flexibility of floor slabs and horizontal flexibility of floor diaphragms, which would be difficult to realistically model with a LMSM. The finite element model was developed from plant drawings and related documents. Each floor of the building above the foundation is constructed of reinforced concrete slabs supported by reinforced concrete walls and columns. The slabs were modeled using shell elements, whereas steel beams and concrete columns were modeled using beam elements. Furthermore, the AB/CB contain a significant amount of active SEL electrical and mechanical equipment at multiple elevations including batteries, switchgear, and transformers. The mass of these equipment and their distribution on its floor was accordingly accounted for in the finite element model.

Fixed-base LMSM for the RB, DGB, ESWS and UHS were developed from plant design documents. Specifically, design-basis fixed-base LMSM were obtained for these structures and enhanced as needed to more realistically assess the demands on equipment. Enhancements included adding elements to represent the load path through concrete fill above the foundation, adding outrigger nodes to automatically capture torsional and rocking effects in time histories, and enhancing torsional response capabilities. Enhanced LMSM met the modeling criteria of SPID [28] Section 6.3.1 and were considered suitable for S-PRA response analysis. Criteria were verified by a review of drawings, review of plant design basis calculations, and by supplemental calculations.

Concrete cracking was considered for both the 3-D finite element model and the LMSM, per the guidance of ASCE 4-16 [42]. To determine cracking, the design shear stresses on each floor were checked against the allowable code limits. In cases when the stresses were determined to exceed the code the stiffness of the affected elements was reduced accordingly.

The guidance of ASCE 4-16 [42] Section 3.2 was applied to properly select structural damping for the buildings. The applied damping values are intended to have a conservative bias to meet the intent of EPRI NP-6041-SL [31] Table 2-5 for estimation of 84% non-exceedance probability in-structure response. Buildings are primarily reinforced concrete shear-wall structures and applied damping is typically in the range of 4% to 7%. For SSI analysis, the effective foundation damping is implicitly accounted for through the frequency-dependent impedance functions used.

A simplified LMSM was also developed for the HCST to capture the seismic response using SSI analysis. Effective masses for the sloshing and impulsive modes were per EPRI NP-6041-SL [31] Appendix H. A 5% structural damping value was applied per EPRI NP-6041-SL [31].

4.3.4 Seismic Structure Response Analysis Technical Adequacy

The CEC S-PRA Seismic Structure Response and Soil Structure Interaction Analysis were subjected to an independent peer review against the pertinent requirements in the PRA Standard [32].

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

4.4 SSC Fragility Analysis

The primary goal of the fragility analysis was to identify and analyze critical SSC. A critical SSC item is one that ranks low in terms of seismic ruggedness and is also important to plant safety. A screening process was followed to select SSC for analysis. The process included review of the plant seismic design bases, performance of seismic walkdowns, and application of industry practice related to seismic margin and S-PRA studies.

The screening evaluation was performed for the 1.2g spectral acceleration level, which corresponds to the 2nd screening lane of EPRI NP-6041-SL [31] (refer to EPRI NP-6041-SL [31], Table 2-3 and 2-4). The seismic capacity inferred by this value is:

Sc $\geq$ 1.2g high confidence of low probability of failure (HCLPF) seismic capacity, as spectral acceleration

The 1.2g value associated with screening is applicable to the 5% spectral acceleration at the ground across a broad frequency range (from about 2 to 8 Hertz per EPRI NP-6041-SL [31] Page 2-44). Spectral accelerations above this range are less damaging and are less important to both structures and equipment.

A HCLPF value for screened-out equipment and structures was determined by comparing the spectral peak of the GMRS to the above screening capacity. A combined logarithmic standard deviation βc (for randomness and uncertainty) was assigned per the SPID [28] guidance.

The guidelines of EPRI NP-6041-SL [31] Table 2-3 were applied for screening of civil structures. The guidelines of EPRI NP-6041-SL [31] Table 2-4 were applied for screening of equipment and subsystems. After initial screening, an item was either screened-in (a fragility was performed) or screened-out (a surrogate value was given).

Analysis was performed for all screened-in items. In addition, analysis was performed for some screened-out equipment to increase the S-PRA model capability. The resulting sample of SSC is broad and supportive of a robust S-PRA model. Some screened-out items required verification of issues that could not be resolved by design review or walkdown. A set of action items was created to track items designated for fragility analysis and to resolve outstanding screening issues. EPRI reports TR-1019200 [38], TR-103959 [39] and NP-6041-SL [31] were used as the basis for calculation of seismic fragility parameters. The lognormal model was used.

Screened-out items were those judged to have a relatively high seismic fragility. Fragility parameters for this category of SSC were addressed by a surrogate element(s) similar to that described in EPRI report TR-1019200 [38]. As an option to increase S-PRA model capability, seismic fragility parameters applicable to an individual screened-out equipment item were also provided. With this option, the reliance on surrogate elements may be reduced.

Table 4-2: summarizes key attributes of the fragility analysis. More details are provided in following sections and in supporting project documents.

Table 4-2: Key Attributes of CEC Fragility Analysis

Attribute	Description
Seismic equipment list	The SEL was provided by the plant response model team to the fragility analysis team. The SEL was the basis for the scope of SSC addressed by the plant fragility analysis. The above process ensured that the fragility analysis addressed all SSC explicitly or implicitly credited in the plant response model.
Screening process	The screening process of EPRI NP-6041-SL [31] was applied. This process is widely used in the nuclear power industry for S-PRA and similar beyond-design-basis studies.
Screening level	The project earthquake screening level was the 1.2g spectral acceleration level identified in EPRI NP-6041-SL **[10]** Section 2. This screening level was found to be sufficient for evaluation of structural failure modes of buildings and passive equipment.

Table 4-2: Key Attributes of CEC Fragility Analysis

Attribute	Description
Walkdown scope and procedures	Comprehensive seismic walkdowns were performed and documented as part of the EPRI NP-6041-SL [31] screening process. All equipment items credited in the plant response model were included in the walkdown scope. Other than a small number of inaccessible items, all equipment was inspected by the SRT. For the walkdowns, emphasis was placed on inspection of as-built anchorage and lateral support, investigation for seismic interactions (including seismic-fire and seismic-flood) and checks for seismic vulnerabilities documented in the earthquake experience database. The walkdowns also helped the fragility analysis team become familiar with the plant construction.
Earthquake characterization	The applied earthquake was based on site specific GMRS that were produced by the recent probabilistic seismic hazard analysis [35].
Building seismic response analysis	New ISRS were produced for the S-PRA project using the site-specific earthquake motion. New building models were created for this purpose including detailed 3-D finite element models for critical buildings.
Treatment of screened-out equipment	Potential failure of screened-out equipment was addressed through development of surrogate elements. Fragility data were derived from the screening levels and assigned to the surrogate elements.
Fragility analysis methods	The methods in EPRI reports NP-6041-SL [31], TR-1019200 [38] and TR-103959 [39] were used for calculation of seismic fragility parameters. The level of analysis effort applied for an SSC item was tied to the best understanding of its importance to the plant seismic response, and feedback from the plant response model was used to sharpen the focus on the most important items. Fragility calculations were performed in stages to make use of the plant response feedback. Seismic analyses were initially performed using the CDFM method of EPRI NP-6041-SL [31]. Refined CDFM analysis were then performed for dominant contributors to risk and if they remained dominant to risk, another refinement using separation of variables (SOV) method was performed for realistic variabilities.
Documentation	Supporting documentation was created to detail the scope, methods and results of fragility analysis, to verify quality, allow revisions and upgrades, and support regulatory and peer review.

4.4.1 SSC Screening Approach

4.4.1.1 Overview

The methods in EPRI reports NP-6041-SL [31], TR-1019200 [38] and TR-103959 [39] were used for calculation of seismic fragility parameters. The seismic fragility of an SSC item is defined as the conditional probability of its failure at a given value of acceleration. The following parameters define the seismic fragility for any specific SSC item:

PGAmed	=	median capacity, stated as peak ground acceleration
βr	=	logarithmic standard deviation, randomness
βu	=	logarithmic standard deviation, uncertainty
βc	=	SRSS(βr, βu) = logarithmic standard deviation, combined

The above parameters are tied to a lognormal probability distribution. *PGAmed* represents the best estimate of the seismic capacity (50% probability of failure). The β parameters address the variability of the estimate. Parameter βr (randomness) accounts for sources of variability that cannot be reduced by more detailed studies or more data.

In general, a fragility is associated with the failure to perform an assigned function. Non-performance may be a consequence of structural failure, of malfunction of an electro-mechanical component, or of some other type of physical change. The governing failure mode is identified in the final listing of fragility parameters.

Parameter *PGAmed* corresponds to earthquake severity and is defined in terms of horizontal PGA at the control point. The fragility analysis accounts for propagation of earthquake ground motion to the SSC item location and the resulting dynamic response of the item and its supporting structure.

4.4.1.2 Initial Analysis

Seismic analyses were initially performed using the CDFM method of EPRI NP-6041-SL [31]. Each analysis produced a HCLPF capacity for the SSC item. Nominally, the HCLPF is the capacity at which there is 95% confidence of less than 5% probability of failure. Fragility parameters were then produced using the scaling approach of EPRI TR-1019200 [38] Section 3.4. This is equivalent to the Hybrid Method discussed in EPRI 1025287 [28] Section 6.4.1. The median capacity was estimated from the HCLPF value using the following equation:

$$PGAmed = (PGAc) \bullet e^{2.325(\beta c)}$$

PGAc = HCLPF capacity, stated as peak ground acceleration

To produce the initial *PGAmed* for SSC, the HCLPF value was calculated and the corresponding βc value is estimated based on S-PRA experience. Per EPRI 1025287 [28] Table 6-2, $\beta c = 0.45$ was typically applied for an item subject to in-structure demand and $\beta c = 0.35$ was typically applied for an item subject to demand at the ground level. In some cases, an item-specific βc was applied. Unless noted otherwise, the randomness component of βr was set to 0.24 per EPRI 1025287 [28] Table 6-2.

Application of the CDFM to screened-in SSC as a first step provides these benefits:

1. The method and criteria are straightforward and not overly reliant on analyst judgment.
2. Sorting the critical SSC by risk-significance is more dependable when seismic fragilities are based on a common method.
3. It is practical to develop item-specific fragility parameters for a large population or equipment (especially when existing plant seismic calculation methods are similar to CDFM).

The CDFM analysis criteria followed are based on EPRI NP-6041-SL [31] Table 2-5 but include the recommended updates of TR-1019200 [38] Appendix A.

4.4.2 SSC Fragility Analysis Methodology

The HCLPF values from the initial analysis were supplied to the plant response model team for preliminary analysis of seismic core damage frequency (SCDF) and large early release frequency (SLERF). Based on that initial data, dominant contributors were identified, and more rigorous methods of analysis were applied for this subset of screened-in SSC. Under this approach, an existing CDFM analysis was used as a reference and a more refined CDFM analysis was performed. If the item remained significant to risk, the SOV approach was used as a more refined analysis method per EPRI TR-103959 [39] and more realistic variabilities were determined. In general, the refined analysis is expected to produce a more accurate median capacity estimate and more accurate log standard deviations.

4.4.3 SSC Fragility Analysis Results and Insights

Section 5 reports the fragilities of the risk important contributors to the SCDF and SLERF. Detailed (separation of variables) calculations have been performed for selected highest risk-significant SSC, as well as for other selected components.

4.4.4 SSC Fragility Analysis Technical Adequacy

The CEC S-PRA SSC Fragility Analysis was subjected to an independent peer review against the pertinent requirements in the PRA Standard [32].

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

5.0 Plant Seismic Logic Model

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

The seismic plant logic model includes combinations of structural, equipment, and human failures that could give rise to significant core damage and large early release sequences. Quantification of this model yields total SCDF and SLERF, including contribution of both seismic-induced and non-seismically induced unavailabilities, and the identification of important seismic risk contributors. The quantification process also includes an evaluation of uncertainty, which provides perspective on how modeling and parametric sources of uncertainty affect the S-PRA insights.

5.1 Development of the S-PRA Plant Seismic Logic Model

The CEC seismic logic model was developed from the internal events at-power PRA model of record. The internal events model was adapted in accordance with the EPRI Seismic PRA Implementation Guide [27] and ASME/ANS PRA Standard [32] requirements. This process included adding seismic fragility events to the logic model, eliminating portions of the logic model irrelevant to seismic risk (e.g., recovery of offsite power), and adjusting the human reliability analysis to account for response during and following an earthquake. The final seismic logic model is a large fault tree with single top events for SCDF and SLERF. The following summarize each of the major changes made to the internal events logic model during S-PRA Model development.

5.1.1 General Approach

Seismic-induced initiating events that could give rise to significant accident sequences were first identified, including seismic-unique initiating events such as building failure. Next, SSCs whose seismic failure could cause an initiating event, or degrade plant response to an initiating event, were identified and consolidated into an SEL. Initially conservative fragility groups and lognormal fragility parameter estimates were identified for each SEL item.

The seismic hazard was discretized into ten (10) intervals, each with a representative ground motion level and occurrence frequency. Fragility events representing seismic failure of individual components, or groups of components were developed for each ground motion interval. These fragility events were inserted into the fault tree using internal events basic events as targets. Human failure events relevant to seismic sequences were quantified using a screening process accounting for earthquake impact on performance shaping factors, and the resulting seismic human error probabilities were incorporated into the S-PRA quantification process. The resulting S-PRA model is capable of quantifying at-power seismic-induced CDF and LERF, including the contributions of both seismic-induced and non-seismic hardware failures.

5.1.2 Initiating Events and Accident Sequences

The initial step of the S-PRA was to systematically identify earthquake-caused initiating events that have the potential to give rise to significant accident sequences. In the initiating event identification process, a hierarchy was developed to ensure earthquakes exceeding an OBE are modeled, and that their frequency is apportioned to an appropriate induced initiating event. The CEC S-PRA includes the following seismic-induced initiating events:

- Direct to Core Damage and Large Early Release
- Large LOCA
- Intermediate LOCA
- Small LOCA
- Very Small LOCA

- Main Streamline Break Outside of Containment
- Loss of all Component Cooling Water
- Loss of Vital DC Bus
- Spurious Safety Injection Signal Due to Relay Chatter
- LOOP and SBO
- Loss of All Service Water

The direct to core damage and large early release initiating events include seismic-unique failures such as building collapse. The potential for seismic-induced very small LOCA is modeled following non-LOCA initiating events. The initiating event and mitigating systems impact of seismic-induced fires and floods is also included quantitatively in the S-PRA.

While the CEC S-PRA uses a largely unmodified version of the internal events accident sequence and system modeling, some changes were required to reflect potentially risk-significant seismic sequences that were not already included in the base internal events model. The significant modifications are listed below:

- Added logic to reflect seismic failures that lead directly to core damage and large early release
- Added logic to reflect seismic-induced very small loss of coolant accident following non-LOCA initiating events.
- Disabled credit for recovery of offsite power.
- Update the mutually exclusive logic
- Modification of the recovery rule file to apply the seismic human reliability analysis
- Used FRANX to create and insert logic reflecting seismic-induced initiating events and mitigating equipment failures.

5.1.3 Modeling of Correlated Components

Fully correlated components were assigned to correlated component groups so that all components in the group fail at the same time with the same probability based on the seismic magnitude for each hazard bin. The model assumes fully correlated response of same or very similar equipment in the same structure, elevation, and orientation. Correlated component groups were developed consistent with the above mentioned criteria and based on insights from component walkdowns.

5.1.4 Modeling of Human Actions

The CEC Seismic HRA consists of the following tasks:

- Operator action identification and definition
- Feasibility assessment
- Screening quantification
- Detailed quantification
- Model integration

Operator actions to be modeled by the S-PRA were identified and defined using the guidance of EPRI 3002008093 [43] Chapter 3, consistent with supporting requirement SPR-D1 of the PRA Standard [32].

- All operator actions modeled by the IE PRA are identified.

- Any operator actions with a screening human error probability (HEP) applied from the internal events HRA (i.e., not detailed analysis provided) are not credited in the S-PRA.
- Pre-initiator HFEs are independent of the initiating event, and therefore there is no seismic impact to these actions.
- Following initial quantification, a review is performed to identify (if applicable) seismically risk-relevant operator actions not already modeled in the IE PRA. Examples include recovery actions for mitigation equipment impacted by seismic-induced relay chatter.

The feasibility of each identified operator action is assessed using the guidance of EPRI 3002008093 [43] Section 4.2. The purpose of the feasibility assessment is to determine if successful completion of each operator action is even possible in the event of an earthquake. All actions determined to be infeasible must either not be modeled by the S-PRA or have their associated HEPs set to 1.0. The feasibility assessment considers seismic impact on the following performance shaping factors: time, manpower, cues, procedures and training, accessible location and environmental factors, and equipment accessibility, availability, and operability.

