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PG&E Letter DCL-18-027

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

10 CFR 50.4
10 CFR 50.54(f)

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Power Plant Units 1 and 2
Seismic Probabilistic Risk Assessment for the Diablo Canyon Power Plant, Units 1 and 2 - Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1: Seismic of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident

References:

1. NRC Letter, "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," dated March 12, 2012 (ADAMS Accession Number ML12053A340)
2. NRC Letter, "Final Determination of Licensee Seismic Probabilistic Risk Assessments Under the Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendation 2.1 'Seismic' of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," dated October 27, 2015 (ADAMS Accession Number ML15194A015)
3. NRC Letter, "Diablo Canyon Power Plant, Unit Nos. 1 and 2 - Nuclear Regulatory Commission Response to Request for Extension of Seismic Probabilistic Risk Assessment Submittal Schedule," dated October 23, 2017 (ADAMS Accession Number ML17269A177)

Dear Commissioners and Staff:

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

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In Reference 2, the NRC indicated that a seismic probabilistic risk assessment is required for Diablo Canyon Power Plant (DCPP), and in Reference 3, the NRC indicated that the seismic probabilistic risk assessment should be submitted by April 30, 2018.

Enclosure 1 of this letter provides the Seismic Probabilistic Risk Assessment Summary Report for DCPP, Units 1 and 2, as requested in Reference 2.

PG&E is also submitting the DCPP Seismic Mitigating Strategies Assessment, but under a separate cover letter.

PG&E is making new and revised regulatory commitments (as defined by NEI 99-04) in this letter. The new and revised commitments are identified in Enclosure 2.

If you have any questions or require additional information, please contact Mr. Hossein Hamzehee at 805-545-4720.

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

Executed on April 24, 2018.

Sincerely,

James M. Welsch
Vice President, Nuclear Generation and Chief Nuclear Officer

mjrm/50702923
Enclosures
cc: Diablo Distribution
cc/enc:
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Pacific Gas and Electric Company

Diablo Canyon Power Plant
Units 1 and 2

License Nos. OL-DPR-80 and OL-DPR-82

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

April 2018

Diablo Canyon Power Plant
Seismic Probabilistic Risk Assessment
Summary Report

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1. Purpose and Objective

Following the accident at the Fukushima Dai-ichi nuclear power plant resulting from the March 11, 2011, Great Tohoku Earthquake and subsequent tsunami, the Nuclear Regulatory Commission (NRC) established a near-term task force (NTTF) to conduct a systematic review of NRC processes and regulations, and to determine if the agency should make additional improvements to its regulatory system. The NTTF developed a set of recommendations intended to clarify and strengthen the regulatory framework for protection against natural phenomena. Subsequently, the NRC issued a request for information letter pursuant to Title 10 of the Code of Federal Regulations, Section 50.54(f) on March 12, 2012 (the "50.54(f) letter)" [1], requesting information to assure that these recommendations are addressed by all U.S. nuclear power plants. The 50.54(f) letter requests that licensees and holders of construction permits, under 10 CFR Part 50, reevaluate the seismic hazards at their sites against present-day NRC requirements and guidance.

A comparison between the reevaluated seismic hazard and the design basis for the Diablo Canyon Power Plant (DCPP) has been performed, in accordance with the guidance in Electric Power Research Institute (EPRI) Technical Report No. 1025287, "Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic" [2], and previously submitted to the NRC in Pacific Gas and Electric (PG&E) Report, "Seismic Hazard and Screening Report - Diablo Canyon Power Plant, Units 1 and 2" [3]. That comparison concluded that the ground motion response spectrum (GMRS), which was developed based on the reevaluated seismic hazard, exceeds the design basis seismic response spectrum in the 1 to 10 Hz range, and a seismic risk assessment is required. An update to the existing DCPP seismic probabilistic risk assessment (SPRA) has been prepared in response to the 50.54(f) letter, specifically Item (8) in Enclosure 1 of the 50.54(f) letter.

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

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

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

2. Information Provided in This Report

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

(1) The list of the significant contributors to the seismic core damage frequency (SCDF), including importance measures (e.g., risk achievement worth (RAW) and Fussell-Vesely (F-V))

(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 walk-downs 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 probabilistic risk assessment (PRA) model to produce the SPRA model and their motivation

  vi. Assumptions about containment performance

(3) Description of the process used to ensure that the SPRA is technically adequate, including the dates and findings of any peer reviews

(4) Identified plant-specific vulnerabilities and actions that are planned or taken

Note that Enclosure 1, Paragraphs 1 through 6 of the 50.54(f) Letter, regarding the seismic hazard evaluation reporting, also apply, but have been satisfied through the previously submitted DCPP Seismic Hazard Submittals ([3] and PG&E Report, "Response to NRC Request for Additional Information dated October 1, 2015 and November 12, 2015 Regarding DCPP Seismic Hazard and Screening Report" [4], as accepted by the NRC Letter, "Diablo Canyon Power Plant, Unit Nos. 1 and 2 - Staff Assessment of Information Provided under Title 10 of the Code of Federal Regulations, Part 50, Section 50.54(f), Seismic Hazard Reevaluation for Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident" [5]). Further, Enclosure 1, Paragraph 9 of the 50.54(f) letter requests information on the spent fuel pool (SFP). This information was submitted separately in PG&E Report, "Site-Specific Spent Fuel Pool Criteria for the Diablo Canyon Power Plant" [6]).

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

The SPID [2] defines the principal parts of an SPRA, and the DCPP SPRA has been updated and documented in accordance with the SPID. The main elements of the DCPP SPRA, updated in response to the 50.54(f) Seismic letter correspond to those described in Section 6.1.1 of the SPID:

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

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

The DCPP SPRA and associated documentation has been peer reviewed against the current American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard (RA-Sb-2013) [7] in accordance with the process defined in Nuclear Energy Institute (NEI) 12-13 [8], as documented in the report on the peer review of the DCPP SPRA [9] and the report on the independent assessment of the closure of the DCPP SPRA Facts and Observations (F&Os) [39]. The DCPP SPRA, complete SPRA documentation, and details of the peer review will be retained by PG&E in accordance with the DCPP PRA record keeping practices and are available for NRC review. As discussed in Appendix A, the DCPP SPRA documentation meets the requirements of the ASME/ANS PRA Standard [7] (see Section 6.8 of the SPID [2]). This submittal provides a summary of the SPRA update, results and insights, and the peer review process and results, sufficient to meet the 50.54(f) information request in a manner intended to enable the NRC to understand and determine the validity of key input data and calculation models used, and to assess the sensitivity of the results to key aspects of the analysis.

The content of this report is organized as follows:

- Section 3 provides information related to the DCPP seismic hazard analysis.

- Section 4 provides information related to the determination of seismic fragilities for DCPP SSCs included in the seismic plant response.
- Section 5 provides information regarding the plant seismic response model (seismic accident sequence model) and the quantification of results.
- Section 6 summarizes the results and conclusions of the SPRA, including identified plant seismic issues and actions taken or planned.
- Section 7 provides references.
- Section 8 provides a list of acronyms used.
- Appendix A provides an assessment of SPRA Technical Adequacy for Response to the 50.54(f) Letter, including a summary of DCPP SPRA peer review.

Table 2-1 - Cross-Reference for 50.54(f) Enclosure 1 SPRA Reporting		
50.54(f) Letter Reporting Item	**Description**	**Location in this Report**
1	List of the significant contributors to SCDF for each seismic acceleration bin, including importance measures.	Section 5.
2	Summary of the methodologies used to estimate the SCDF and SLERF.	Sections 3, 4, and 5.
2i	Methodologies used to quantify the seismic fragilities of SSCs, together with key assumptions.	Section 4.
2ii	SSC fragility values with reference to the method of seismic qualification, the dominant failure mode(s), and the source of information.	Tables 5.4-2 and 5.5-2 provide the fragility parameters (A_m, β_r, and β_u), failure mode information, and method of determining the fragilities for the top risk significant SSCs based on F-V importance. Fragility calculations are documented in component specific calculation.
2iii	Seismic fragility parameters.	Tables 5.4-2 and 5.5-2 provide the fragility parameters (A_m, β_r, and β_u) for the top risk significant SSCs based on F-V importance.
2iv	Important findings from plant walk-downs and any corrective actions taken.	Section 4.2 addresses the walk-downs and walk-down insights.
2v	Process used in the seismic plant response analysis and quantification, including specific adaptations made in the internal events PRA model to produce the SPRA model and their motivation.	Section 5.1 provides this information.
2vi	Assumptions about containment performance.	Sections 4.3 and 5.3.2 address the Containment and related SSC performance.

Table 2-1 - Cross-Reference for 50.54(f) Enclosure 1 SPRA Reporting		
50.54(f) Letter Reporting Item	**Description**	**Location in this Report**
3	Description of the process used to ensure that the SPRA is technically adequate, including the dates and findings of any peer reviews.	Appendix A describes the assessment of SPRA technical adequacy for the 50.54(f) submittal and results of the SPRA peer review.
4	Identified plant-specific vulnerabilities and actions that are planned or taken.	Section 6 addresses the plant-specific vulnerabilities. Although no seismic hazard-related vulnerabilities were identified, the deficiency in the installation ducts associated with the 480V switchgear room ventilation system, identified during the walk-downs performed in support of the SPRA update, will be addressed through the design and implementation of modifications.

Table 2-2 - Cross-Reference for Additional SPID Section 6.8 SPRA Reporting	
SPID Section 6.8 Item [(1)] Description	**Location in this Report**
A report should be submitted to the NRC summarizing the SPRA inputs, methods, and results.	The entirety of this report addresses this.
The level of detail needed in the submittal should be sufficient to enable NRC to understand and determine the validity of all input data and calculation models used.	The entirety of this report addresses this.
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.	The entirety of this report addresses this. The results of the sensitivities are discussed in the following sections: • 5.7 (SPRA model sensitivities) • 4.4.1 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.	The entirety of this report addresses this.
It is not necessary to submit all of the SPRA documentation for such an NRC review. Relevant documentation should be cited in the submittal, and be available for NRC review in easily retrievable form.	The entirety of this report addresses this. This report summarizes important information from the SPRA, with detailed information in lower tier documentation which is referenced throughout this report.
Documentation criteria for a SPRA are identified throughout the ASME/ANS PRA Standard [7]. Utilities are expected to retain that documentation consistent with the Standard.	This is an expectation relative to documentation of the SPRA that the utility retains to support application of the SPRA to risk-informed plant decision-making.

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

3. DCPP 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.

DCPP is located on the central California coast, approximately 19 kilometers (km) (12 miles (mi.)) west of San Luis Obispo, California. The plant is on the southwestern margin of the Irish Hills, an area of moderate relief bordered by Morro Bay on the north, San Luis Obispo Bay on the south, and Los Osos Valley on the east.