All operator actions carried forward from the IE PRA have been determined feasible, in the context of internal initiating events. The conclusion was that the internal events feasibility assessment of each action remains valid in the context of seismic events, and any exceptions are addressed by the screening quantification process.

In the course of the HRA analysis, a number of HRA events were identified and refined. Significant HRA events are identified during the review of the quantification results. Depending on the risk significant of the actions, an iterative process is used to refine and update the HRA events and to assess their impact on quantification results. The following events went through a detailed HRA refinement:

- SH2-EF-XHE-FO-ESWREC – OPS FAILS TO MANUALLY START AND ALIGN ESW SYSTEM BEFORE RX TRIP
- SH2-EF-XHE-FO-MANESW – OPERATOR FAILS TO MANUALLY START AND ALIGN ESW SYSTEM
- SH2-FB-XHE-FO-FANDB – OPERATOR FAILS TO ESTABLISH FEED AND BLEED
- SH2-OP-COG-CCW – OPERATORS FAIL TO DIAGNOSE LOSS OF CCW
- SH2-OP-XHE-FO-AEPS1 – OPERATOR FAILS TO ALIGN AEPS
- SH2-OP-XHE-FO-DEPRES – OPERATOR FAILS TO COOLDOWN AND DEPRESSURIZER RCS
- SH2-OP-XHE-FO-NSAFP – OPERATOR FAILS TO SUPPLY NSAFP WITHIN 45 MINS
- SH2-OP-XHE-FO-RCPTRP – OPERATORS FAIL TO TRIP RCP FROM CONTROL ROOM

Additionally, based on risk insights from the S-PRA quantification process, a seismic specific operator action (NE-XHE-FO-EDG-RLYSET) was defined in accordance with supporting requirement SPR-D2 of the PRA Standard [32] to address recovery from the effects of seismic-induced relay chatter for relay group Relay_0.18DG. Feasibility of this ex-control room action has not yet been assessed via a dedicated walkdown. A sensitivity analysis was performed to capture this impact on model results.

5.1.5 Seismic LERF Model

The transition from Level 1 sequences to Level 2 sequences developed for the IE PRA is considered valid for the S-PRA. A specific initiator evolving into a release scenario is generated by a seismic event does not change the phenomenology associated with containment failure and individual components associated with containment isolation function. The components associated with containment failure are captured in the SEL and are included within the scope of the fragility analysis and the subsequent PRA modeling. Operator actions involved in the Level 2 analysis are similarly captured in the human reliability analysis.

Currently, all containment penetrations are treated as correlated and are assigned to one fragility group (SF-RB-PEN). Failure of containment penetrations is modeled separately from Reactor Building collapse via the fragility associated with Reactor Building equipment hatches.

5.2 S-PRA Plant Seismic Logic Model Technical Adequacy

The CEC S-PRA seismic plant response methodology and analysis were subjected to an independent peer review against the pertinent requirements in the ASME/ANS PRA standard [13].

The S-PRA was peer reviewed [14] relative to Capability Category II for the full set of requirements in the Standard. After completion of the subsequent independent assessment [15], the full set of supporting requirements was met with the exception of SPR-B2 and SPR-E6 which were not reviewed by the independent assessment team. The Seismic plant response analysis was determined to be acceptable for use in the S-PRA. The Peer review assessment, and subsequent disposition of peer review Facts and Observations (F&O) through an independent assessment, is further described in Appendix A and References [14] and [15].

5.3 Seismic Risk Quantification

S-PRA quantification involves assembling the results of the seismic hazard analysis, fragility analysis, and the seismic accident sequence model to estimate the frequencies of core damage and large early release. The risk quantification considers both seismic failures and non-seismic failures, and the applicable operator actions. This section describes the S-PRA quantification methodology and important modeling assumptions.

5.3.1 S-PRA Quantification Methodology

The Callaway S-PRA is a large fault tree with separate top events for SCDF and SLERF. The FRANX software is used to create an integrated one-top PRA model. The fault tree is quantified using FTREX through the PRAQuant graphical user interface (GUI), results are post-processed using ACUBE, and seismic human error probabilities are applied via cutset post-processing with QRecover. The approach used by the FRANX tool is a scenario-based approach which divides the seismic hazard curve into discrete seismic magnitude intervals. PRAQuant is used to quantify each scenario (i.e., %G01, %G02, and finishing with the final hazard interval selected). Each seismic interval is assigned a representative ground motion magnitude and corresponding occurrence frequency. The seismic interval frequencies become the frequency (of occurrence) for the seismic initiators, and the representative magnitude at each interval is used as input to the fragility calculations for the fragility events modeled in the S-PRA.

Seismic HRA dependencies for CDF and LERF were assessed at each seismic HRA bin and the impact of dependent combinations is explicitly included in the S-PRA model quantification and results.

The S-PRA quantification generates a wealth of data, which generally require post-processing to extract meaningful insights. At a minimum, the Callaway S-PRA results are presented in the following forms:

- Total seismic CDF and LERF
- Fractional CDF and LERF contributions of each ground motion interval
- CCDP and CLERP for each ground motion interval
- Occurrence frequency for each ground motion interval
- Ground motion at which the estimated likelihood of core damage and release are 1.0 (plant-level fragility)

- Documented cutset review, including sampling of non-significant cutsets
- Uncertainty analysis for CDF and LERF
- Dedicated sensitivity studies to quantitatively evaluate key assumptions and sources of model uncertainty

5.3.2 S-PRA Model and Quantification Assumptions

Significant assumptions and sources of uncertainty for the CEC S-PRA are summarized as follows:

Hazard Uncertainty

The hazard intervals used for quantification are "tailored" for the current risk profile as an iterative approach was used to determine the number and width of intervals which would result in an increasing seismic CCDP/CLERP percent by hazard interval (up to the PAF) and would also result in a seismic percent CDF/LERF by hazard interval distribution skewed around the plant HCLPF value. A study on hazard intervals was completed and demonstrates convergence based on varying the hazard intervals. Additionally, the epistemic uncertainties identified through the SSHAC process in the hazard are effectively covered via the PSHA in the distribution that describes the entire family of hazard curves. The seismic hazard (exceedance frequency over a range of credible PGA) includes both mean and fractile curves. When FRANX discretizes the hazard and injects seismic initiators into the .rr database, it assigns distribution parameters to characterize the uncertainty of each hazard interval. While the hazard uncertainty is significant, it is quantitatively included in the uncertainty quantification.

Fragilities

No major assumptions or sources of model uncertainty were identified during the fragility analyses that merit a sensitivity study in the final risk quantification. Similar to the hazard, SSC fragility estimates are also subject to significant uncertainty but are quantitatively included in the uncertainty evaluation as each fragility is represented as a lognormal distribution using its median capacity (Am), uncertainty variable (Beta U, which is the uncertainty (state-of-knowledge) of what the true median capacity is), randomness variable (Beta R, which represents how the failure probability of SSC changes based on different seismic intensities) and composite uncertainty variable (Beta C).

Model Development

Significant assumptions and source of model uncertainty identified during the development of the S-PRA model are characterized for their impact on the S-PRA results below:

1. Correlation – The CEC S-PRA is performed under the generic assumption of full correlation of seismically-induced failures of similar components. This is a recognized conservative assumption that is commonly used in the developed of S-PRAs and is consistent with industry practice. While research activities are performed in the industry to investigate alternative modeling of potential correlations between seismic failures, these are judged to not be ready yet to be implemented in the CEC S-PRA. Correlation grouping was initially determined based on equipment type and building location. For dominant contributors, correlation grouping was iteratively refined where appropriate based on additional considerations consistent with SFR-A2 (e.g., building elevation, anchorage similarity, orientation, etc). For significant fragility groups (FV > 2% for CDF or LERF), as identified by the fragility ranking sensitivities, the basis for the correlation assumptions is reviewed.
2. LLOCA – Seismically-induced failure of the pressurizer supports are assumed to lead to unrestrained motion of the pressurizer and subsequent failure of the pressurizer surge line. No other concurrent failures resulting from the unrestrained motion of the pressurizer are assumed. This is a recognized conservative assumption used for modeling purposes as it is understood that the motion of the pressurizer can be impeded by other structures in the containment and that pipe flexibility can probably accommodate a significant motion from the pressurizer before generating

a full break of the line. This modeling assumption is expected to artificially increase the importance of seismically induced LLOCA. To reduce the modeling uncertainty associated with this modeling assumption a more refined assessment of the actual impact of the pressurizer motion once the supports are failed could be performed, although to deterministically or probabilistically assess the impact of the unrestrained motion of the pressurizer is beyond current practice in the S-PRA. Additionally, further assessment is not warranted at this time based on LLOCA due to pressurizer failure not being a significant contributor to the model results.

3. MLOCA – The generic MLOCA fragility estimates available in NUREG/CR-4840 [21] are considered applicable to the CEC S-PRA. These estimates are recognized as dated and generic (i.e., not plant or design specific). A plant-specific fragility analysis for pipe break is an alternative approach which is not performed for the CEC S-PRA on the basis that seismically-induced Intermediate LOCAs are not expected to be significant or lead contributors to the CEC seismic risk profile. Note that the EPRI S-PRA implementation guide provides an alternative source for generic MLOCA fragility parameters (with a higher median capacity) which may be a potential source of margin. Additionally, further assessment is not warranted at this time based on MLOCA not being a significant contributor to the model results.

4. VSLOCA – RCS leakage within the very small LOCA range is conservatively assumed to occur for all seismic events above the OBE, regardless of the presence of the CVCS system. The intent of this modeling is based on ASME/ANS-RA-Sb-2013 [2] supporting requirement SPR-B8 which imposes this approach to ensure that some form of RCS makeup is demanded for all seismic sequences. Note that the Code Case [13] essentially subsumes this requirement into SPR-A1. The approach is conservative, especially at low ground motion levels, given that much of the earthquake operating experience did not involve loss of coolant.

5. LOCA Location – The same LOCA split factions used in the internal events PRA were assumed to be also applicable to the S-PRA. Such split fractions assign equal probability of a LOCA to happen to any RCS loop. A plant-specific investigation to assess whether any of the RCS loops has an appreciably different seismic capacity is an alternative approach to validate this assumption but it was not performed for the CEC S-PRA as this goes beyond current practice in S-PRA and is judged not to be a significant limitation in the analysis. Also, note that limiting a seismically-induced LOCA on a specific RCS loop is in some extent inconsistent with the generic item #1 above on full correlation between seismic-induced failures. This is a recognized inconsistency that is driven by the practical limitations in developing detailed fragility estimates for each individual RCS pipe section.

6. Building Failure – A seismic event that is strong enough to induce catastrophic failure of a Seismic Category I building is also assumed to generate a loss of offsite power event that is not recoverable within the S-PRA mission time. This is considered a realistic assumption.

7. Building Failure – Catastrophic events such as major structural collapses are assumed to lead directly to a core damage scenario and a large early release. This assumption is considered conservative because of the relatively conservative approach normally used to generate the fragility of major structural collapses. There are no sensitives that are explicitly designed to monitor the epistemic uncertainty associated with this assumption but the importance of the scenario leading directly to core damage has been monitored during the development of the CEC S-PRA and the fragilities associated with these scenarios have been refined to a point where it is considered cost effective (i.e., the effort invested in the fragility analysis for these scenarios matches the relative importance of the events).

8. Rule-of-the-box – A number of basic events in the CAFTA internal events model file refer to MSIVs, MFIVs, MFRVs, and MFR bypass valve actuators which are expected to be rule-of-the-

box with the MSIVs, MFIVs, MFRVs and MFR bypass valves themselves. A dedicated entry in the SEL is therefore not provided for these subcomponents.

9. Rule-of-the-box – A number of basic events in the CAFTA file refer to limit switches associated with MOVs. The limit switches do not normally have a dedicated tag number and they are expected to be rule-of-the-box within the MOVs themselves. A dedicated entry in the SEL is therefore not provided for these subcomponents.

10. Mission Time – The CEC S-PRA uses the same overall mission time used by the IE PRA (i.e., 24 hours). This is a critical assumption that allows the use of the same set of event trees originally developed for the IE PRA. This is based on the inherent assumption that offsite power is recovered within 24 hours from the event. The CEC S-PRA does not credit the LOOP recovery that is credited in the IE PRA, but the overall mission time of the IE PRA inherently assumes that LOOP is fully recovered within 24 hours. While this is a reasonable assumption for seismic events of lower magnitude (considering that event for the Mineral 5.8 Earthquake, which was relatively severe, offsite power was recovered at the station in the evening of the day of the event, thus roughly 10 hours after the event), it may be less realistic for a seismic event of very high magnitude (at Fukushima, where major infrastructure damage was also caused by the tsunami, offsite power was restored 11 days after the event). One can therefore conclude that the epistemic uncertainty associated with this assumption is higher for seismic-induced events that are high-consequence events (e.g., Large LOCA) and is of less importance for lower magnitude and lower consequence events. Maintaining the same overall mission time is a practical approach that is consistently used in the current practice and an alternative approach of re-evaluating the mission time for each individual sequence and component/system may require extensive logic change and generation of new success criteria thermal-hydraulic simulations. Extending the mission time beyond 24 hours would also introduce significant departure from the as-operated condition of the plant because the longer the time from event and the higher the possibility that damaged equipment may be repaired or that non-proceduralized activities may be performed. Repairs and non-proceduralized are not normally included in the PRA. This would therefore introduce significant uncertainties in the fidelity and realism of the seismic risk profile. The assumption that the same mission time is applicable, is supplemented by extending the definition of the SEL; for example including equipment needed for refueling of the DG (i.e., the S-PRA does not solely rely on the DG day tank). The CEC FLEX Integrated Plan describes the site program for coping with earthquakes requiring mitigation beyond 24 hours. While, conservatively, the internal events and S-PRAs do not credit FLEX implementation, the existence of the systematic and regulated FLEX process increases confidence that the plant can mitigate longer-term accident conditions, and this is especially true for the low to mid-level ground motion events for which the S-PRA does not already consider to have resulted in core damage and release.

11. Seismic Initiating Event Tree – Development of the seismic initiating event tree (SIET) involves a ranking of seismic initiating events from greatest to least in terms of potential risk significance, with the purpose of ensuring each ground motion level is assigned to the most challenging initiating event that could be credibly induced by that ground motion level. This ranking involves judgment and is a source of epistemic uncertainty. For example, while large LOCA occurs earlier in the SIET than small LOCA, it is not immediately clear that seismic-induced large LOCA is more challenging that a small LOCA in terms of risk significance. The actual risk significant of these induced initiating events depends on the mitigating systems they demand and the fragilities for those systems. One consequence of the SIET initiating events being disordered is that a larger fraction of the seismic frequency may be inappropriately apportioned to a plant impact of lesser consequence. This source of epistemic uncertainty may be assessed via sensitivity studies using the S-PRA model.

12. Accident Sequence Models, Success Criteria, and System Models – The CEC internal events model includes internal events initiator basic events in the accident sequence modeling and also in some cases within system models to address conditional system dependencies and conditional dependencies for I&C signals. Seismic induced initiating events are included into the model based on their corresponding internal event initiating event. This introduces a source of uncertainty since the internal event logic may be limited in addressing the uniqueness of a seismic induced event that could satisfy multiple internal events at the same time (e.g., Loss of offsite power and very small LOCA at the same time).
13. HRA – The selection of breaking points for the seismic HRA plant damage bins are left to interpretation and judgement of the analyst. The definition of Plant Damage Bin SH3 describes the boundary between Plant Damage Bins SH3 and SH4 as a source of uncertainty. As it is observed that the majority of the fragilities with median capacity between 0.4g and 0.6g are actually concentrated in the 0.5g to 0.6g range, a sensitivity can be performed in the quantification notebook to evaluate the epistemic uncertainty associated with the boundary between SH3 and SH4 pushing the threshold as low as 0.5g PGA. It is assumed that the "long time frame" branches in the EPRI seismic HRA decision tree can be used for the seismic version of operator action OP-XHE-FO-ECLRS2. This is based on the fact that this is an action that takes place more than 3 hours after the period of strong shaking and also because of the fact that the internal events HRA indicates a potential longer time frame for recovery, which is currently not credited in the internal events model. This represents a partial refinement of this operator action (i.e., as opposed to manipulating the HRA calculator file).
14. Level 1 and 2 Linkage – The interface between Level 1 and Level 2 is assumed applicable to the S-PRA without changes. This assumption is supported by the fact that there are no late releases with containment failure timing close to the 12 hours timing used to define LERF in the internal events Level 2 analysis. It is nevertheless recognized that, because the potential LERF re-categorization timing is essentially dependent on offsite conditions it has inherent uncertainties.
15. Spatial Interaction – The spatial interaction between the block walls at elevation 2016' of the Control Building are not explicitly modeled as impacting the NK equipment in that area. This is based on the significant difference between the functional HCLPF of the modeled components and the HCLPF of the block walls, which is similar in turn to the direct to CD and direct to LERF fragility associated with soil failure. It is therefore judged that no significant completeness uncertainties are added by this model approach while on the other hand the model is maintained less heavy.