Bedrock in DCPP's vicinity includes highly deformed Mesozoic and Cenozoic sedimentary and volcanic rocks. Foundations of principal plant buildings are founded directly on volcaniclastic rocks of the Miocene Obispo Formation (Fm.), as described in Section 2.1 of the DCPP Seismic Hazard and Screening Report [3]. Directly adjacent to the foundation area, this volcaniclastic sub-unit is locally unconformably overlain by Quaternary surficial units including alluvial fan sediments and marine terrace deposits. Additionally, engineered fill underlies portions of the roadways and infrastructure at the DCPP site.

The site profile was developed using a combination of data collected from borings, as well as three-dimensional (3D) tomography from active-source seismic data collected in 2011 and 2012, as described in PG&E Report No. GEO.DCPP.TR.16.01 [41]. Additional details regarding the site characterization are provided in [3] and the PG&E response to NRC request for additional information regarding the DCPP NTTF 2.1 Seismic Hazard and Screening Report [4]. The time averaged shear wave velocity to a depth of 30 meters (m) (V_{s30}) at the control point (see Section 3.1.1 for definition of the control point) is 968 m/second (m/s) [41].

3.1 Seismic Hazard Analysis

This section discusses the seismic hazard analysis methodology, presents the final seismic hazard results used in the SPRA, and discusses important assumptions and important sources of uncertainty. The seismic hazard analysis is in conformance with the requirements of Section 2 of the SPID [2].

The seismic hazard analysis determines the annual frequency of exceedance for selected ground motion parameters. The analysis involves use of earthquake source models (see Section 2.2 of the SPID [2]), ground motion attenuation models (see Section 2.3 of the SPID [2]), characterization of the site response (e.g., soil column) (see Section 2.4 of the SPID [2]), and accounts for the uncertainties and randomness of these parameters to arrive at the site seismic hazard. Detailed information regarding the DCPP site hazard was provided to NRC in the seismic hazard information submitted to the NRC in response to the NTTF 2.1 Seismic information request [3, 4].

3.1.1 Seismic Hazard Analysis Methodology

For the DCPP SPRA, the following method was used.

The details of the seismic hazard analysis are reported in [3,4]. The approach used included site amplification calculated through empirical and analytical methods to define the hazard curves at the control point. The control point elevation is defined as finished grade level, which corresponds to an elevation of 26 m (85 feet (ft.)) mean sea level, consistent with the DCPP Updated Final Safety Analysis Report (UFSAR) [21] (see Section 2.4 of the SPID [2]). The control point hazard curves were used to develop uniform hazard response spectra (UHRS) and the GMRS. Table 6-1 and Figure 6-1 in [4] provide the mean UHRS for 10^{-4} and 10^{-5} and the GMRS accelerations for a range of spectral frequencies.

Foundation input response spectra (FIRS) were developed for use in soil-structure interaction (SSI) calculations, as reported in [41]. Several different approaches were used to calculate the FIRS, and each of the approaches is described as follows.

For the power-block structures, structure-specific velocity profiles were developed based on the one-dimensional (1D) profiles extracted from the 3D velocity model. The range of the velocity profiles represents the variability of the velocity profile under the footprint structure for the best 3D model. An additional 15 percent uncertainty was added to account for uncertainty in the estimated velocities from the tomographic inversion. The nonlinear material properties and the kappa values of the rock motion are modeled using the same logic tree structure that was used for the control point GMRS. For each base profile, there are 3 base case velocity models, 3 kappa values, and 3 non-linear materials. When combined with the 30 randomized soil profiles, this leads to 270 soil profiles for each of the 3 base cases and a total of 810 profiles for each structure.

Each of the power-block structures (containment structures, auxiliary building, and turbine building) is founded at a different elevation, but calculation of the three FIRS follow the same approach. Consistent with the DCPP GMRS, the FIRS were developed using a combination of empirical and analytical approaches while also accounting for the structure-specific site profile and the foundation depth using the following process. First, the GMRS was recomputed using only the analytical site response. Next, the analytical site response was used to compute the outcrop FIRS at the foundation elevation. Finally, the hybrid GMRS (empirical and analytical site response) was then modified by the ratio of the analytical FIRS and GMRS to compute the hybrid FIRS. Using this process, the horizontal hybrid FIRS for the three structures were computed, and are tabulated in Table 3-2.

The intake structure is embedded in rock near sea level and the outdoor water storage tanks (OWSTs) are founded on rock east of the auxiliary building. For the intake structure and OWSTs, the horizontal FIRS were computed using a simplified approach in which the horizontal GMRS is adjusted based on the difference in the V_{s30} for the control point and the V_{s30}, measured from the foundation depth, for these other structures. The V_{s30} scaling in the Next Generation Attenuation Relationships - Western United States, Second Generation ground motion prediction equation (GMPE) is used to scale the GMRS. The details of the calculation are provided in Section 9.2 of [41].

The 230 kV switchyard is located inland of DCPP, on about 80 ft. of fill. The horizontal FIRS for the 230 kV switchyard was developed using site-specific site response analysis, which is documented in Section 9.1 of [41].

The vertical FIRS were computed by applying the V/H ratio model from Gülerce and Abrahamson (2011) [46]. Given that the V_{s30} scaling for rock sites in the V/H model from [46] is not well constrained, the V/H ratio is simply applied to each of the structures without structure-specific adjustments considering location-specific velocity profiles (i.e., V_{s30}). Analysis of the 230 kV switchyard does not require vertical FIRS, and thus they were not developed as part of this effort.

Using the horizontal and vertical FIRS as targets, suites of 30 spectrally matched time histories were developed as described in [41] and summarized as follows. An initial collection of 100 3-component time histories were selected based on the magnitude, distances, V_{s30}, and usable frequency range. Checking the spectral shape of the vertical motion and requiring the cross correlation to be less than 0.3 reduced the initial 100 sets to 40 candidate sets. This set was then reduced to 30 3-component time histories considering the duration distribution. Each of the horizontal time histories was then spectrally matched to individual target spectra based on the FIRS and modified to accommodate peak-to-trough variability not included in the GMPEs. The vertical component was fit to the vertical FIRS with no modification for peak-to-trough variability. This process was repeated independently for each of the FIRS.

3.1.2 Seismic Hazard Analysis Technical Adequacy

The DCPP SPRA hazard methodology and analysis associated with the horizontal GMRS were submitted to the NRC as part of [3, 4], and found to be technically acceptable by the NRC for application to the DCPP SPRA, as documented in their Staff Assessment [5]. No changes to the hazard were made subsequent to these submittals.

The DCPP hazard analysis was also subjected to an independent peer review against the pertinent requirements in the ASME/ANS PRA Standard [7]. The peer review assessment, and subsequent disposition of peer review findings through an independent assessment, is described in Appendix A, and establishes that the DCPP seismic hazards analysis is suitable for the SPRA application.

3.1.3 Seismic Hazard Analysis Results and Insights

This section provides the final seismic hazard results used in the DCPP SPRA.

The ground motion characterization model for DCPP is contained in the Southwestern United States (SWUS) Ground Motion Characterization Report [20] and provides the full description of the development, technical justification, and evaluation of the ground motion characterization effort. Based on the parallel timing of the seismic source characterization and ground motion characterization model development, the respective hazard feedback analyses were based on either a simplified, but representative seismic source characterization or the ground motion characterization model shown in Chapter 14 of the ground motion characterization and seismic source characterization reports [19, 20]. An abbreviated discussion of the results and insights are provided below.

The rock (site-specific site effects not included) hazard curves for the 5 Hz spectral acceleration at DCPP is shown in Figure 3-1, and indicates that the seismic sources which are closest to the DCPP site location control the total hazard. The fractional contribution of the different seismic sources to the total mean hazard for DCPP for 5 Hz spectral acceleration is shown in Figure 3-2. The individual curves for the Hosgri fault (solid red line), Shoreline fault (solid green line), Los Osos fault (solid purple line), San Luis Bay fault (solid blue line), the combination of these four local faults (dashed green line), regional faults (dotted black line) and background source zone (dashed blue line) are plotted. The combined local faults contribute more than 90 percent of the total hazard for ground motions greater than about 0.3 g.

Sensitivity of alternatives in the ground motion characterization are identified using tornado plots. For this project, the tornado plots show the ratio of the ground motion from the isolated branch to the ground motion for the mean hazard obtained with the full logic tree weights (called the base case). The tornado plot for 5 Hz spectral acceleration is shown in Figure 3-3 and indicates that the largest uncertainty is for the selection of the representative common-form models, which reflect uncertainty in the median ground motion model, followed by the total uncertainty models.

The uncertainties in the ground motion characterization models are the main contribution to the hazard fractiles. In particular, the uncertainty in the median ground motion model is the dominant contributor to the total uncertainty range. For the DCPP site, the uncertainty in the SSC model does not lead to a large uncertainty in the hazard for four reasons: (1) the slip rates of the faults are relatively well constrained, (2) the controlling scenarios are large magnitude earthquakes at short distances and due to the magnitude saturation in the

GMPEs, the ground motion at short distances is not sensitive to the earthquake magnitude, (3) the seismicity rate is high enough so that the controlling epsilon of the ground motion, required to reach hazard levels of 10^{-4} and 10^{-5}, are between 1 and 2.5; for high epsilons, the hazard curves are steep and the uncertainty is controlled by the GMPE uncertainty, and (4) with 4 key sources within 10 km of the site, changing the parameters for one source does not change the hazard as much as if there was only one controlling source.

The mean and fractiles of the DCPP control point horizontal hazard are shown in Figure 3-4. Information on the vertical hazard is discussed in Section 3.1.4.

Additionally, non-vibratory hazards were evaluated consistent with Section 4.1.1 of the ASME/ANS PRA Standard [7]. Based on the results of screening analyses [60, 61], detailed calculations were prepared to address seismic slope stability [62], tsunami [63], and fault rupture [64] hazards. The results of these calculations were used to specify the hazard input to the SPRA.