Quantification

The S-PRA quantification method itself is a recognized limitation. The CEC S-PRA model discretizes the hazard into 10 intervals, calculates SSC failure probabilities at each interval, and CCDP/CLERP at each interval using the plant response model. In addition, the minimal cutset upper bound approximation used by FTREX to calculate CCDP/CLERP from cutsets can be over-conservative where the conditional probabilities exceed 0.1, as is the case for the dominant sequences. To minimize, but not eliminate, this conservatism, ACUBE is applied to all CDF and LERF cutsets at each hazard interval. Seismic HRA dependency is explicitly assessed and modeled for CDF and LERF.

5.4 SCDF Results

This section presents the base SCDF results, a list of SSCs that are significant contributors, including risk importance measures, and a discussion of significant cutsets

The total point estimate SCDF is 5.59E-05 per reactor year (note that ACUBE was used to post-process the CAFTA cutsets, and all cutsets are post-processed through ACUBE). Table 5-1 provides the PGA, earthquake occurrence frequency, truncation limit, ACUBE CCDP, ACUBE CDF, and contribution for each ground motion interval. Note that the reported SCDF includes a 0.90 plant availability factor. The dominant PGA range is between 0.3 and 0.4g. Note that the CDF percentages displayed in Table 5-1 may be overestimates due to not being able to post-process all cutsets with ACUBE. Although it appears that the dominant range is between 0.5g to 0.55g, PGA range 0.3g to 0.4g remains dominant.

Table 5-1: Contribution to SCDF by Acceleration Level

Interval	PGA (g)	Frequency (/yr)	Truncation	ACUBE CCDP	ACUBE CDF	% CDF
%G01: 0.1g to 0.2g	0.14	8.02E-04	1.00E-11	3.68E-04	2.95E-07	1%
%G02: 0.2g to 0.3g	0.24	1.80E-04	1.00E-13	1.41E-02	2.53E-06	5%
%G03: 0.3g to 0.4g	0.35	6.48E-05	1.00E-12	1.14E-01	7.38E-06	13%
%G04: 0.4g to 0.45g	0.42	1.67E-05	1.00E-12	3.94E-01	6.58E-06	12%
%G05: 0.45g to 0.5g	0.47	1.16E-05	1.00E-11	6.07E-01	7.04E-06	13%
%G06: 0.5g to 0.55g	0.52	8.13E-06	1.00E-10	9.50E-01	7.72E-06	14%
%G07: 0.55g to 0.6g	0.57	5.87E-06	1.00E-10	9.97E-01	5.85E-06	10%
%G08: 0.6g to 0.7g	0.65	7.60E-06	1.00E-08	8.66E-01	6.58E-06	12%
%G09: 0.7g to 0.8g	0.75	4.45E-06	1.00E-08	8.88E-01	3.95E-06	7%
%G10: > 0.8g	0.88	8.85E-06	1.00E-07	8.98E-01	7.95E-06	14%
				Total	**5.59E-05**	

Table 5-2 identifies a sample of 10 significant CDF cutsets. Note that post-processing of cutsets with ACUBE does not output new cutsets but only updated frequencies. Therefore, the cutset review is performed on the CAFTA cutset file.

Table 5-2: Sample CDF Cutsets

ID	Cutset(1)	Description
1	FL-KCECCS-FAIL I FL-S-TC I FL-TC-S10 I G02 I NOLOSPCONDITION-C-G02 I PAF I SF-NB01C%G02 I SF-NB01CFF I SH2-EG-XHE-FO-STBTRN I SH2-OP-XHE-FO-LETDWN I SH2-OP-XHE-FO-RCPTRP-CCW I SH2-COMBINATION_35	A seismic-induced loss of component cooling water occurs due to the seismic-induced failure of NB01 combined with the operators failing to align the standby CCW train prior to reactor trip. The reactor is successfully tripped, there is no consequential LOOP, and AFW is available. Operators fail to isolate letdown following the loss of CCW which results in loss of the normal charging pump. Operators fail to trip the RCPs on a loss of CCW and Fire Protection is not aligned to ECCS which results in core damage.

Table 5-2: Sample CDF Cutsets		
ID	Cutset[1]	Description
2	FL-DEPLETED I FL-DGA-FTR I FL-DGB-FTR I FL-ESWA-FAILS I FL-ESWB-FAILS I FLEXACTODCFAIL I FL-S-TSW I FL-SW-FAILS I FL-TSW-S09 I G02 I NOLOSPCONDITION-C-G02 I PAF I SF-FR-YDXFRC%G02 I SF-FR-YDXFRCFF I SF-IE-SWC%G02 I SF-IE-SWCFF I SH2-EF-XHE-FO-MANESW I SH2-OP-XHE-FO-AEPS1 I SH2-OP-XHE-FO-RCPTRP-CCW I XFR_IGNITES I SH2-COMBINATION_22	A seismic-induced loss of service water occurs in combination with a seismic-induced failure of the yard transformer housing resulting in a total loss of service water. The reactor is successfully tripped, there is no consequential LOOP, and AFW is available. The NCP fails to provide seal injection due to the loss of the yard transformer housing. The operators fail to trip the RCPs. Fire Protection is available to ECCS, but the CCPs/SIPs fail to inject due to power failure (including failure of the FLEX strategy, See Note 2) which results in core damage.
3	FL-S-TSW I FL-SW-FAILS I FL-TSW-S08 I G02 I NOLOSPCONDITION-C-G02 I PAF I SF-IE-SWC%G02 I SF-IE-SWCFF I SF-RL0XXC%G02 I SF-RL0XXCFF	A seismic-induced loss of service water occurs in combination with a seismic-induced failure of the main control room boards which results in a loss of all mitigation capabilities and leads to core damage.
4	FL-DGA-FTS I FL-DGB-FTS I FLEXAFWFAIL I FL-SHLD-S I FL-S-T1 I FL-T1-S08 I FL-T1SAE-S05 I FL-TDP-FTR I G02 I NN-INV-TM-NN17 I PAF I SF-FR-YDXFRC%G02 I SF-FR-YDXFRCFF I SF-IE-T1C%G02 I SF-IE-T1CFF I SF-NN1X-1C%G02 I SF-NN1X-1CFF I SH2-AL-XHE-FO-SBOSGL I SH2-NE-XHE-FO-EDG I SH2-OP-XHE-FO-ACRECV I XFR_IGNITES I SH2-COMBINATION_1987	A seismic-induced loss of offsite power occurs. The reactor is successfully tripped, but the diesels fail due to a combination of seismic-induced failure of the 120 VAC Distribution Panel and the operators failing to start and align the other EDG. AEPS is available, any open pressurizer PORV recloses, and the shutdown seal is successful; however, AFW and the CCPs fail due to combination of the seismic-induced failure of the yard transformers, operators failing to maintain SG level after a complex event, failure of the FLEX strategy (see Note 2), and operators failing to recover from a loss of offsite power which results in core damage.

Table 5-2: Sample CDF Cutsets		
ID	**Cutset[(1)]**	**Description**
5	FL-DGA-FTS I FL-DGB-FTS I FLEXAFWFAIL I FL-SHLD-S I FL-S-T1 I FL-T1-S08 I FL-T1SAE-S10 I FL-T1S-S17 I FL-TDP-FTR I FTREC I G02 I NN-INV-TM-NN17 I PAF I SF-FR-YDXFRC%G02 I SF-FR-YDXFRCFF I SF-IE-T1C%G02 I SF-IE-T1CFF I SF-NN1X-1C%G02 I SF-NN1X-1CFF I SH2-AL-XHE-FO-SBOSGL I SH2-NE-XHE-FO-EDG I XFR_IGNITES I SH2-COMBINATION_99	A seismic-induced loss of offsite power occurs. The reactor is successfully tripped, but the diesels fail due to a combination of seismic-induced failure of the 120 VAC Distribution Panel and the operators failing to start and align the other EDG. AEPS fails due to the seismic-induced failure of the yard transformer which results in a SBO. The shutdown seal successfully actuates, but AFW fails due to a combination of the seismic-induced failure of the yard transformer, failure of the AFW FLEX strategy (see Note 2), and operators failing to maintain SG Level after a complex event which results in core damage.
6	FL-DEP I FL-DGA-FTS I FL-DGB-FTS I FL-SHLD-S I FL-S-T1 I FL-T1-S08 I FL-T1SAE-S10 I FL-T1S-S11 I FTREC I G02 I PAF I SF-IE-T1C%G02 I SF-IE-T1CFF I SF-NN0XC%G02 I SF-NN0XCFF I SH2-NE-XHE-FO-EDG I SH2-OP-XHE-FO-AEPS1 I SH2-OP-XHE-FO-TDPMNL I SH2-COMBINATION_204	A seismic-induced loss of offsite power occurs. The reactor is successfully tripped, but the diesels fail due to a combination of seismic-induced failure of the 120 VAC Distribution panels and the operators failing to start and align the other EDG. Operators fail to align AEPS which results in an SBO. The shutdown seal successfully actuates, AFW is available, and any stuck open PORV/SRV recloses. Cooldown and depressurization fail due to the seismic-induced failure of the 120 VAC Distribution panels combined with the loss of the steam dumps. Offsite power is not restored (as it is not credited for a seismic event), and the operators fail to manual operate the TDAFP which results in core damage.
7	FL-DGA-FTS I FL-DGB-FTS I FLEXAFWFAIL I FL-SHLD-S I FL-S-T1 I FL-T1-S08 I FL-T1SAE-S04 I G02 I PAF I SF-IE-T1C%G02 I SF-IE-T1CFF I SF-NN0XC%G02 I SF-NN0XCFF I SH2-AL-XHE-FO-MDAFP I SH2-AL-XHE-FO-TDAFP I SH2-FB-XHE-FO-PORV1S I SH2-NE-XHE-FO-EDG I SH2-OP-XHE-FO-RFLN2A I SH2-COMBINATION_440	A seismic-induced loss of offsite power occurs. The reactor is successfully tripped, but the diesels fail due to a combination of seismic-induced failure of the 120 VAC Distribution panels and the operators failing to start and align the other EDG. AEPS is available, any open pressurizer PORV recloses, and the shutdown seal successfully actuates. AFW fails due a combination of the loss of 120 VAC Distribution panels, the operators failing to start the MDAFP, and the failure of the FLEX AFW strategy (see Note 2). The CCPs inject, but Feed and bleed fails due to the operators failing to establish an RCS bleed path with the PORVs which results in core damage.

Table 5-2: Sample CDF Cutsets		
ID	Cutset(1)	Description
8	FL-KCECCS-FAIL I FL-S-TC I FL-TC-S10 I G03 I NOLOSPCONDITION-C-G03 I PAF I SF-NB01C%G03 I SF-NB01CFF I SH2-EG-XHE-FO-STBTRN I SH2-OP-XHE-FO-LETDWN I SH2-OP-XHE-FO-RCPTRP-CCW I SH2-COMBINATION_35	A seismic-induced loss of component cooling water occurs due to the seismic-induced failure of NB01 combined with the operators failing to align the standby CCW train prior to reactor trip. The reactor is successfully tripped, there is no consequential LOOP, and AFW is available. Operators fail to isolate letdown following the loss of CCW which results in loss of the normal charging pump. Operators fail to trip the RCPs on a loss of CCW and Fire Protection is not aligned to ECCS which results in core damage
9	FL-DEP I FL-DGA-FTS I FL-DGB-FTS I FL-SHLD-S I FL-S-T1 I FL-T1-S08 I FL-T1SAE-S10 I FL-T1S-S11 I FTREC I G03 I PAF I SF-IE-T1C%G03 I SF-IE-T1CFF I SF-NN0XC%G03 I SF-NN0XCFF I SH2-NE-XHE-FO-EDG I SH2-OP-XHE-FO-AEPS1 I SH2-OP-XHE-FO-TDPMNL I SH2-COMBINATION	A seismic-induced loss of offsite power occurs. The reactor is successfully tripped, but the diesels fail due to a combination of seismic-induced failure of the 120 VAC Distribution panels and the operators failing to start and align the other EDG. Operators fail to align AEPS which results in an SBO. The shutdown seal successfully actuates, AFW is available, and any stuck open PORV/SRV recloses. Cooldown and depressurization fail due to the seismic-induced failure of the 120 VAC Distribution panels combined with the loss of the steam dumps. Offsite power is not restored (as it is not credited for a seismic event), and the operators fail to manual operate the TDAFP which results in core damage.
10	FL-DGA-FTS I FL-DGB-FTS I FLEXAFWFAIL I FL-SHLD-S I FL-S-T1 I FL-T1-S08 I FL-T1SAE-S04 I FL-TDP-FTR I G03 I PAF I SF-IE-T1C%G03 I SF-IE-T1CFF I SF-NN0XC%G03 I SF-NN0XCFF I SH2-AL-XHE-FO-MDAFP I SH2-AL-XHE-FO-SBOSGL I SH2-FB-XHE-FO-PORV1S I SH2-NE-XHE-FO-EDGI SH2-OP-XHE-FO-RFLN2A I SH2-COMBINATION_1105	A seismic-induced loss of offsite power occurs. The reactor is successfully tripped, but the diesels fail due to a combination of seismic-induced failure of the 120 VAC Distribution panels and the operators failing to start and align the other EDG. AEPS is available, any open pressurizer PORV recloses, and the shutdown seal successfully actuates. AFW fails due a combination of the loss of 120 VAC Distribution panels, the operators failing to start the MDAFP, and the failure of the FLEX AFW strategy (see Note 2). The CCPs inject, but Feed and bleed fails due to the operators failing to establish an RCS bleed path with the PORVs which results in core damage.

Table 5-2: Sample CDF Cutsets		
ID	**Cutset(1)**	**Description**
Notes: 1. Events with a suffix "-CFF" are correlation factors (set to 1.0) and are added by FRANX. 2. FLEX events are included in the model to estimate the overall impact on the SPRA model. The FLEX events in the PRA model are currently set to a value of 0.99. A sensitivity study is performed to estimate the overall risk metric reduction from refining these values.		

Table 5-3 summarizes the fragilities with a CDF Fussell-Vesely importance of greater than 2%. The Fussell-Vesely for each fragility was approximated by assuming the component was rugged then recalculating the failure probability at each ground motion interval, imposing those failure probabilities onto the cutsets, recalculating the CDF by ACUBE, and finally calculating the percent reductions of total CDF that the improved capacity affords.

Offsite power (SF-IE-T1) has the greatest importance, which is due to its low assumed capacity, its assumed non-recoverability, and the significant plant impact that losing offsite power creates. Seismic-induced failure of service water (SF-IE-SW) is significant because of its low capacity. Seismic-induced failure of the yard transformer housings (SF-FR-YDXFR) is significant due to its ability to impact power for a large subset of equipment. Note that non-seismic failures generally do not contribute significantly to SCDF.

Table 5-3: SCDF Importance Measures Ranked by Fussell-Vesely

Fragility	Description	F-V	A_m (g)	β_u	β_r	Failure Mode	Method
SF-IE-T1	Seismic-Induced Loss of Offsite Power	0.365	0.3	1.0E-07	0.55	Yard-Centered Loss of Offsite Power	Generic (Conservative Estimate)
SF-IE-SW	Seismic Induced failure of service water (NSCI)	0.061	0.24	0.38	0.24	Loss of Non-Nuclear Safety Equipment	Generic (Conservative Estimate)
SF-FR-YDXFR	Seismic rupture of the yard transformer housings, oil leakage, and subsequent ignition	0.055	0.24	0.38	0.24	Loss of Non-Nuclear Safety Equipment	Generic (Conservative Estimate)
SF-NSCI	Seismic Induced failure of Non-SC-I SSCs	0.055	0.24	0.38	0.24	Loss of Non-Nuclear Safety Equipment	Generic (Conservative Estimate)
SF-NB01	Seismic Induced Failure of the 4.16 KV Switchgear NB01	0.052	0.99	0.52	0.26	Loss of Switchgear NB01	Detailed Analysis
SF-NK02	Seismic Induced Failure of 125 V DC Bus NK02	0.036	0.90	0.38	0.24	Loss of 125V DC Bus NK02	CDFM
Relay_0.23	Relay Fragility Group	0.030	0.58	0.32	0.24	Relay Chatter	CDFM
SF-NG02	Seismic Induced Failure of 480 V Load Center NG02	0.027	0.58	0.32	0.24	Loss of 480 V Load Center NG02	CDFM
SF-NGXC-1	Seismic Induced Failure of the MCC NG05E and NG06E	0.022	0.76	0.32	0.24	Loss of MCCs NG05E and NG06E	CDFM
SF-NG01	Seismic Induced Failure of 480 V Load Center NG01	0.022	0.68	0.32	0.24	Loss of 480 V Load Center NG01	CDFM
SF-NN0X	Seismic Induced Failure of 120 VAC Distribution Panels NN01, NN02, NN03 and NN04	0.020	0.914	0.38	0.24	Loss of 120 VAC Distribution Panels NN01/2/3/4	CDFM Based on Seismic Test Data

Table 5-4 identifies the relative contribution of each initiator to total CDF for those initiators comprising the top 95% of CDF, or individually contributing greater than 1% to total CDF.