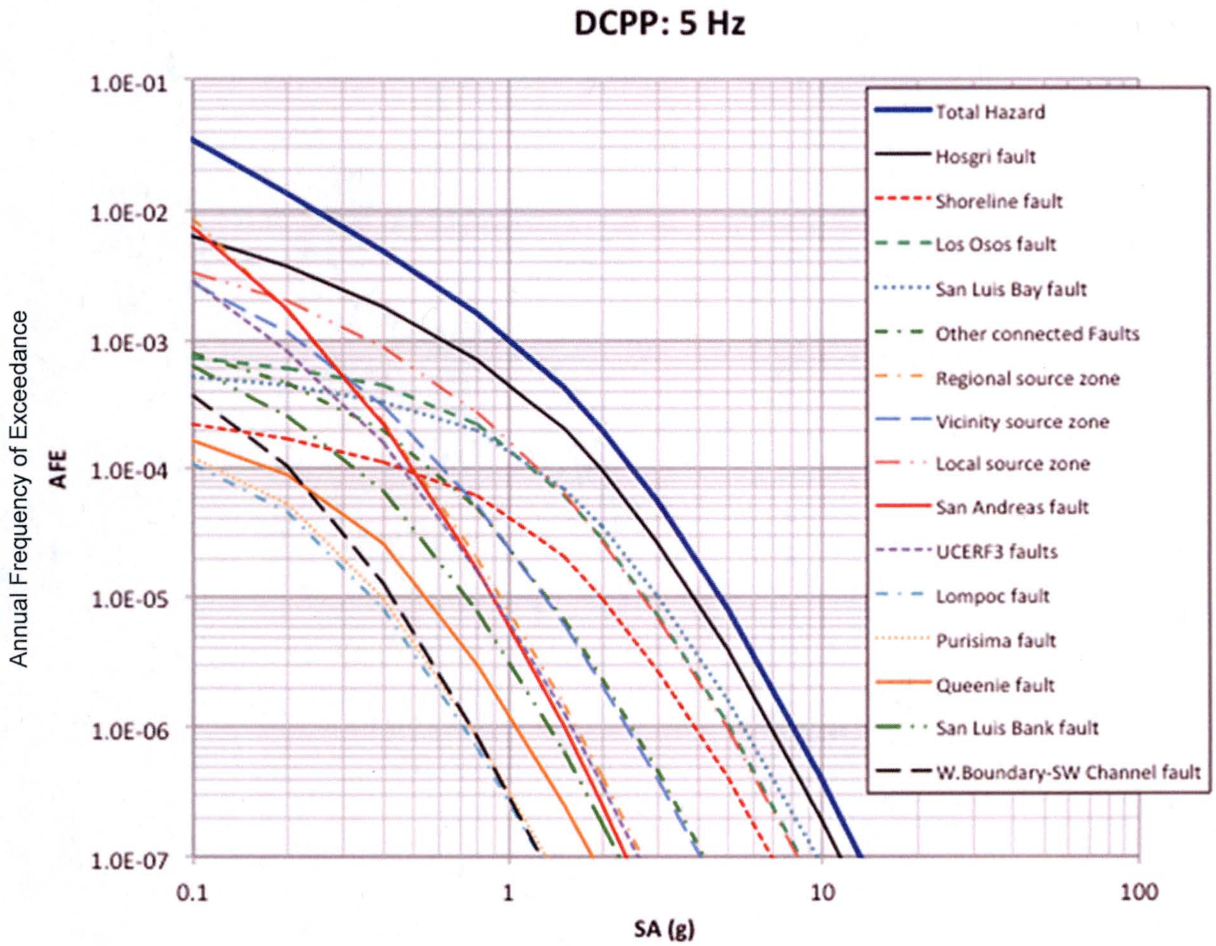

Figure 3-1 – Reference Rock Hazard by Source for 5 Hz Spectral Acceleration ([41], Figure 2-1d) **(AFE = annual frequency of exceedance)**

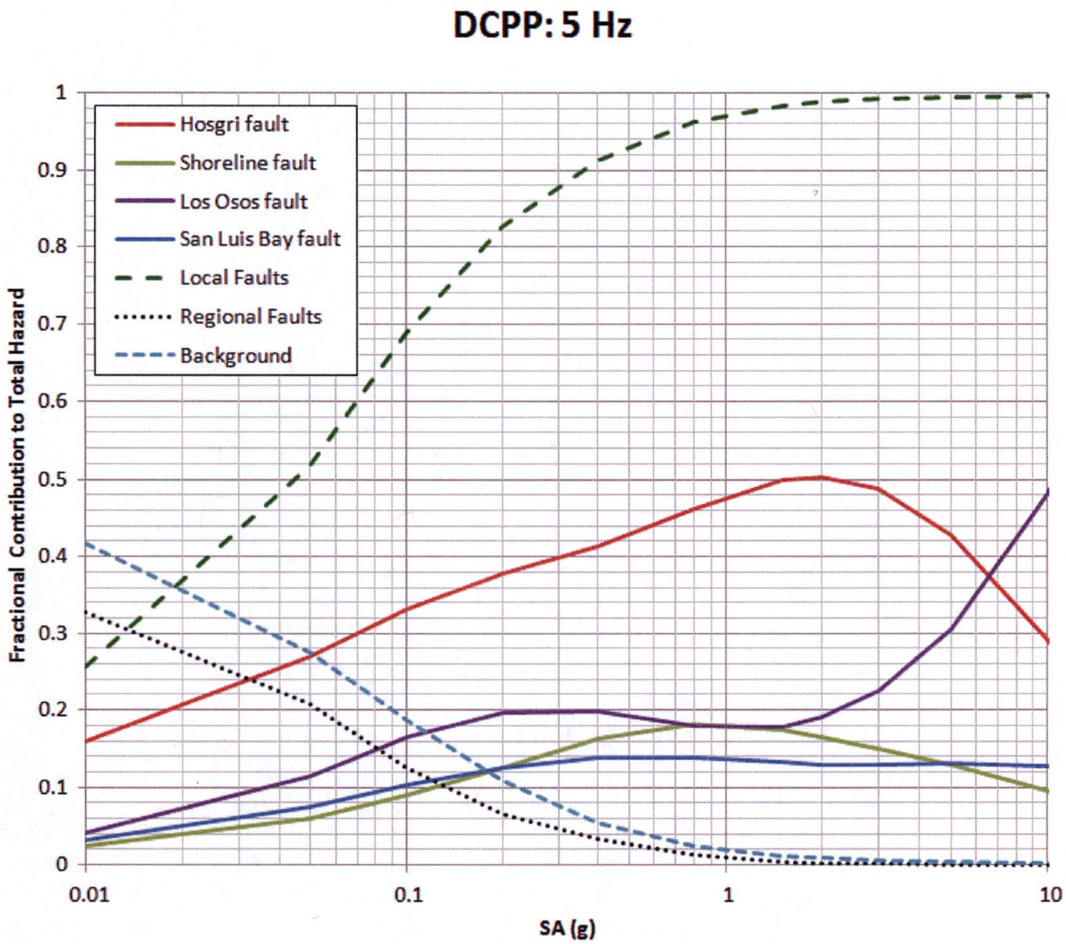

Figure 3-2 – Deaggregation by Source as a Function of Ground Motions for 5 Hz Spectral Acceleration ([20], Figure 14.2-1)

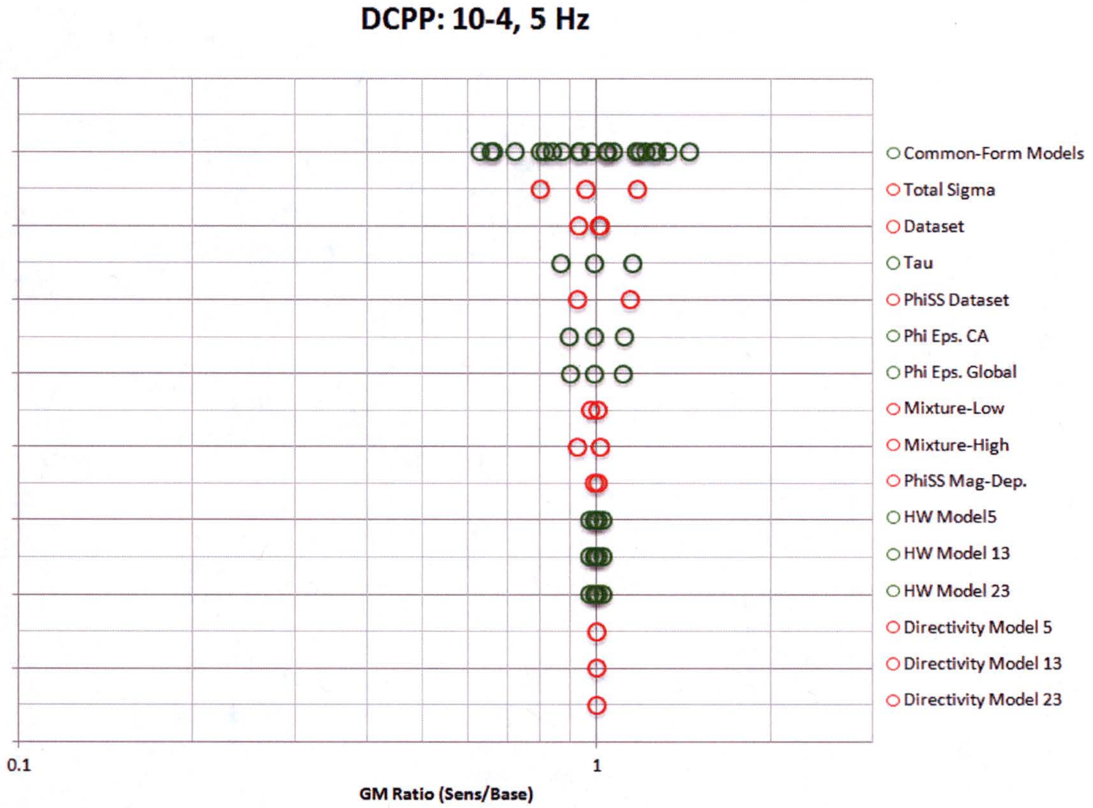

Figure 3-3 – Summary Tornado Plot for DCPP for 5 Hz Spectral Acceleration at the 10^{-4} Hazard Level ([20], Figure 14.2-9a)

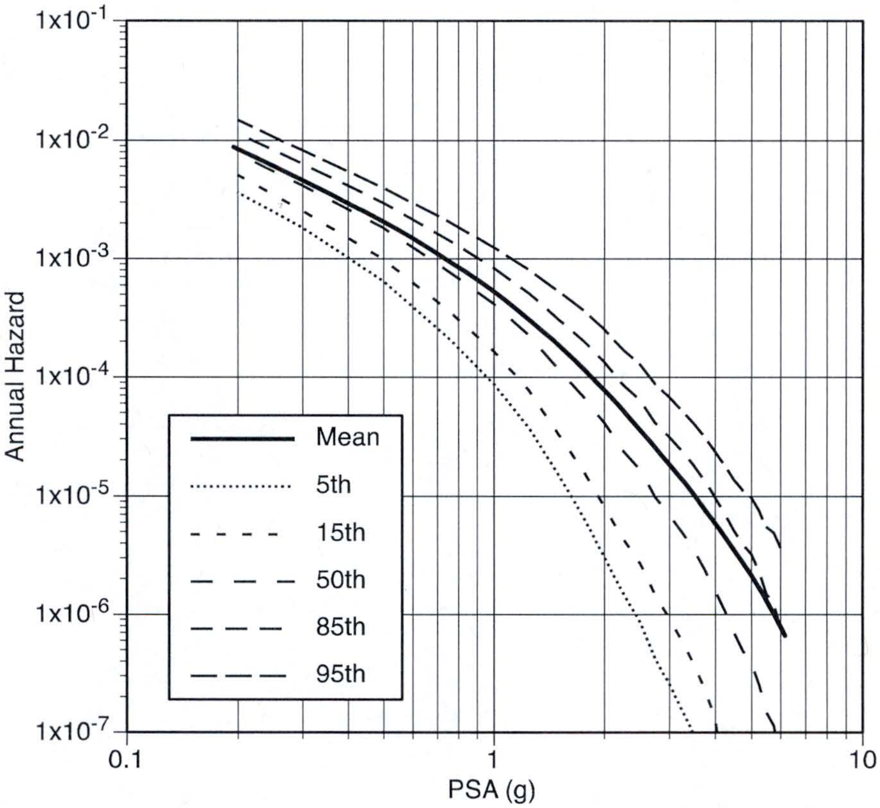

Figure 3-4 – 5 Hz Control Point Mean and Fractiles Horizontal Hazard (PG&E Calculation No. GEO.DCPP.15.02 [47], Figure 7.3-1)