Table 5-4: Relative Contribution of Each Initiator to Total SCDF	
Initiator	**Contribution**
Loss of Offsite Power	41.0%
SBO	15.5%
Loss of Seal Cooling	13.8%
Loss of NK02	11.9%
ATWS	4.3%
Direct to Core Damage	3.9%
Loss of Service Water	2.9%
Loss of CCW	1.5%

5.5 SLERF Results

This section presents the base SLERF results, a list of SSCs that are significant contributors, including risk importance measures, and a discussion of significant cutsets.

The total point estimate SLERF is 2.90E-06 per reactor year (note that ACUBE was used to post-process the CAFTA cutsets, and all cutsets are post-processed through ACUBE). Table 5-5 provides the PGA, earthquake occurrence frequency, truncation limit, ACUBE CLERP, ACUBE LERF, and contribution for each ground motion interval. Note that the reported SLERF includes a 0.90 plant availability factor. The largest individual contribution to SLERF is from interval %G05, which represents ground motion between 0.8g to 1.0g.

Table 5-5: Contribution to SLERF by Acceleration Level

Interval	PGA (g)	Frequency (/yr)	Truncation	ACUBE CLERP	ACUBE LERF	% LERF
%G01: 0.1g to 0.2g	0.14	8.02E-04	1.00E-14	1.09E-06	8.73E-10	0%
%G02: 0.2g to 0.5g	0.32	2.73E-04	1.00E-12	1.34E-04	3.65E-08	1%
%G03: 0.5g to 0.6g	0.55	1.40E-05	1.00E-10	4.44E-03	6.21E-08	2%
%G04: 0.6g to 0.8g	0.69	1.21E-05	1.00E-09	2.27E-02	2.75E-07	9%
%G05: 0.8g to 1g	0.89	4.49E-06	1.00E-08	1.10E-01	4.93E-07	17%
%G06: 1g to 1.2g	1.10	1.89E-06	7.00E-08	2.51E-01	4.75E-07	16%
%G07: 1.2g to 1.4g	1.30	1.01E-06	1.00E-07	4.54E-01	4.59E-07	16%
%G08: 1.4g to 1.6g	1.50	5.68E-07	1.00E-07	6.34E-01	3.60E-07	12%
%G09: 1.6g to 2g	1.79	5.29E-07	2.00E-07	8.00E-01	4.23E-07	15%
%G10: > 2g	2.20	3.63E-07	2.00E-07	8.82E-01	3.20E-07	11%
				Total	**2.90E-06**	

Table 5-6 identifies a sample of 10 significant SLERF cutsets. Note that post-processing of cutsets with ACUBE does not output new cutsets but only updated frequencies. Therefore, the cutset review is performed on the CAFTA cutset file.

Table 5-6: Sample SLERF Cutsets

ID	Cutset(1)	Description
1	PAF I SF-SOIL I SF-SOILCFF	A seismic-induced soil failure occurs and leads directly to LERF.

Table 5-6: Sample SLERF Cutsets		
ID	Cutset(1)	Description
2	FL-SHLD-S I FL-S-T1 I FL-T1-S06 I FL-TRCP-S05 I EARLY I NO_BYPASS I NON-SBO I PAF I RELAY_0.18CB I RELAY_0.18CBCFF I SF-AB-SQ07 I SF-AB-SQ07CFF I SF-IE-T1 I SF-IE-T1CFF I SF-RB-PEN I SF-RB-PENCFF I SF-SGLXXX I SF-SGLXXXCFF	A seismic-induced loss of offsite power occurs. The reactor is successfully tripped, 1 of 2 DGs provide power, and any open pressurizer PORV recloses. Seal cooling is lost due to seismic-induced relay chatter from relay group 0.18CB. The operators manually trip the RCPs, and the shutdown seal actuates. AFW is lost due to the seismic-induced failure of the Auxiliary Building MOVs (SQ07). FPS is available to ECCS, but the CCPs fail to inject due to the seismic-induced failure of the SI room coolers which leads to core damage. There is no SBO, containment is not bypassed, containment isolation fails due to failure of the RB penetrations, and there is early containment failure which results in LERF.
3	FL-SHLD-S I FL-S-T1 I FL-T1-S06 I FL-TRCP-S05 I EARLY I NO_BYPASS I NON-SBO I PAF I SF-AB-SQ07 I SF-AB-SQ07CFF I SF-IE-T1 I SF-IE-T1CFF I SF-RB-PEN I SF-RB-PENCFF I SF-RCPSEALHX I SF-RCPSEALHXCFF I SF-SGLXXX I SF-SGLXXXCFF	A seismic-induced loss of offsite power occurs. The reactor is successfully tripped, 1 of 2 DGs provide power, and any open pressurizer PORV recloses. Seal cooling is lost due to seismic-induced failure of the RCP Seal HX. The operators manually trip the RCPs, and the shutdown seal actuates. AFW is lost due to the seismic-induced failure of the Auxiliary Building MOVs (SQ07). FPS is available to ECCS, but the CCPs fail to inject due to the seismic-induced failure of the SI room coolers which leads to core damage. There is no SBO, containment is not bypassed, containment isolation fails due to failure of the RB penetrations, and there is early containment failure which results in LERF.

Table 5-6: Sample SLERF Cutsets		
ID	**Cutset(1)**	**Description**
4	FL-S-T1 I FL-T1-S03 I EARLY I NO_BYPASS I NON-SBO I PAF I RELAY_0.23 I RELAY_0.23CFF I SF-AB-SQ07 I SF-AB-SQ07CFF I SF-IE-T1 I SF-IE-T1CFF I SF-NG01 I SF-NG01CFF I SF-RB-PEN I SF-RB-PENCFF	A seismic-induced loss of offsite power occurs. The reactor successfully trips, 1 of 2 DGs provide power, any open pressurizer PORV recloses, and RCP seal cooling is maintained. AFW fails due to the seismic-induced loss of the Auxiliary Building MOVs (SQ07) and seismic-induced relay chatter from relay group 0.23. The CCPs inject, and Feed and Bleed is successful; however, the CCPs/SIPs fail during recirculation due to the seismic-induced failure of the Auxiliary Building MOVs which results in core damage. There is no SBO, the containment is not bypassed, containment isolation fails due to failure of the RB penetrations, and there is an early containment failure which results in LERF.
5	FL-S-T1 I FL-T1-S03 I EARLY I NO_BYPASS I NON-SBO I PAF I SF-AB-SQ07 I SF-AB-SQ07CFF I SF-IE-T1 I SF-IE-T1CFF I SF-NG01 I SF-NG01CFF I SF-NG02 I SF-NG02CFF I SF-RB-PEN I SF-RB-PENCFF	A seismic-induced loss of offsite power occurs. The reactor successfully trips, 1 of 2 DGs provide power, any open pressurizer PORV recloses, and RCP seal cooling is maintained. AFW fails due to the seismic-induced loss of the Auxiliary Building MOVs (SQ07) and seismic-induced relay chatter from relay group 0.23. The CCPs inject, and Feed and Bleed is successful; however, the CCPs/SIPs fail during recirculation due to the seismic-induced failure of the Auxiliary Building MOVs which results in core damage. There is no SBO, the containment is not bypassed, containment isolation fails due to failure of the RB penetrations, and there is an early containment failure which results in LERF.

Table 5-6: Sample SLERF Cutsets		
ID	**Cutset[(1)]**	**Description**
6	FL-S-T1 I FL-T1-S03 I EARLY I NO_BYPASS I NON-SBO I PAF I SF-AB-SQ07 I SF-AB-SQ07CFF I SF-AB-SQ08 I SF-AB-SQ08CFF I SF-IE-T1 I SF-IE-T1CFF I SF-RB-PEN I SF-RB-PENCFF	A seismic-induced loss of offsite power occurs. The reactor successfully trips, 1 of 2 DGs provide power, any open pressurizer PORV recloses, and RCP seal cooling is maintained. AFW fails due to a combination of the Auxiliary Building MOVs (SQ07 and SQ08). The CCPs inject, and Feed and Bleed is successful; however, the CCPs/SIPs fail during recirculation due to a combination of the Auxiliary Building MOVs (SQ07 and SQ08) failures which results in core damage. There is no SBO, the containment is not bypassed, containment isolation fails due to the failure of the RB penetrations, and there is an early containment failure which results in LERF.
7	FL-SHLD-S I FL-S-T1 I FL-T1-S06 I FL-TRCP-S05 I EARLY I NO_BYPASS I NON-SBO I PAF I SF-AB-SQ07 I SF-AB-SQ07CFF I SF-IE-T1 I SF-IE-T1CFF I SF-RB-PEN I SF-RB-PENCFF I SF-RB-SQ08 I SF-RB-SQ08CFF I SF-SGLXXX I SF-SGLXXXCFF	A seismic-induced loss of offsite power occurs. The reactor is successfully tripped, 1 of 2 DGs provide power, any open pressurizer PORV recloses. RCP seal cooling fails due to the seismic-induced failure of the SI room coolers. Operators manually trip the RCPs, and the shutdown seal successfully actuates. TDAF is unsuccessful due to the combination of seismic-induced failure of the Auxiliary Building MOVs (SQ07 and SQ08). FPS is available to ECCS, but the CCPs/SIPs fail to inject due to the seismic-induced failure of the SI room coolers which leads to core damage. There is no SBO, the containment is not bypassed, containment isolation fails due to seismic-induced failure of the RB penetrations, and the containment fails early results in LERF.

Table 5-6: Sample SLERF Cutsets		
ID	**Cutset(1)**	**Description**
8	FL-S-T1 I FL-T1-S03 I EARLY I NO_BYPASS I NON-SBO I PAF I RELAY_0.23 I RELAY_0.23CFF I RELAY_0.27 I RELAY_0.27CFF I SF-AB-SQ07 I SF-AB-SQ07CFF I SF-IE-T1 I SF-IE-T1CFF I SF-RB-PEN I SF-RB-PENCFF	A seismic-induced loss of offsite power occurs. The reactor is successfully tripped, 1 of 2 DGs provide power, any open pressurizer PORV recloses, and RCP seal cooling is maintained. AFW fails due to a combination of seismic-induced relay chatter from relay group 0.23, group 0.27, and seismic-induced failure of the Auxiliary Building MOVs (SQ07). The CCPs inject, and feed and bleed is successful; however, the CCPs/SIPs fails during recirculation due to seismic-induced failure of the Auxiliary Building MOVs (SQ07) which results in core damage. There is no SBO, the containment is not bypassed, containment isolation fails due to seismic-induce failure of the RB penetrations, and the containment fails early which results in LERF.
9	FL-S-T1 I FL-T1-S03 I EARLY I FLEXAFWFAIL I NO_BYPASS I NON-SBO I PAF I SF-AB-SQ08 I SF-AB-SQ08CFF I SF-IE-T1 I SF-IE-T1CFF I SF-RB-PEN I SF-RB-PENCFF	A seismic-induced loss of offsite power occurs. The reactor is successfully tripped, 1 of 2 DGs provide power, any open pressurizer PORV recloses, and RCP seal cooling is maintained. AFW fails due to a combination of seismic-induced failure of the Auxiliary Building MOVs (SQ08) and failure of the AFW FLEX strategy (see Note 2). The CCPs inject, and feed and bleed is successful; however, the CCPs/SIPs fail during recirculation due to seismic-induced failure of the Auxiliary Building MOVs (SQ08) which results in core damage. There is no SBO, the containment is not bypassed, containment isolation fails due to seismic-induced failure of the RB penetrations, and the containment fails early which results in LERF.

Table 5-6: Sample SLERF Cutsets

ID	Cutset(1)	Description
%G10	FL-DGA-FTR I FL-DGB-FTR I FL-SHLD-S I FL-S-T1 I FL-T1-S08 I FL-T1SAE-S10 I FL-T1S-S16 I EARLY I G10 I NO_BYPASS I PAF I RELAY_0.23 I RELAY_0.23CFF I RELAY_0.27 I RELAY_0.27CFF I SBO I SF-AB-SQ07 I SF-AB-SQ07CFF I SF-IE-T1 I SF-IE-T1CFF I SF-RB-PEN I SF-RB-PENCFF I SF-XPB05 I SF-XPB05CFF	A seismic-induced loss of offsite power occurs. The reactor is successfully tripped, but the diesels fail due to a combination of seismic-induced relay chatter from group 0.23 and seismic-induced failure of the 480V Load center NG03. AEPS is available, any open pressurizer PORV recloses, and the shutdown seal successfully actuates. AFW fails due to the seismic-induced failure of the Aux Building MOVs (SQ07), the CCPs inject, Feed and Bleed is successful, but the CCPs/SIPs fail during recirculation due to the combination of seismic-induced failures which results in core damage. There is no SBO, containment is not bypassed, containment isolation fails due to the seismic-induced failure of the reactor building penetrations, and the containment fails early which results in LERF.

Notes:

1. Events with a suffix "-CFF" are correlation factors (set to 1.0) and are added by FRANX.
2. FLEX events are included in the model to estimate the overall impact on the SPRA model. The FLEX events in the PRA model are currently set to a value of 0.99. A sensitivity study is performed to estimate the overall risk metric reduction from refining these values.

Table 5-7 summarizes the fragilities with a LERF Fussell-Vesely importance of greater than 2%. The Fussell-Vesely for each fragility was approximated by multiplying the median capacity by a factor of 1,000,000, then recalculating the failure probability at each ground motion interval, imposing those failure probabilities onto the cutsets, recalculating the LERF by ACUBE, and finally calculating the percent reduction of total LERF that the improved capacity affords.

Fragilities most important to LERF are those representing SSCs whose failure affects containment isolation and those failures that lead directly to LERF. Fragilities such as Seismic-induced soil failure (SF-SOIL) and seismic-induced failure of the steam generator support (SF-NSSF) lead directly to LERF. Whereas seismic-induced failure of the reactor building penetrations (SF-RB-PEN) result in containment isolation failure. Note that random (non-seismic) failures generally do not contribute significantly to the seismic LERF.

Table 5-7: SLERF Importance Measures Ranked by Fussell-Vesely

Fragility	Description	F-V	A_m (g)	β_u	β_r	Failure Mode	Method
SF-SOIL	Seismic-Induced Soil Failure	19.5%	1.67	0.26	0.24	Soil Bearing Capacity	CDFM
SF-NSSG	Seismic-Induced Failure of the Steam Generator Supports	19.4%	1.86	0.42	0.09	Structural (SG Column)	SOV
SF-RB-PEN	Seismic-Induced Failure of the Reactor Building Penetrations	14.6%	1.78	0.26	0.24	Shear Failure	CDFM
SF-IE-T1	Seismic-Induced Loss of Offsite Power	9.3%	0.3	1.0E-07	0.55	Yard-Centered Loss of Offsite Power	Generic (Conservative Estimate)

5.6 S-PRA Quantification Uncertainty Analysis

Parametric uncertainty in the S-PRA results originates from seismic hazard curve uncertainty, the SSC fragility uncertainties, and basic event failure parameter uncertainties from the internal events PRA. Parametric uncertainty quantification was performed using the UNCERT code in conjunction with the FRANX sampling equation method. Table 5-8 documents the SCDF and SLERF uncertainty quantification results, using 10,000 samples and ACUBE applied to all cutsets.

Table 5-8: Uncertainty Quantification Results(1)

Metric	Point Estimate	Mean	5th	Median	95th	Standard Deviation
SCDF	1.75E-05	7.26E-05	3.99E-06	3.21E-05	2.59E-04	1.41E-04
SLERF	4.72E-06	8.27E-06	2.20E-06	5.67E-06	2.16E-05	9.62E-06

1. The mean values documented in the table are the result of being unable to post-process a significant amount of cutsets with ACUBE when running the uncertainty quantification. Since only a small subset of cutsets (500) could be processed exactly, the mean estimates provided in this table represent an overestimate of the mean SCDF and SLERF; however, the uncertainty quantification provides a relative bound on the SPRA results.