3.1.4 Horizontal and Vertical Ground Motion Response Spectra

This section provides the control point horizontal GMRS, as well as the horizontal and vertical FIRS for the containment structures, auxiliary building, and turbine building.

The control point UHRS and GMRS are tabulated in Table 3-1 and shown in Figure 3-1.

For the power-block structures, the horizontal FIRS are tabulated in Table 3-2, and the associated V/H ratio and vertical FIRS are tabulated in Table 3-3. The horizontal and vertical FIRS are shown in Figure 3-2, and the associated V/H ratio is shown in Figure 3-3.

Table 3-1 - Horizontal Component 5 Hz Control Point Mean and Fractiles Hazard ([47], Table 7.2-1)

Frequency (Hz)	UHRS 1E-4 (g)	UHRS 1E-5 (g)	GMRS (g)
100.00	0.856	1.621	0.856
50.00	0.878	1.665	0.879
39.84	0.902	1.720	0.907

Table 3-1 - Horizontal Component 5 Hz Control Point Mean and Fractiles Hazard ([47], Table 7.2-1)			
Frequency (Hz)	**UHRS 1E-4 (g)**	**UHRS 1E-5 (g)**	**GMRS (g)**
33.33	0.912	1.737	0.916
25.13	0.994	1.905	1.004
20.00	1.088	2.075	1.094
16.58	1.217	2.322	1.224
13.33	1.437	2.718	1.437
11.75	1.489	2.822	1.490
10.00	1.509	2.863	1.511
8.32	1.583	3.002	1.585
6.67	1.723	3.277	1.729
5.89	1.762	3.368	1.775
5.00	1.850	3.528	1.861
4.47	1.817	3.511	1.847
4.00	1.842	3.562	1.873
3.71	1.755	3.401	1.788
3.33	1.701	3.305	1.736
2.82	1.825	3.652	1.907
2.50	1.913	3.899	2.029
2.24	1.816	3.697	1.924
2.00	1.716	3.460	1.804
1.66	1.507	3.154	1.633
1.33	1.283	2.753	1.418
1.17	1.074	2.299	1.185
1.00	0.859	1.844	0.950
0.79	0.626	1.398	0.714
0.67	0.499	1.122	0.572
0.58	0.410	0.928	0.473
0.50	0.337	0.773	0.393
0.40	0.243	0.549	0.280
0.33	0.195	0.434	0.222

Table 3-2 – Horizontal Component FIRS at 5% Damping ([65], Table 7.2-1)			
	5%-Damped Horizontal FIRS (g)		
Frequency (Hz)	**Containment Structures**	**Turbine Building**	**Auxiliary Building**
100.00	0.774	0.875	0.737
50.00	0.797	0.895	0.756
39.84	0.815	0.912	0.776
33.33	0.809	0.910	0.775
25.13	0.876	0.989	0.831

Table 3-2 – Horizontal Component FIRS at 5% Damping ([65], Table 7.2-1)			
Frequency (Hz)	5%-Damped Horizontal FIRS (g)		
	Containment Structures	Turbine Building	Auxiliary Building
20.00	0.949	1.079	0.880
16.58	1.042	1.210	0.980
13.33	1.251	1.434	1.187
11.75	1.269	1.497	1.197
10.00	1.267	1.516	1.163
8.32	1.351	1.607	1.221
6.67	1.527	1.779	1.425
5.89	1.612	1.838	1.526
5.00	1.703	1.930	1.646
4.47	1.733	1.974	1.660
4.00	1.723	1.930	1.659
3.71	1.657	1.865	1.584
3.33	1.622	1.843	1.574
2.82	1.771	2.010	1.738
2.50	1.932	2.169	1.869
2.24	1.810	1.997	1.771
2.00	1.708	1.812	1.689
1.66	1.569	1.692	1.558
1.33	1.370	1.447	1.352
1.17	1.158	1.209	1.128
1.00	0.931	0.981	0.910
0.79	0.693	0.719	0.692
0.67	0.561	0.587	0.555
0.58	0.469	0.486	0.463
0.50	0.399	0.418	0.393
0.40	0.279	0.292	0.277
0.33	0.215	0.222	0.213

Table 3-3 – Vertical Component FIRS at 5% Damping ([65], Table 7.2-2)				
Frequency (Hz)	V/H Ratio	5%-Damped Vertical FIRS (g)		
		Containment Structures	Turbine Building	Auxiliary Building
100.00	0.803	0.622	0.703	0.592
50.00	0.803	0.640	0.719	0.607
39.84	0.850	0.693	0.775	0.660
33.33	0.911	0.737	0.829	0.706
25.13	1.002	0.878	0.991	0.833
20.00	1.083	1.028	1.169	0.953