The error factor (defined as [95^{th}/Median], or [Median/5^{th}]) for SCDF and SLERF is approximately eight (8) and four (4) respectively, which is generally high compared to the internal events PRA. This is caused by relatively high fragility uncertainty, indicated by parameters β_u and β_r, as well as hazard uncertainty. The relatively high error factor (as compared to the internal events PRA) resulting from the seismic uncertainty quantification is expected and consistent with industry operating experience.

Figure 5-1 and Figure 5-2 depict the probability density function and cumulative distribution function for SCDF and SLERF. Note that the median was skewed to the right of the mode for seismic CDF and LERF. This is not atypical for Seismic PRAs and is likely due to the high CDF contribution associated with higher g levels (i.e., hazard interval %G10). This shows the impact of the larger uncertainty for the hazard at higher acceleration levels given that the CCDP (0.9) is relatively insensitive to uncertainty.

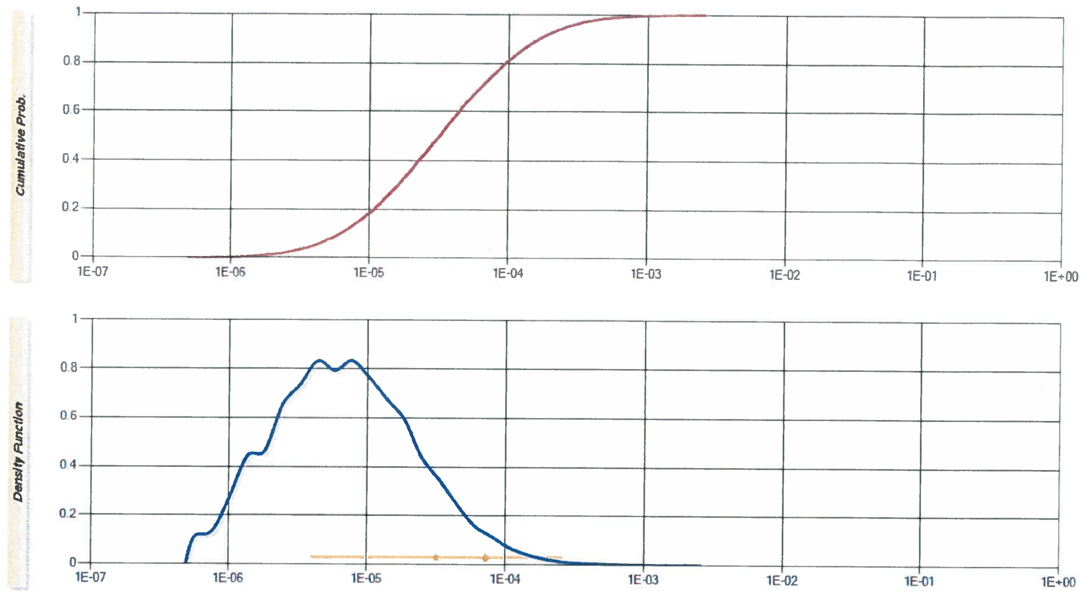

Figure 5-1: Seismic CDF Uncertainty (10,000 sample)

The resulting histogram cumulative distribution function indicates that there is approximately a 80% likelihood that seismic CDF is less than 1E-04/yr and a 20% likelihood that seismic CDF is less than 1E-05/yr.

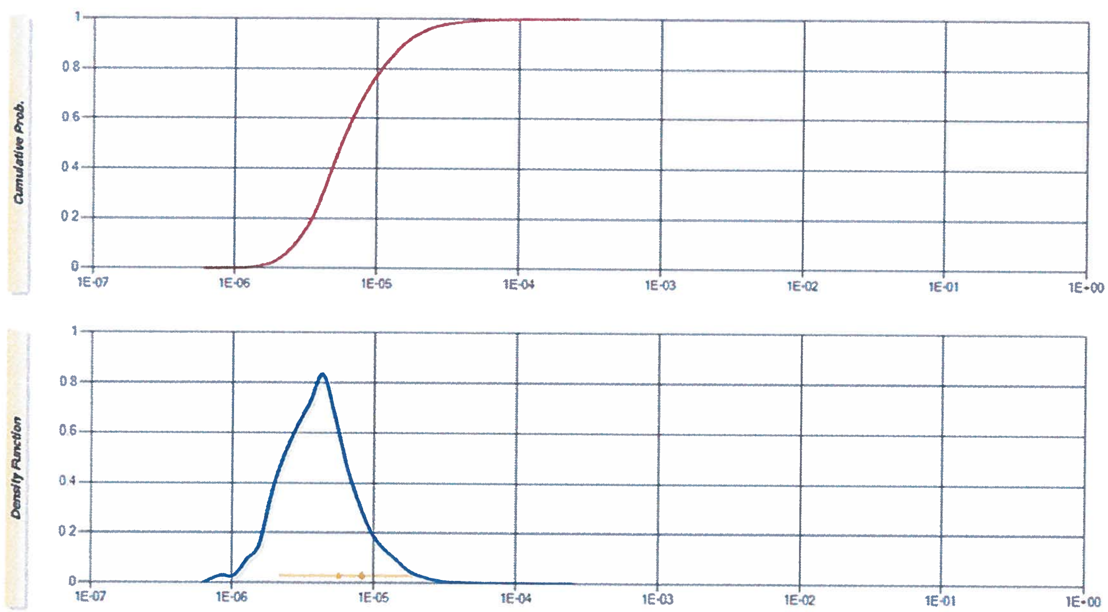

Figure 5-2: Seismic LERF Uncertainty (10,000 sample)

The resulting histogram distribution function indicates that there is approximately a 80% likelihood that seismic LERF is less than 1E-05/yr and nearly a 5% likelihood that seismic LERF is less than 2E-06/yr.

Model uncertainty is introduced when assumptions are made by the S-PRA model and inputs to represent plant response, when there may be alternative approaches to particular aspects of the modeling, or when there is no consensus approach for a particular issue. For the CEC S-PRA, the important model uncertainties are addressed through the sensitivity studies to determine the potential impact on SCDF or SLERF.

Completeness uncertainty relates to potential risk contributors that are not in the model. The scope of the CEC S-PRA is for at-power operation, and does not include risk contributors from low-power shutdown operation, or for spent fuel pool risk. In addition, there may be potential issues related to factors that are not included, such as the impact of aging on equipment reliability and fragility. Other potential issues include impacts of plant organizational performance on risk, and unknown omitted phenomena and failure mechanisms. By their nature, the impacts on risk of these types of uncertainties are not known.

5.7 S-PRA Quantification Sensitivity Analysis

The CEC S-PRA includes quantitative sensitivity studies for the following elements to assess model stability and sources of epistemic uncertainty:

- Truncation Limits for Model Convergence
- Hazard Interval Study
- Non-Safety Component Fragility Sensitivity
- Mission Time Sensitivity
- On-Site FLEX Equipment Sensitivity
- Model Sensitivity to Seismic HRA
- Model Sensitivity to Open F&Os

5.7.1 Truncation Limits for Model Convergence

In the framework of seismic PRA, truncation is only one of the aspects used to assess model stability, others being considerations on the extension of the hazards integration and size and number of the integration bins (discussed in other sections). Truncation is more relevant at lower g levels (where additional cutsets can increase CCDP) and it becomes less and less relevant at higher g levels, with CCDP approaching 1.0. At the top level, if CCDP is 1.0, truncation is irrelevant.

QU-B3 of ASME/ANS RA-Sb-2013 [20] specifically discusses assessing truncation against the overall model and provides examples of truncation limits for internal events. In traditional truncation studies performed for Internal Events PRA, cutsets are truncated until successive reductions in the truncation limit by one decade result in a total plant CDF and LERF which converge to a value less than 5% different from the previous quantification. For seismic PRAs, demonstrating convergence using the traditional definition of less than 5% change from the previous quantification can be challenging due to more complex models and computing and memory limitations and is unnecessary to prove model stability, based on the discussion above.

The truncation study is performed across each ground motion using a range of truncation limits as opposed to a single truncation limit for all ground motions. The percent change shown per decade is calculated as the delta result for each hazard interval contribution in comparison to total CDF/LERF. ACUBE was applied to all cutsets for each ground motion interval. It can be estimated that the sum of each hazard interval contribution would represent the total change in CDF/LERF.

The truncation study documented in **Error! Reference source not found.** and **Error! Reference source not found.** is performed across each ground motion using a range of truncation limits as opposed to a single truncation limit for all ground motions.

At low g levels, where the hazard interval bins contribute less than 5% to total CDF and LERF, decreasing the truncation limit further for these bins would not impact the overall risk insights from the S-PRA model, even though additional insights at low g level may be gained through generation of additional cutsets (although it can be noted that at lower g level the seismic failures have lower probability and the insights tend to converge to the insights of the internal events PRA). At high g levels, where the CCDP/CLERP of the hazard interval bins is approaching 1.0 (the PAF at 0.9 limits the CCDP to .9), no additional insights can be gained by decreasing truncation since regardless of the number of cutsets generated the numerical solution will not be impacted. Based on the assessment of each hazard interval as compared to total CDF/LERF it is shown that for each decrease in truncation, a significant impact to the total results from any one hazard interval does not occur. Due to computing and memory limitations, quantification below these truncation limits cannot be achieved at this point; however it is qualitatively assessed that the overall risk insights would not change, even if truncation limits could be lowered for a given hazard interval bin. Based on the above considerations, it is judged that the model is sufficiently converged, consistent with the intention of the standard.

Table 5-9: Model Convergence on Truncation Value - CDF

Scenario	Truncation	# Cutsets	MCUB	% Cutsets Post-Processed)	Lower ACUBE CDF (1)	ACUBE CDF (2)	Lower ACUBE Percent Change	ACUBE Percent Change(3)
%G01	1.00E-09	2	3.17E-09	100.00%	3.17E-09	3.17E-09	N/A	N/A
	1.00E-10	39	4.39E-07	100.00%	2.89E-07	2.89E-07	9016.72%	9016.72%
	1.00E-11	176	4.57E-07	100.00%	2.95E-07	2.95E-07	2.08%	2.08%
%G02	1.00E-09	50	2.62E-06	100.00%	1.17E-06	1.17E-06	N/A	N/A
	1.00E-10	361	3.72E-06	100.00%	1.77E-06	1.77E-06	51.28%	51.28%
	1.00E-11	1706	4.30E-06	100.00%	1.96E-06	1.96E-06	10.73%	10.73%
	1.00E-12	8796	5.19E-06	11.37%	2.22E-06	2.26E-06	13.27%	15.31%
	1.00E-13	43587	6.79E-06	2.29%	2.43E-06	2.53E-06	9.46%	11.95%
%G03	1.00E-08	35	3.63E-06	100.00%	2.44E-06	2.44E-06	N/A	N/A
	1.00E-09	395	7.56E-06	100.00%	3.71E-06	3.71E-06	52.05%	52.05%
	1.00E-10	2583	1.22E-05	77.43%	4.98E-06	4.98E-06	34.23%	34.23%
	1.00E-11	14365	1.89E-05	10.44%	5.91E-06	6.23E-06	18.67%	25.10%
	1.00E-12	69028	3.22E-05	1.45%	6.50E-06	7.38E-06	9.98%	18.46%
%G04	1.00E-07	4	9.67E-07	100.00%	9.43E-07	9.43E-07	N/A	N/A
	1.00E-08	94	4.23E-06	100.00%	2.52E-06	2.52E-06	167.23%	167.23%
	1.00E-09	956	9.27E-06	100.00%	3.49E-06	3.49E-06	38.49%	38.49%
	1.00E-10	6646	1.72E-05	37.62%	4.46E-06	4.59E-06	27.79%	31.52%
	1.00E-11	36909	3.25E-05	6.77%	5.05E-06	5.64E-06	13.23%	22.88%
	1.00E-12	165180	5.35E-05	0.91%	5.24E-06	6.58E-06	3.76%	16.67%

Table 5-9: Model Convergence on Truncation Value - CDF

Scenario	Truncation	# Cutsets	MCUB	% Cutsets Post-Processed)	Lower ACUBE CDF (1)	ACUBE CDF (2)	Lower ACUBE Percent Change	ACUBE Percent Change(3)
%G05	1.00E-07	6	1.54E-06	100.00%	1.45E-06	1.45E-06	N/A	N/A
	1.00E-08	223	8.94E-06	100.00%	3.70E-06	3.70E-06	155.17%	155.17%
	1.00E-09	2389	2.13E-05	100.00%	4.55E-06	4.55E-06	22.97%	22.97%
	1.00E-10	15940	4.45E-05	18.82%	5.44E-06	5.96E-06	19.56%	30.99%
	1.00E-11	80032	7.85E-05	3.12%	5.75E-06	7.04E-06	5.70%	18.12%
%G06	1.00E-07	26	4.59E-06	100.00%	2.88E-06	2.88E-06	N/A	N/A
	1.00E-08	685	2.59E-05	100.00%	4.46E-06	4.46E-06	54.86%	54.86%
	1.00E-09	7148	8.12E-05	41.97%	5.57E-06	5.68E-06	24.89%	27.35%
	1.00E-10	45447	2.00E-04	2.86%	5.87E-06	7.72E-06	5.39%	35.92%
%G07	1.00E-07	57	9.85E-06	100.00%	3.40E-06	3.40E-06	N/A	N/A
	1.00E-08	1456	5.60E-05	100.00%	4.13E-06	4.13E-06	21.47%	21.47%
	1.00E-09	14116	1.70E-04	21.25%	4.64E-06	5.35E-06	12.35%	29.54%
	1.00E-10	81437	3.90E-04	2.46%	4.76E-06	5.85E-06	2.59%	9.35%
%G08	1.00E-07	483	1.16E-04	100.00%	6.46E-06	6.46E-06	N/A	N/A
	1.00E-08	9056	8.94E-04	100.00%	6.58E-06	6.58E-06	1.86%	1.86%
%G09	1.00E-07	1381	3.02E-04	100.00%	3.93E-06	3.93E-06	N/A	N/A
	1.00E-08	21332	2.08E-03	100.00%	3.95E-06	3.95E-06	0.51%	0.51%
%G10	1.00E-06	585	8.69E-04	100.00%	7.92E-06	7.92E-06	N/A	N/A
	1.00E-07	17824	8.64E-03	100.00%	7.95E-06	7.95E-06	0.38%	0.38%

Table 5-9: Model Convergence on Truncation Value - CDF

Scenario	Truncation	# Cutsets	MCUB	% Cutsets Post-Processed)	Lower ACUBE CDF (1)	ACUBE CDF (2)	Lower ACUBE Percent Change	ACUBE Percent Change(3)

Notes:

1. The Lower ACUBE result represents only the portion of the results that have been processed exactly.
2. The ACUBE results are based on the number of cutsets post-processed. The values do not necessarily reflect the best estimate of CDF for cases in which all cutsets are not post-processed. Note that the ACUBE results represent the sum of the cutsets processed exactly and those that are estimated using the MCUB approximation.
3. Percent change represents the change in the ACUBE CDF results across each decade. Note in some cases all cutsets could not be post-processed in ACUBE. It is possible that convergence may be shown at higher truncation levels if all cutsets could be post-processed. For this reason, the percent reduction between the lower ACUBE range is shown.

Table 5-10: Model Convergence on Truncation Value - LERF

Scenario	Truncation	# Cutsets	MCUB	% Cutsets Post-Processed)	Lower ACUBE LERF (1)	ACUBE LERF (2)	Lower ACUBE Percent Change	Percent Change(3)
%G01	1.00E-11	3	4.44E-11	100.00%	4.44E-11	4.44E-11		
	1.00E-12	41	1.32E-10	100.00%	1.26E-10	1.26E-10	183.78%	183.78%
	1.00E-13	361	8.12E-10	100.00%	7.96E-10	7.96E-10	531.75%	531.75%
	1.00E-14	1546	8.98E-10	100.00%	8.73E-10	8.73E-10	9.67%	9.67%
%G02	1.00E-09	4	8.42E-09	100.00%	8.42E-09	8.42E-09		
	1.00E-10	16	1.17E-08	100.00%	1.15E-08	1.15E-08	36.58%	36.58%
	1.00E-11	224	3.39E-08	100.00%	2.45E-08	2.45E-08	113.04%	113.04%
	1.00E-12	2189	4.22E-08	100.00%	3.65E-08	3.65E-08	48.98%	48.98%
%G03	1.00E-08	2	3.82E-08	100.00%	3.81E-08	3.81E-08		
	1.00E-09	4	4.13E-08	100.00%	4.13E-08	4.13E-08	8.40%	8.40%
	1.00E-10	294	9.91E-08	100.00%	6.21E-08	6.21E-08	50.36%	50.36%

Table 5-10: Model Convergence on Truncation Value - LERF

Scenario	Truncation	# Cutsets	MCUB	% Cutsets Post-Processed)	Lower ACUBE LERF (1)	ACUBE LERF (2)	Lower ACUBE Percent Change	Percent Change(3)
%04	1.00E-07	1	1.17E-07	100.00%	1.17E-07	1.17E-07		
	1.00E-08	3	2.00E-07	100.00%	1.99E-07	1.99E-07	70.09%	70.09%
	1.00E-09	708	1.43E-06	100.00%	2.75E-07	2.75E-07	38.19%	38.19%
%G05	1.00E-07	2	3.35E-07	100.00%	3.28E-07	3.28E-07		
	1.00E-08	1934	1.94E-05	100.00%	4.93E-07	4.93E-07	50.30%	50.30%
%G06	1.00E-07	7	9.12E-07	100.00%	4.71E-07	4.71E-07	N/A	N/A
	7.00E-08	208	1.07E-05	100.00%	4.75E-07	4.75E-07	N/A	N/A
%G07	1.00E-07	768	4.40E-05	100.00%	4.59E-07	4.59E-07	N/A	N/A
%G08	1.00E-07	3523	1.48E-04	100.00%	3.60E-07	3.60E-07	N/A	N/A
%G09	1.00E-07	2558	1.92E-04	100.00%	4.23E-07	4.23E-07	N/A	N/A
%G10	1.00E-07	15360	9.08E-04	100.00%	3.20E-07	3.20E-07	N/A	N/A

Table 5-10: Model Convergence on Truncation Value - LERF

Scenario	Truncation	# Cutsets	MCUB	% Cutsets Post-Processed)	Lower ACUBE LERF (1)	ACUBE LERF (2)	Lower ACUBE Percent Change	Percent Change(3)

Notes:

(1) The Lower ACUBE result represents only the portion of the results that have been processed exactly.

(2) The ACUBE results are based on the number of cutsets post-processed. The values do not necessarily reflect the best estimate of LERF for cases in which all cutsets are not post-processed. Note that the ACUBE results represent the sum of the cutsets processed exactly and those that are estimated using the MCUB approximation.

(3) Percent change represents the change in the ACUBE LERF results across each decade. Note in some cases all cutsets could not be post-processed in ACUBE. It is possible that convergence may be shown at higher truncation levels if all cutsets could be post-processed. For this reason, the percent reduction between the lower ACUBE range is shown.

5.7.2 Hazard Interval Study

The CEC S-PRA discretizes the seismic hazard for both CDF and LERF. For CDF, the hazard is discretized, starting at 0.1g, into 10 intervals with varying widths (.1g for %G01to %G03 and %G08 to %G10, and 0.05 g for %G04 to %G07). For LERF, the hazard is discretized, starting at 0.1g, into 10 intervals, with varying widths (0.1g for %G01, and %G03, 0.3g for %G02, and 0.2g for %G04 to %G10). Convolution of the mean hazard with the point estimate plant level fragilities for core damage and large early release confirmed that the CDF and LERF estimates are converged at the selected 10-interval hazard discretization.

5.7.3 Non-Safety Component Fragility Sensitivity

Non-safety components basic events were assigned a generic fragility value and were assumed to be fully correlated. This treatment is generally regarded to be conservative. However, to ensure that the impact of crediting non-safety equipment with a generic fragility is not a significant contributor to seismic risk, a sensitivity analysis was performed by setting the associated NSCI fragility to true. The sensitivity case resulted in a 2.27% increase in CDF and a 0% change in LERF.

5.7.4 Mission Time Sensitivity

A mission time of 24 hours was used in the CEC S-PRA model consistent with the internal events model. For this sensitivity study, all CEC basic events associated with a mission time were altered from the standard 24 hour mission time to a 48 hour mission time. The sensitivity case resulted in a 2.65% increase in CDF and a 0.34% change in LERF.

5.7.5 On-site FLEX Equipment Sensitivity

For this sensitivity the potential impact of crediting FLEX was investigated. FLEX equipment has the potential to be incorporated into a plant given a severe accident on site and with some assumed conditions during the extreme event. The S-PRA model currently includes minimal credit for this equipment in the baseline model (flags set to 0.99). For this sensitivity the flags were modified and set to an optimistic value of 0.1. The sensitivity case resulted in an 11.47% decrease in CDF and a 1.38% decrease in LERF.

5.7.6 Model Sensitivity to Open F&Os

This sensitivity looks at the potential impact of open F&Os on the SPRA model and results. One F&O remains open that has not been resolved in the internal events PRA model related to loss of room cooling and impact on devices such as digital I&C equipment and protection devices. A qualitative assessment is performed to characterize the impacts of the F&O on the S-PRA model. Using industry guidance, the open F&O is expected to have minor impacts on the model, and furthermore, not expected to impact risk insights of the current SPRA model results presented in this submittal. The following discussion provides more detail:

The F&O Closure resulted in closure of all open internal events, internal flood, high winds, and seismic PRA F&Os with the exception of four (4) F&Os.

- Two SRs (SPR-B2, and SPR-E6) remain at less than CC II due to F&Os (25-12, and 25-19) from the Seismic PRA peer review.
- One F&O (22-3) remains open from the internal events PRA Peer review.
- One new F&O (13-1) was written based on the results from a focused-scope peer review in regards to LE-D6.

The two open F&Os (25-12 and 25-19) from the seismic PRA peer review are related to open findings in the internal events model. These F&Os are related to supporting requirements that require a review of any open F&Os pertaining to the internal events model. All internal events F&Os have been closed with the exception of two (22-3 and 13-1). All updates to the PRA model and supporting documentation as a result of the internal events F&O closure have been incorporated into the SPRA model documented in this submittal.

In regards to F&O 13-1, the CEC internal events PRA model was updated with a plant-specific analysis to determine values for PI-SGTR and TI-SGTR. These values have been incorporated into the internal events PRA model used as the backbone of the SPRA model; therefore, no assessment is needed in regards to the impact of this F&O on the SPRA model.

The open F&O (22-3) is discussed below:

F&O 22-3 (SC-B4):

GOTHIC temperature failure acceptance criterion are addressed in Section 4.2 of the Room Cooling Analysis Notebook. Exceptions to the temperature failure acceptance criterion include the digital I&C equipment and protective devices. The content associated with exception are contained in Appendix A and Appendix B which is not available and remains an open item.

Provide documentation of the exceptions to the temperature failure acceptance criterion include the digital I&C equipment and protective devices.

Recently, the PWROG released a topical report for considering loss of room cooling in PRA models (PWROG-18027-NP). The report provides recommended screening criteria for excluding room cooling in PRA models, but also discusses modeling options if the screening criteria cannot be met. The screening criteria, in general, recommends screening out room cooling if room temperatures, when stabilized, do not exceed 150°F. The report lists a subset of caveats to this requirement specifically for: equipment housed in non-vented cabinets, digital I&C cabinets and consoles, and equipment protection devices. The report notes that for equipment protection devices and digital I&C cabinets and consoles

<u>Digital I&C Cabinets and Consoles:</u>
This equipment is especially vulnerable to loss of room cooling and the general screening criteria of 150°F over the PRA mission time should not be applied to such components. These components are generally kept at less than 80°F and likely only capable of functioning up to ~120°F due to sensitivity of temperature increases.

<u>Equipment protection devices:</u>
Circuit breakers, thermal overload relays and fuses are used to protect equipment form overcurrent conditions. Such devices would need to have margin for derating to support continued operation without tripping.

Current draft GOTHIC calculations from Callaway show that the 120°F for certain areas with protection equipment may be exceeded. The PWROG-18027-NP report recommends two options for components that cannot be screened out of the PRA: (1) model the components with a realistic failure probability when temperature limits have been exceeded or (2) model the components as failed when temperature limits have been exceeded. Modeling approach 1 is the preferred approach as it allows the most realistic modeling. In addition to this, an HVAC model would need to be developed to consider potential mitigation of the loss of room cooling.

Currently, the Callaway model does not include HVAC models for some of the areas of concern, and therefore, would need to be developed. Previous revisions of the CEC IE model evaluated the failure of HVAC which included independent failures of the room coolers and subsequent operator recovery in scenarios where the components were unavailable. This model was assessed to have a failure of approximately 2E-06 which does not include an additional probability of failure (failure is assumed) for the specified equipment after reaching its temperature limitation.

The quantitative impact of this modeling may be more significant when considering impacts from a seismic event. Seismic events usually result in failure of multiple equipment and some of the redundancy in the mitigation may be lost. An increase iof an order of magnitude or two is not unreasonable from the quantitative aspect, but still results in a relatively low failure probability as compared to other seismic-induced failures.

Furthermore, current draft GOTHIC calculations show that temperature limitations are not reached until approximately the 19-hour mark after the loss of room cooling. Given that the modeled mission time of 24 hours, an increased failure rate would only be applicable over a short period of time, reducing the overall impact of the failure mode on the model results. Additionally, it is expected that given that the heat up would occur over a period of hours, operators would likely take actions to open doors and establish some type of compensatory measure to provide room cooling. While this impact is not currently quantified, it provides additional assurance that the probability of equipment failure is further reduced. Given these considerations, the overall impact on the SPRA model is expected to be negligible and would not impact risk insights from the current SPRA model evaluation.

5.7.7 Summary of Sensitivity Study Results

Table 5-11 summarizes the quantitative sensitivity study results discussed in preceding sections.

Table 5-11: Summary of Quantitative Sensitivity Study Results

Study	ΔCDF	ΔLERF
Non-Safety Component Fragility	+2.27%	0%
Mission Time	+2.65%	0.34%
On-site FLEX Equipment	-11.47%	-1.38%
Seismic HRA	-5.8%	-1.4%

5.8 S-PRA Quantification Technical Adequacy

The CEC S-PRA quantification methodology and analysis were subjected to an independent peer review against the pertinent requirements in the ASME/ANS PRA Standard [13]. The S-PRA was peer reviewed relative to Capability Category II for the full set of requirements in the Standard. After completion of the subsequent independent assessment, the full set of supporting requirements was met with the exception of SPR-B2 and SPR-E6. The seismic plant response quantification analysis was determined to be acceptable to use in the S-PRA. The peer review Facts and Observations (F&O) through an independent assessment, is further described in Appendix A and References [14] and [15].

6.0 Conclusions

A S-PRA has been performed for Callaway Energy Center in accordance with applicable requirements of the ASME/ANS PRA Standard [13]. The S-PRA shows that the point estimate seismic CDF is 5.59E-05/yr and the seismic LERF is 2.90E-06/yr. Uncertainty, importance, and sensitivity analyses were performed. Sensitivity studies were performed to identify critical assumptions, test the sensitivity to quantification parameters and the seismic hazard, and identify potential areas to consider for the reduction of seismic risk.

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8.0 Acronyms

ACB	Auxiliary and Control Buildings
ADAMS	Agencywide Documents Access and Management System
AEPS	Alternate Electric Power System
AFW	Auxiliary Feedwater
ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient without SCRAM
BE	best-estimate
CAFTA	Computer Aided Fault Tree Analysis
CCDP	Conditional Core Damage Probability
CCW	Component Cooling Water
CDF	Core Damage Frequency
CDFM	Conservative Deterministic Failure Margin
CEC	Callaway Energy Center
CEUS	Central and Eastern United States
CLERF	Conditional Large Early Release Frequency
CLERP	Conditional Large Early Release Probability
CVCS	Chemical and Volume Control Systems
EDG	Emergency Diesel Generator
EL	Elevation
EPRI	Electric Power Research Institute
FIRS	Foundation Input Response Spectra
FLEX	Diverse and Flexible Mitigation Capability
F&Os	Facts & Observations
F-PRA	Fire Probabilistic Risk Assessment
fsp	feet per section
FTR	Fail to Run
FTS	Fail to Start
F-V	Fussell-Vesely
GERS	generic equipment ruggedness spectra
GMC	Ground Motion Characterization
GMM	Ground Motion Model
GMRS	Ground Motion Response Spectra
GUI	General User Interface
HCLPF	High Confidence of Low Probability of Failure
HEP	Human Error Probability
HFE	Human Failure Event
HLR	High Level Requirement
HRA	Human Reliability Analysis
IE	Internal Events
IEPRA	Internal Events Probabilistic Risk Assessment
ISRS	Instructure Response Spectra
JCNRM	Joint Committee on Nuclear Risk Management
LB	lower bound
LERF	Large Early Release Frequency
LLOCA	Large Loss of Coolant Accident
LMSM	

LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LOSP	Loss of Offsite Power
MLOCA	Medium Loss of Coolant Accident
MSO	Multiple Spurious Operations
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NSCI	Non-Seismic Class I
NTTF	Near-Term Task Force
OBE	Operating Basis Earthquake
PAF	Plant Availability Factor
PGA	Peak Ground Acceleration
PRA	Probabilistic Risk Assessment
PSHA	Probabilistic Seismic Hazard Analysis
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RRW	Risk Reduction Worth
RWST	Refueling Water Storage Tank
SBO	Station Blackout
SCDF	Seismic Core Damage Frequency
SEL	Seismic Equipment List
SIET	Seismic Initiating Event Tree
SLERF	Seismic Large Early Release Frequency
SOV	Separation of Variables
SPID	Screening, Prioritization and Implementation Details
S-PRA	Seismic Probabilistic Risk Assessment
SRA	Site Response Analysis
SSC	Seismic Source Characterization (used in Seismic Hazard discussion)
SHHAC	Senior Seismic Hazard Analysis Committee
SSC	Structures, Systems and Components.
SSI	Soil Structure Interaction
TDAFP	Turbine-Driven Auxiliary Feedwater Pump
UHRS	Uniform Hazard Response Spectra
UHS	Ultimate Heat Sink
UP	upper bound
V/H	vertical-to-horizontal
VSLOCA	Very Small Loss of Coolant Accident

Appendix A - Summary of S-PRA Peer Review and Assessment of PRA Technical Adequacy

This Appendix provides a summary of the peer review of the CEC S-PRA, the peer review F&O closure reviews, and provides the bases for why the S-PRA is technically adequate for the response to the NRC's request under 50.54(f) [1].

The CEC S-PRA was subjected to an independent peer review against the pertinent requirements of the ASME/ANS PRA Standard [13] as detailed in Section A.1. The information presented in this Appendix establishes that the S-PRA has been peer reviewed by a team with adequate credentials to perform the assessment, establishes that the peer review process followed meets the intent of the peer review characteristics and attributes in Table 16 of RG 1.200, Revision 2 [23].

A.1 Overview of the Peer Review

The peer review assessment [14], and subsequent disposition of peer review findings [15] are summarized in this Appendix. The scope of the reviews encompassed the set of technical elements and supporting requirements (SRs) for the SHA (seismic hazard), SFR (seismic fragilities), and SPR (seismic PRA modeling) elements for seismic CDF and LERF.

The CEC S-PRA peer review occurred during the week of June 18, 2018. The CEC S-PRA F&O closure review occurred during the week of March 11, 2019.

A.2 Summary of the Peer Review Process

The June 2018 peer review [14] was performed against the requirements in the PRA Standard Code Case for Part 5 [13], using the peer review process defined in NEI 12-13 [24]. For supporting requirements in the Code Case that referred back to requirements in Part 2, Addendum B of the PRA Standard [21] was utilized.

The NEI 12-13 S-PRA peer review process [24] involves an examination by each reviewer of their assigned PRA technical elements against the requirements in the Standard [20] and [13] to ensure the robustness of the model relative to all of the requirements.

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

For each technical element, i.e., SHA, SFR, and SPR, a team of two (at least two) 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 (3) types of F&Os defined in NEI 12-13 [24]: 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 reviews have noted as potentially important but not requiring resolution to meet the SRs; and Best Practices

– which reflect the reviewers' opinion that a particular aspect of the review exceeds normal industry practice. The focus in this Appendix is on Findings and their disposition relative to this submittal.

A.3 Peer Review Team Qualifications

The June 2018 CEC S-PRA peer review was led by Mr. Kenneth Kiper of Westinghouse Electric Company. Team members included: Dr. Martin McCann of Jack Benjamin & Associates, Dr. Glenn Rix of Geosyntec Consultants, Inc., Stephen J. Eder of Facility Risk Consultants, Inc., Dr. Ram Srinivasan, independent consultant, Colter D. Somerville of Southern Nuclear Operating Company, Rick Summit of RLS Consulting, and Deepak Rao of Entergy Nuclear. The lead and reviewer qualifications were reviewed by Ameren Missouri and have been confirmed to be consistent with requirements in the ASME/ANS PRA Standard [20] and [13] and the guidelines of NEI-12-13 [24]. The members of the peer review team were independent of the CEC S-PRA. They were not involved in performing or directly supervising work on any PRA Element evaluated in the overall CEC S-PRA.

Mr. Kenneth Kiper, the team lead, has over 35 years of experience at Westinghouse and, previously at Seabrook Station, in the nuclear safety area generally and PRA specifically for both existing and new nuclear power plants. He has led a number of peer reviews, including reviews of internal events PRAs, internal flood PRAs, fire PRAs, high wind PRAs, and several S-PRAs

Dr. Martin McCann was the lead for the review of the Seismic Hazard Analysis (SHA) technical element. Dr. McCann has over 35 years of experience in engineering seismology including site response analysis and specification of ground motion. Dr. McCann has served as SHA lead reviewer for a number of recent S-PRAs.

Dr. McCann was assisted by Dr. Glenn Rix. Dr. Rix has nearly 30 years of experience in seismic hazard evaluation, geotechnical earthquake engineering, and performance-based and risk-based analyses. Dr. Rix joined Geosyntec in 2013 after a distinguished 24 year career as a Professor in the School of Civil and Environmental Engineering at the Georgia Institute of Technology specializing in geotechnical and earthquake engineering. Dr. Rix has participated in a number of S-PRA peer reviews.

Mr. Stephen J. Eder led the Seismic Fragility Analysis (SFR) review. Mr. Eder is an industry leader in the seismic fragility analysis of systems and components at nuclear power facilities with over 35 years of experience. He has led the seismic probabilistic risk analysis fragility peer reviews for Beaver Valley, Davis Bessie, Fermi, Perry, and Vogtle nuclear power plants. He has provided overall technical direction for walkdowns and seismic fragility analyses for a number of U.S. nuclear power plants.

Mr. Eder was assisted by Dr. Ram Srinivasan and Mr. Colter D. Somerville. Dr. Srinivasan has over 45 years of experience in the nuclear industry, principally in the design, analysis (static and dynamic, including seismic), and construction of nuclear power plant structures. He is actively involved in the Post-Fukushima Seismic Assessments (NRC NTTF 2.1 and 2.3) and is a member of the NEI Seismic Task Force and the ASME/ANS JCNRM, Part 5 Working Group (Seismic and other External Hazards PRA). He has participated on several previous S-PRA peer reviews, either as reviewer or utility consultant.

Mr. Colter D. Somerville is the seismic technical support engineer for the Southern Nuclear Operating Company (SNC) nuclear power plants, including Vogtle, Hatch, and Farley. He has participated in several peer reviews defending the seismic fragility work for the SNC plants. He was a working observer at two S-PRA peer reviews in 2017, Indian Point 2 and Diablo Canyon.

Mr. Rick Summitt was the lead for the review of the Seismic System Response Analysis (SPR) technical element. Mr. Summitt has 38 years of experience in risk analysis and has used both deterministic and probabilistic techniques to manage and supply lead technical guidance in the assessment of safety and reliability concerns for both nuclear and chemical projects.

Mr. Summitt was assisted by Mr. Deepak Rao. Mr. Rao is a nuclear engineer and mechanical engineer with over 30 years' experience working in the nuclear power industry. He has supported a number of industry PRA peer reviews, including internal events, Fire PRA, Seismic PRA and High Winds PRA in the last several years. He has participated in a number of peer reviews, including internal events PRAs and S-PRAs.

Dr. Mohamed Talaat from Simpson, Gumpertz & Heger, Inc. and Mr. JC Patel from Wolf Creek Nuclear Operating Company served as working observers for the SFR and SPR technical elements, respectively. Any observations and findings that Dr. Talaat or Mr. Patel generated were given to the peer review team for their review and ownership. As such, Dr. Talaat and Mr. Patel assisted with the review but were not formal members of the peer review team.

A.4 Summary of the Peer Review Conclusions

The review team's assessment of the S-PRA elements is excerpted from the peer review report [14] 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.

A.4.1 Seismic Hazard Analysis

The Standard requires the seismic hazard input to the S-PRA be determined on the basis of a site-specific probabilistic seismic hazard analysis (PSHA). The site-specific PSHA must be based on "current geological, seismological, and geophysical data; local site topography; and surficial geologic and geotechnical site properties." The Callaway PSHA:

1. Used the existing regional Seismic Source Characterization (SSC) model for the Central and Eastern United States (CEUS);
2. Used the regional CEUS ground motion characterization (GMC) models;
3. Evaluated the effects of local site conditions on the ground motions that would be experienced by plant structures, systems and components; and
4. Considered the potential for other seismic hazards, including ground failures that might occur due to soil liquefaction, slope failures, fault displacement.

The existing regional-scale SSC model for the CEUS largely meets the intent of the Standard for sites in the CEUS. At the same time the Standard is clear that the SSC model must be based on current information. The review team understands that some work was done to compile an up-to-date earth science database (available since the CEUS SSC model was developed) that could impact the estimate of the seismic hazard at the site. These impacts could be modifications to existing seismic source models or the addition of new sources. The work that was done in terms of compiling new information and evaluating its impact does not fully meet the intent of the Standard. Specifically, the documentation needs to be enhanced to explicitly summarize the information reviewed and the technical basis for determining no impacts on the PSHA input models used. For this reason, SHA-C4 which relates to the evaluation of this new information was Not Met.

The PSHA requirements associated with incorporating the effects of local site conditions on ground motions are defined in SHA-E. The effect of site conditions were modeled by means of amplification factors derived from probabilistic site response analyses that incorporate site-specific information on surficial geologic deposits, and site geotechnical properties. Epistemic uncertainty and aleatory variability in shear wave velocity, layer thickness, and nonlinear properties are considered and propagated separately in the site response analyses.

The Standard requires that a screening analysis be performed to assess whether in addition to vibratory ground motion, other seismic hazards, such as fault displacement, landslide, soil liquefaction, or soil settlement, need to be included in the S-PRA. As part of the PSHA, a systematic evaluation was carried out for other hazards.

SHA-J defines the requirements for documentation of the PSHA. These requirements set a high bar with regard to the documentation that should be provided. It must support PRA applications, peer review, and future updates. Overall, the documentation for the PSHA is generally complete and meets the intent of the Standard; therefore the Supporting requirements for SHA-J are met. The documentation of the Callaway PSHA is provided in a group of documents, including the PSHA contractor self-assessment. However, there are elements of the PSHA documentation that could be better integrated and enhanced. Several of the findings are associated with PSHA documentation issues and, if these are adequately addressed, will result in PSHA documentation that meets the intent of the Standard.

A.4.2 Seismic Fragility Analysis

The SFR assessment covered the three principal elements of the fragility analysis; namely, site-specific seismic response analysis, plant walkdown, and fragility analysis calculations. A summary of the three elements are briefly summarized below.

Site-specific seismic response analysis of the various buildings housing the SEL items was performed using the ground motions corresponding to the Uniform Hazard Response Spectra (UHRS) shape provided in the plant Probabilistic Seismic Hazard Analysis (PSHA) reports. The Reference Level Earthquake (RLE) utilized corresponds to the GMRS, which is anchored to 0.39g peak ground acceleration (PGA). Selection of this level was based on the interim results of the PRA quantifications and corresponds to the expected failure levels of most of the top contributors to seismic risk. The appropriateness of this ground motion level was reviewed relative to the capacities of top contributors to CDF/LERF. The review showed that the selected 0.39g PGA is appropriate for top contributors to CDF. However, the failure level of the top contributors to LERF occurs for ground motion levels significantly exceeding 0.4g PGA.

The seismic input to the building response analysis is based on a single time history set (3 components) matched to the GMRS. Structural properties (uncracked concrete) and strain dependent soil properties compatible with the GMRS conditions were used in the building models. Out-of-plane cracking of floor slabs was not considered. The development of building models relies heavily in the modification of existing lumped-mass stick models. These modifications are documented in the respective calculations. Vertical floor slab flexibility was neglected.

For the Auxiliary and Control Buildings (ACB), a new 3D finite element building model was developed. Structural stiffness variation was not addressed. Dynamic analyses included soil-structure interaction (SSI) analysis of structures. SSI analysis for the ACB was performed using the software ACS-SASSI (3D finite element model) while for the other buildings (lumped-mass stick models) EKSSI was used. EKSSI uses impedance functions to represent the soil mass and stiffness properties. The SSI analyses considered three soil profiles; best-estimate, lower-bound, and upper-bound; and a median-centered structural model. Instructure Response Spectra (ISRS) were generated at various locations in the different buildings for the fragility analysis of the components. Some of the ISRS results did not appear to be realistic.

A walkdown of the Callaway Nuclear Plant was performed as part of this Peer Review. The walkdown review focused on the dominant risk contributors in seismic CDF and LERF, but also included some additional example SSCs to confirm findings or observations made by the Seismic Review Team. The peer review walkdown included the Auxiliary Building, Control Building, Diesel Generator Building, Essential Service Water System (ESWS) Pumphouse, ultimate heat sink (UHS) Cooling Tower, Turbine Building, and the Yard.

The peer review team observed the plant to be spacious, well laid out, and free of congestion. Equipment was observed to be well anchored, and piping and pipe supports appeared to be rugged. The level of seismic housekeeping in the plant was observed to be very good. It was a general observation that the robustness of the plant was not reflected in the current seismic capacities determined by the fragility analysis.

The S-PRA seismic walkdowns by the seismic review team (SRT) were performed in four sessions and reviewed attributes for functional capacity, anchorage, and spatial interactions for each component. During the walkdown, seismic vulnerabilities were identified and documented by the SRT. However, it became apparent to the peer review team that no follow-up walkdowns had been conducted to verify and ensure that the governing failure modes used in the fragility calculations were realistic and plant specific.

The S-PRA seismic walkdown addressed seismic-induced fire and flood. The seismic-induced fire review focused on the hydrogen piping system, which was found to be rugged. However, the potential for fire initiation due to seismic failure of high-voltage non-safety electrical cabinets was not investigated. The seismic-induced flood review focused on the fire protection system and identified several vulnerabilities. However, pre-action fire protection systems were assumed to be dry and were not investigated for potential seismic vulnerabilities as a potential flood source.

Fragility parameters were calculated for all the SEL items credited in the plant S-PRA Model. No capacity-based screening was performed. Rather, surrogate fragility parameters were developed and used for the components judged to be of higher seismic capacity. The failure modes considered in the building structure and retention pond fragility evaluations included structural integrity, soil bearing capacity, and seismic interaction considerations. The failure modes considered by the fragility evaluations included anchorage, functional, and seismic interaction considerations.

The fragility calculations for the top contributors to SCDF were reviewed and found to be from an initial set of conservative deterministic failure margin (CDFM) hybrid fragilities with conservative assumptions. These fragility calculations contain various significant conservatisms and thus are not realistic. At the time of the peer review, it became apparent that more rigorous methods had not yet been employed to determine realistic median capacities and associated uncertainty parameters for the top contributors to SCDF and SLERF. The peer review noted that there was a plan in place to perform this additional analysis:

- "Refinement of fragility data for dominant contributors is to be performed. It is assumed fragility data will be updated as needed after this refinement."
- "More rigorous methods for calculation for fragility parameters may be applied to a small subset of SSC after initial S-PRA results are available and the dominant contributors to risk are identified. Methods will be per EPRI TR-103959 including numerical simulations and/or the separation of variables approach."

The peer review team was able to perform the peer review using the documentation received from the project team. Aspects of the review were difficult and required considerable research as well as question and answer sessions.

In summary, the fragility analysis generally meets the applicable requirements of the ASME/ANS RA-S Standard CODE CASE 1 [13]. However, the work is not complete for developing realistic fragilities for top contributors to SCDF and SLERF.

A.4.3 Seismic Plant Response

The Callaway SPR model integrates the site-specific hazard, fragilities and system-analysis and accident sequence aspects. The Callaway SPR model appropriately modified the Full Power Internal Event model to include seismically-induced initiating events including industry experience from EPRI guidance documents, seismically-induced initiating events caused by secondary hazards (internal and external floods), seismically-induced SSC failures and non-seismically induced unavailabilities and human actions.

The assessment of potential seismic initiating events considered a wide range of potential consequences defined by the internal events analysis, postulated external initiating events and industry guidance documents. The disposition of some initiating events as being encompassed by loss of offsite power may be non-conservative for lower accelerations where the generic capacity of non-safety equipment is lower than that assumed for LOSP.

The model accurately maps defined fragilities to the appropriate components in the PRA model and expands the internal events to address seismic impacts. However, the fragilities as mapped may overstate the correlation among relays and may result in links with non-correlated equipment causing broad impacts on components and systems. The impact of this conservatism is not fully resolved and may be addressed by future refinements. The conditional flooding and fire events postulated for seismic events are also mapped appropriately, but in some cases the mapping may not reflect the conditions of the seismic event. As an example, some fire piping faults are screened based on preaction design but that may not be appropriate for seismic events where spurious actuation of detectors is possible.

The S-PRA includes operator actions from the IEPRA, with the HEPs evaluated based on seismic specific challenges. The S-PRA model does not include any seismic-specific operator actions. The addition of operator actions to recover the impacts of relay chatter should be considered if the actions are feasible and impact risk significant relay chatter.

The model assembly incorporates the hazard, fragility and SSCs in a manner supporting quantification. The quantification process provides an approach for integrating this information in order to develop both CDF and LERF results and to propagate uncertainties. The uncertainty and sensitivity studies provide additional insights into the CDF and LERF characteristics as well as model realism and areas of the study that are sensitive to model assumptions.

A.5 Revision of S-PRA Model and Documentation

Following the peer review, the S-PRA model and documentation were updated to address all F&Os from the 2018 S-PRA Peer Review [14] except for Finding 25-19 (linked to SPR-B2) and Finding 25-12 (SPR-E6). In addition, CEC generated closure documentation for each of these F&Os. Subsequently, the updated S-PRA model and documentation were subjected to an independent assessment in March 2019 of the F&O closure.

A.6 Finding Closure by Independent Assessment and Focused Scope Peer Review

An independent assessment of CEC's resolution of open S-PRA F&Os was performed in March 2019 and is documented in PWROG-19011-P [15]. The process used for the independent assessment is outlined in Section X.1.3 (Close Out F&Os by Independent Assessment) of Appendix X to NEI 12-13 [24], which has been accepted by the NRC [44], with two conditions:

1. Use of New Methods: "A PRA method is new if it has not been reviewed by the NRC staff. There are two way new methods are considered accepted by the NRC staff: (1) they have been explicitly accepted by the NRC (i.e., they have been reviewed, and the acceptance has been documented in a safety evaluation, frequently-asked-questions, or other publicly available organizational endorsement), or (2) they have been implicitly accepted by the NRC (i.e., there has been no documented denial) in multiple risk-informed licensing applications. The NRC's treatment of a new PRA method for closure of F&Os is described in memorandum "U.S. Nuclear Regulatory Commission Staff Expectations for an Industry Facts and Observations Independent Assessment Process," dated May 1, 2017 (ADAMS Accession No. ML17121A271)."
2. Use of Appendix X in its entirety: "In order for the NRC to consider the F&Os closed so that they need not be provided in submissions of future risk-informed licensing applications, the licensee should adhere to the guidance in Appendix X in its entirety. Following the Appendix X guidance will reinforce the NRC staff's confidence the F&O closure process and potentially obviate the need for a more in-depth review."

The result of this independent assessment was intended to support future CEC licensee amendment request submittals, other regulatory interactions, risk-informed applications, and risk-informed decision-making. Finding resolution reviewed and determined to have been adequately addressed through this independent assessment are considered "closed" and no longer relevant to the current PRA model, and thus need to be carried forward nor discussed in such future activities.

A.6.1 Selection of Independent Assessment Team Members

The independent review team consisted of Mr. Kenneth Kiper, Dr. Marty McCann, Mr. Glenn Rix, Mr. Steve Eder, Dr. Ram Srinivasan, Mr. Rick Summit and Mr. Deepak Rao. Note that all the reviewers were members of the team that performed the 2018 S-PRA peer review [14]. The independent assessment team qualifications are discussed in Section A.3.

A.6.2 Host Utility Preparation

CEC provided the complete and relevant review material to the independent assessment team in advance to allow the reviewers to prepare and conduct a more efficient technical review. As input to the review, CEC provided the following documentation:

- Exact wording of each original F&O within scope of the independent assessment
- A summary description of how each F&O was dispositioned
- CEC self-assessment of whether the F&O closure involved an upgrade or a maintenance activity, based on the definition of upgrade vs. maintenance documented in the PRA Standard [32].
- Documents that were revised to resolve the F&Os

A.6.3 Offsite Review

All material generated in supported to the F&O closure activities performed by CEC were provided to the independent assessment team two weeks before the onsite review and consensus session. The review team started the review and familiarization of the documentation.

A.6.4 Onsite Review and Consensus

During the onsite review and consensus session, the team achieved the following for each reviewed F&O:

- Consensus on the status of the F&O (i.e., CLOSED, OPEN, or PARTIALLY CLOSED). This conclusion was reached through a review of the original basis and description of the F&O and on the technical work and documentation provided by CEC to resolve the issued identified in the F&O.
- Consensus on whether the activities performed to close the F&O are to be considered maintenance or upgrade, per the appropriate definition of the PRA Standard.
- If the F&O was associated with an SR that was originally judged as Not Met or Met at Capability Category I, upon confirming closure of the associated F&Os, the SR has been re-assessed to reach consensus on whether the intent of the SR is now Met at Capability Category II or higher.

A.6.5 Treatment of "New Methods"

All of the changes to the CEC S-PRA were classified as either PRA maintenance or PRA upgrade by the independent assessment team. Therefore, no new methods were identified during the independent assessment.

A.6.6 Use of Remote Reviewers

All team members participated on-site except for the two SHA reviewers (Dr. Marty McCann and Mr. Glenn Rix). This remote participation was supported with web and teleconference to the onsite review team. The SHA reviewers were needed only for a limited number of F&O reviews, and thus, these reviewers participated remotely.

A.6.7 Status of Findings at End of Independent Assessment

The following bullets summarize the independent assessment conclusions:

- SHA: 5 F&Os were closed, 0 remain open. Two SRs (SHA-C4 and SHA-H1) were reassessed and determined to be Met at Capability Category II. No new issues were identified for element SHA.
- SFR: 27 F&Os were closed, 0 remain open. SFR-D3, SFR-E3, SFR-E5 and SFR-F1 were reassessed and determined to be Met at capability category II. No new issues were identified for element SFR.
- SPR: 32 F&Os were closed, 2 remain open but were not in-scope for the IAT during this review. SPR-D2 and SPR-E5 were reassessed and determined to be Met at Capability Category II. No new issues were identified for element SPR.

A.6.8 Final Independent Assessment Report

A final report was provided at the end of the independent assessment, which documented the review and its conclusions [15]. This report includes the following information:

- Descriptions of the F&O independent assessment process.
- Description of the scope of the independent assessment.
- A summary of the review team's decisions for each finding within the scope of the review, along with the rationale for determination of adequacy of inadequacy for closure of each finding in relation to the affected portions of the associated SR.
- For each finding, assessment of whether the resolution was determined to be a PRA upgrade, maintenance update, or other, and the basis for that determination.

The final report included each of the independent assessment team members' resumes and summary of their experience as it applies to qualification guidelines of NEI guidance documents and the ASME/ANS PRA Standard.

This report will be retained by CEC in accordance with maintenance of their peer review and PRA recordkeeping practices, and it is available for review and audit.

A.6.9 Summary of Independent Assessment Team Conclusions

Of the F&Os reviewed, the independent assessment team concurred that all can be considered closed (with exception to the two F&Os that were not reviewed). As a result of the closure of the associated F&Os, SRs originally judged as Not met are now judged to be Met.

The independent assessment team recognized a significant amount of work invested in the resolution of the F&Os from the original peer review, including the generation of new fragility, additional sensitivity studies, improved documentation, and an additional walkdown for risk-significant seismic-induced flood and fire sources. The independent assessment team concluded that, as a result of the closure of the associated F&Os, the CEC S-PRA more realistically reflects the current seismic risk at the site.

A.6.10 Compliance of Independent Assessment with NRC Conditions

As indicated in Section A.6, the NRC's acceptance of the F&O closure process described in Appendix X to NEI 12-13 [24], as documented in [44], includes two conditions. The independent assessment of the finding closure for the CEC S-PRA satisfies the conditions as follows:

1. Use of New Methods: As indicated in Section A.6.5, new methods were not employed in the resolution of findings associated with the CEC Unit 1 S-PRA. Therefore, this condition does not apply.
2. Use of Appendix X in its Entirety: The finding closure process encompasses all of the elements of Appendix X

Therefore, the application of the Appendix X process to the closure of the findings identified during the CEC S-PRA peer review is in conformance with the NRC's requirements.

A.7 Summary of Technical Adequacy of the S-PRA for the 50.54(f) Response

The set of supporting requirements from the ASME/ANS PRA Standard that are identified in Tables 6-4 through 6-6 of the SPID [25] define the technical attributes of a PRA model required for a S-PRA used to respond to the 50.54(f) letter. The conclusions of the peer review discussed above and summarized in this submittal demonstrates that the CEC S-PRA model meets the expectations for PRA scope and technical adequacy as presented in RG 1.200, Revision 2 [23] as clarified in the SPID [25]. The main body of this report provides a description of the S-PRA methodology, including:

- Summary of the seismic hazard analysis (Section 3.0)
- Summary of the structures and fragilities analysis (Section 4.0)
- Summary of the seismic walkdowns performed (Section 4.2)
- Summary of the internal events at power PRA model on which the S-PRA is based, for CDF and LERF (Section 5.1)
- Summary of adaptations made in the internal events PRA model to produce the seismic PRA model and bases for the adaptations (Section 5.1).

Detailed archival information for the S-PRA consistent with the listing in Section 4.1 of RG 1.200 Rev. 2 [23] is available if required to facilitate the NRC staff's review of this submittal.

The CEC S-PRA reflects the as-built and as-operated plant as of the cutoff date for the S-PRA, 3/8/2019.

The peer review observations and conclusions noted in A.4, the F&O closure review discussion in Section A.6, demonstrate that the CEC S-PRA is technically adequate in all aspects for this submittal. Subsequent to the CEC peer review, the peer review findings have been appropriately dispositioned, and the S-PRA model has been updated to reflect these dispositions and further refine several fragility values. The results presented in this submittal reflect the updated model as of March 2019.

A.8 Summary of S-PRA Capability Relative to SPID Tables 6-4 through 6-6

The PWR Owners Group performed a full scope peer review of the CEC internal events PRA and internal flooding that forms the basis of the S-PRA to determine compliance with ASME/ANS PRA Standard [20] and [13]and Regulatory Guide 1.200 [23] in April 2019. This review documented findings for all supporting requirements (SRs) which failed to meet at least Capability Category II.

The PWR Owners Group peer review of the CEC S-PRA was conducted in June 2018. An independent assessment of S-PRA F&O closure was conducted in March 2019. The results of these peer reviews 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 teams expressed the opinion that the CEC S-PRA model is of good quality and integrates the seismic hazard, the seismic fragilities, and the systems-analysis aspects appropriately to quantify core damage frequency and large early release frequency. The general conclusion of the peer reviews was that the CEC S-PRA is judged to be suitable for risk-informed applications.

- Table A-1 provides a summary of the disposition of SRs judged by the peer reviews to be not met, or not meeting Capability Category II.
- Table A-2 provides a summary of the of potentially important sources of uncertainty in the CEC S-PRA

- Table A-3 provides a listing of plant modifications that have not yet been incorporated into the S-PRA.
- Table A-4 provides an assessment of the expected impact on the results of the CEC S-PRA of those SRs and peer review Findings that have not been fully addressed.

Table A-1: Disposition of SRs Assessed as Not Met or Not Met at Capability Category II

SR#	Assessed Capability Category	Associated Open Finding F&Os	Impact on S-PRA Results
SPR-B2	Not Met	25-19	See Table A-4
SPR-E6	Met at CCI	25-12	See Table A-4

A.9 Identification of Key Assumptions and Uncertainties Relevant to the S-PRA Results

The ASME/ANS PRA Standard [13] 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 [26] and EPRI 1016737 provide guidance on assessment of uncertainty for applications of a PRA. As described in NUREG-1855, sources of uncertainty include "parametric" uncertainties, "modeling" uncertainties and "completeness" (or scope and level of detail) uncertainties.

- Parametric uncertainty was addressed as part of the CEC S-PRA model quantification (See Section 5.7 of this submittal)
- Modeling uncertainties are considered in both the base internal events PRA and the S-PRA. 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 CEC S-PRA technical elements are noted in the S-PRA documentation that was subject to peer review, and a summary of important modeling assumptions is included in Section 5.3.2.
- Completeness uncertainty addresses scope and level of detail. Uncertainties associated with scope and level of detail are documented in the PRA but are only considered for their impact on a specific application. No specific issues of PRA completeness were identified in the S-PRA peer review.

A summary of potentially important sources of uncertainty in the CEC S-PRA is listed in Table A-2.

Table A-2: Potentially Important Sources of Uncertainty in CEC S-PRA

PRA Element	Summary of Treatment of Sources of Uncertainty per Peer Review	Potential Impact on S-PRA Results
Seismic Hazard	The CEC S-PRA 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 CEC site.	The seismic hazard reasonably reflects sources of uncertainty.
Seismic Fragilities	The CEC S-PRA peer review team noted that the seismic response analysis of buildings was supplemented by a number of sensitivity studies used to assess important modelling assumptions. These sensitivity studies included difference in response between single and multiple time histories; effect of spatial incoherency of ground motion in building response; and control building in-structure sensitivity study. These studies served to reinforce engineering judgments and assumptions implemented in the fragility analysis of SSCs.	Several of the sensitivity studies described in Section 5.7 of this report evaluate the impact of changes to fragilities on the S-PRA results as one means of assessing the impact of fragilities uncertainties on the S-PRA results.
Seismic PRA Model	The CEC S-PRA peer review team noted that parametric uncertainty analyses and sensitivity studies are performed and risk importances results are provided.	A characterization of the mean SCDF and SLERF is provided in Section 5.6 of this report. Several sources of model uncertainty are discussed in Section 5.7 along with sensitives performed to evaluate the impact of possible changes to address these.

A.10 Identification of Plant Changes Not Reflected in the S-PRA

The CEC S-PRA reflects the plant as of the cutoff date for the S-PRA, which was April 2020. Table A-3 lists significant plant changes subsequent to this date and provides a qualitative assessment of the likely impact of those changes on the S-PRA results and insights.

Table A-3: Summary of Significant Plant Changes Since S-PRA Cutoff Date

Description of Plant Change	Impact on S-PRA Results
MP 16-0021: MDV53 and MDV55 Replacement – This modification replaces two existing 345kV circuit breakers and four vertical disconnect switches in the Callaway Switchyard. In addition, 4 vertical disconnect switches: MDV71A, MDV71B, MDV75A, and MDV81A are replaced with Ameren standard Southern States type RDA-1 horizontal disconnect switches.	Impacts configuration risk model only, potential to no longer require the security diesel for switchyard breaker support. Minimal impact.
MP 17-0028: Switchyard Breakers Replacement MDV45, MDV51, and MDV85 – The following changes are part of the modification: • The ITE Switchyard breakers were at the end of their service lives and were replaced. The existing breakers utilize a two-pressure system with pneumatic mechanism to trip and close the break contacts. The new breakers are "puffer" type breakers that have a spring-operated mechanism to trip and close the break contacts. • Relays and cabling associated with breakers MDV41, MDV43, MDV45, MDV51, and MDV85 are at the end of their useful lives and will be replaced and indicators associated with the breakers updated. Existing 345kV vertical disconnect switch MDV55B is replace with Ameren standard Southern States RDA-1 type double-end break disconnect switches.	Impacts configuration risk model only, potential to no longer require the security diesel for switchyard breaker support. Minimal impact.
MP 09-0048: Cathodic Protection System Upgrade – This modification modifies the various components of the CPS: Rectifiers, Deepwell Anode Beds, and Shallow Anode Beds.	FPRA Documentation Only

Table A-3: Summary of Significant Plant Changes Since S-PRA Cutoff Date	
Description of Plant Change	**Impact on S-PRA Results**
MP 16-0041: Provide Power to New Maintenance Storage Building – This modification provides a new 300 Loop power feed and transformer to provide available power source to the new Maintenance Storage Building, the Steam Generator Storage Facility and the Quality Control Radiographic Bunker by installing new power feed from 289PG30401 to new switch 289PG32401, new transformer XPG324, and 480V power panel PPPG324. The existing feed from Construction Quality control Building and the Construction Fab Shop can no longer be used.	FPRA Documentation Only
Switchyard Physical and Cyber Security Modification 10/12/2 - This mod added more fencing, lighting, cameras and card readers to the switchyard.	FPRA Documentation Only
FLEX Portable Equipment Mods	Waiting on Industry resolution of data and HRA issues. Sensitivities assessed in S-PRA.
MP 08-0027: Main Turbine Controls Replacement – This modification replaced the existing GE Mark II turbine control system, the turbine over speed protection system, and the Low Load Valve control system. The existing control systems were analog and were replaced with digital technology to eliminate single-point vulnerabilities, address the limited support available for the Mark II system, address obsolescence, improve reliability, and reduce operator burden.	FPRA Only. Minimal Impact.
MP 16-0024: SGK05 Supplemental Cooling System – The modification includes the addition of two supplementary cooling systems (one to provide cooling from SGK05A and one to provide cooling from SGK05B) comprised of normally open fire dampers, circulating fans, and cross-train ductwork with isolation dampers. This modification also includes modifications to the 2032' elevation slab and concrete masonry unit walls of the Class 1E equipment rooms along with associated structural steel support for wall penetrations, fans, dampers, and ductwork.	Internal Flooding and Fire. Minimal Impact.

Table A-3: Summary of Significant Plant Changes Since S-PRA Cutoff Date	
Description of Plant Change	**Impact on S-PRA Results**
MP 16-0039: Upgrade EHC Room HVAC – The Main Turbine Controls Upgrade project MP 08-0027 will install new digital controls in the EHC room increased the heat load in the room. The new equipment requires that all room penetrations must have fire-rated seals. Two new 5-ton HVAC units were installed and existing windows units SGE13 and SGE20 were removed and fire dampers were installed in the new ducts.	FPRA only. Minimal impact.
MP 17-0006: ESW Water Hammer Mitigation Modification – This modification adds pressurized accumulators at the CACs, CRACs, and Component Cooling Water (CCW) heat exchangers that inject non-condensable gas at low pressure to reduce the pressure response present at those components.	All PRAs Documentation Only. Internal Flooding small pipe additions, minimal impact.
MP 18-0042: ESW Water Hammer Modification CRAC Tie Ins - - Install safety-related system branch locations during SGK04A/B Technical Specification Outages to facilitate the future Control Room Air Conditioning (CRAC) air accumulator injection points to mitigate the water hammer event on the supply and return piping of the "A" and "B" CRACs, SGK04A/B.	All PRAs Documentation Only. Internal Flooding small pipe additions, minimal impact.
17-0020, 17-0021, and 17-0037 Chemical Addition Pad and skid	FPRA Documentation Only
MP 17-0005: Instrument Tunnel Sump Pump – This modification replaces the instrument sump pump 7-5/16" impeller with a 8-3/8" impeller for PLF07A. A 7-7/8" impeller is acceptable for PLF07B.	FPRA minimal impact.
MP 14-0032: Install CCW Vent Valves – Installation of 7 vents on the CCW "B" train and 12 vents on the CCW "A" train, and modification of pipe EG-196-HBC-4" to address voiding concerns.	Modification completion delayed. Internal Flooding only. Minimal impact.
MP 10-0010: Modify Cooling Tower Basin Level Instrumentation – Installation of new level indication system for the Callaway Cooling Tower Basin and uninstall the obsolete capacitance style indication.	FPRA Only minimal impact.

Table A-4: Summary of Open Finding F&Os and Disposition Status

SR	F&O	Description	Basis	Suggested Resolution	Disposition
SPR-B2	25-19	The draft documentation provided to the peer team included a summary of the status of Findings against the IE-IF PRA and FPRA, either Open, Closed, or Pending Closed (But considered open for S-PRA). In addition, the summary indicated whether the resolution to the F&O had been integrated into the S-PRA. Additional information was provided regarding that integration, indicating that some of the open F&Os were documentation only and thus, would have no impact on the S-PRA. Other open F&Os were listed as "no anticipated impact on the S-PRA model." However, this SR is based on relevance to the S-PRA, not whether there would be a significant impact. Of the 181 F&Os, 89 were listed as Open or Pending Closed (Considered open for the S-PRA). While some of these are documentation only F&Os, the majority of the open F&Os are more than documentation.	A significant number of F&OS remain open and there is no clear basis for considering the potential resolution to be "not relevant to the S-PRA."	Provide official documentation of all open F&Os (with regard to the S-PRA model) and provide a justification for any open F&Os that are judged to be "not relevant to the S-PRA." Alternatively, close the F&Os and assess the impact of the changes on the S-PRA model, whether they are maintenance or upgrade changes. For any identified as upgrade, a focused peer review would be required to close the F&O.	A disposition of this F&O is provided in Section 5.7.6.
SPR-E6	25-12	Although many elements of the internal events LERF assessment is currently at least category II, the specific identified SRs are met at category I. Therefore the S-PRA is assessed at Cat I.	The current status of the CEC internal events LERF analysis identified that several of the required SRs were met at Cat I. Since the S-PRA utilizes the internal event PRA model these SRs carry forward and restrict the SR to Cat I.	Elevate the internal events assessment to meet Cat II and validate that the improvement is appropriate for the S-PRA.	A disposition of this F&O is provided in Section 5.7.6.