Table 3-3 – Vertical Component FIRS at 5% Damping ([65], Table 7.2-2)				
Frequency (Hz)	V/H Ratio	5%-Damped Vertical FIRS (g)		
		Containment Structures	Turbine Building	Auxiliary Building
16.58	1.090	1.136	1.319	1.068
13.33	0.998	1.248	1.431	1.185
11.75	0.918	1.165	1.374	1.099
10.00	0.823	1.043	1.248	0.957
8.32	0.726	0.981	1.167	0.886
6.67	0.651	0.994	1.158	0.928
5.89	0.617	0.995	1.134	0.942
5.00	0.580	0.988	1.119	0.955
4.47	0.571	0.990	1.127	0.948
4.00	0.563	0.970	1.087	0.934
3.71	0.561	0.930	1.046	0.889
3.33	0.561	0.910	1.034	0.883
2.82	0.563	0.997	1.132	0.978
2.50	0.561	1.084	1.217	1.049
2.24	0.559	1.012	1.116	0.990
2.00	0.556	0.950	1.007	0.939
1.66	0.574	0.901	0.971	0.894
1.33	0.609	0.834	0.881	0.823
1.17	0.630	0.730	0.762	0.711
1.00	0.630	0.587	0.618	0.573
0.79	0.630	0.437	0.453	0.436
0.67	0.630	0.353	0.370	0.350
0.58	0.630	0.295	0.306	0.292
0.50	0.630	0.251	0.263	0.248
0.40	0.630	0.176	0.184	0.175
0.33	0.630	0.135	0.140	0.134

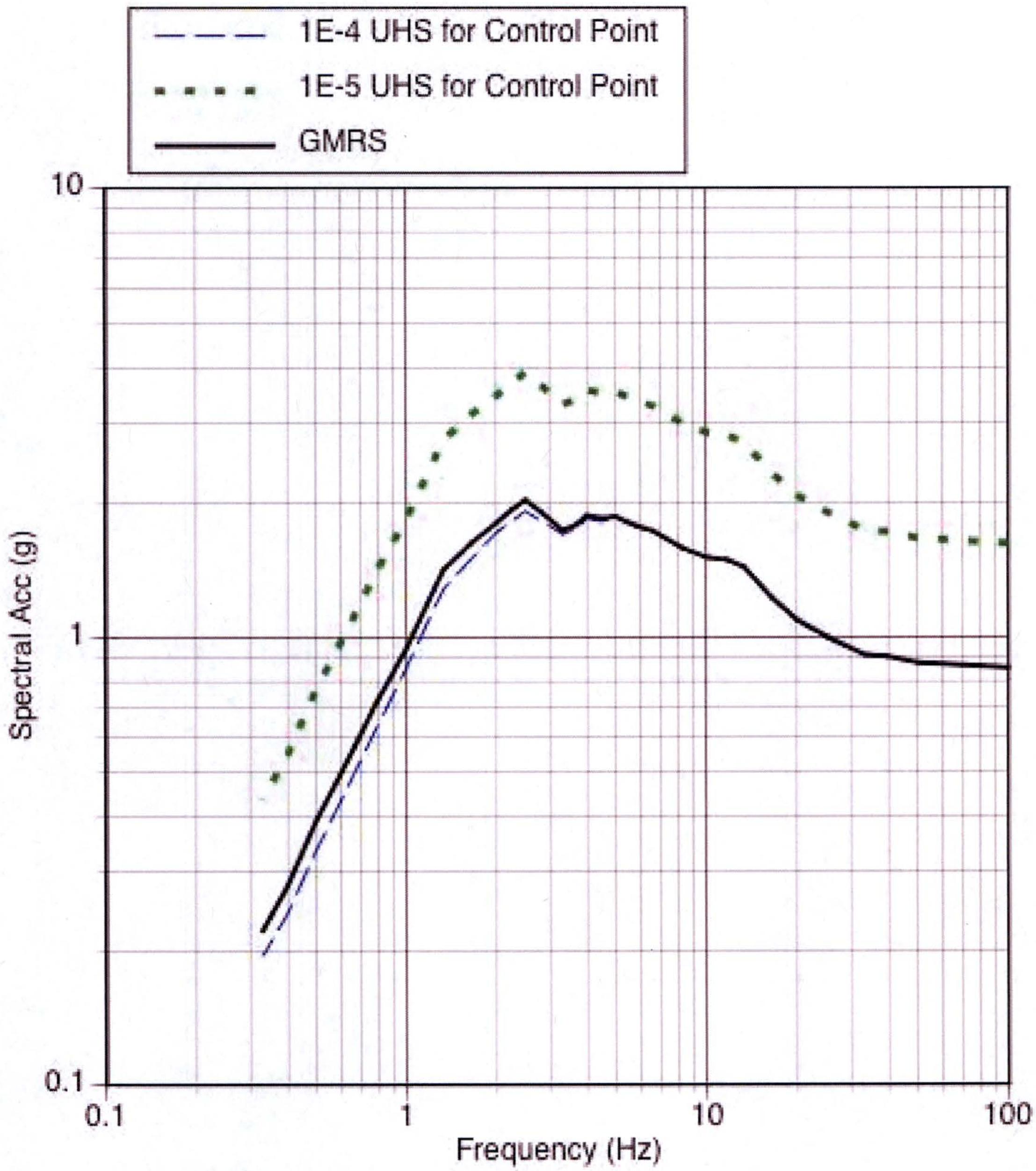

Figure 3-1 – Horizontal Component UHRS and GMRS for the Control Point ([4], Figure 6-1)

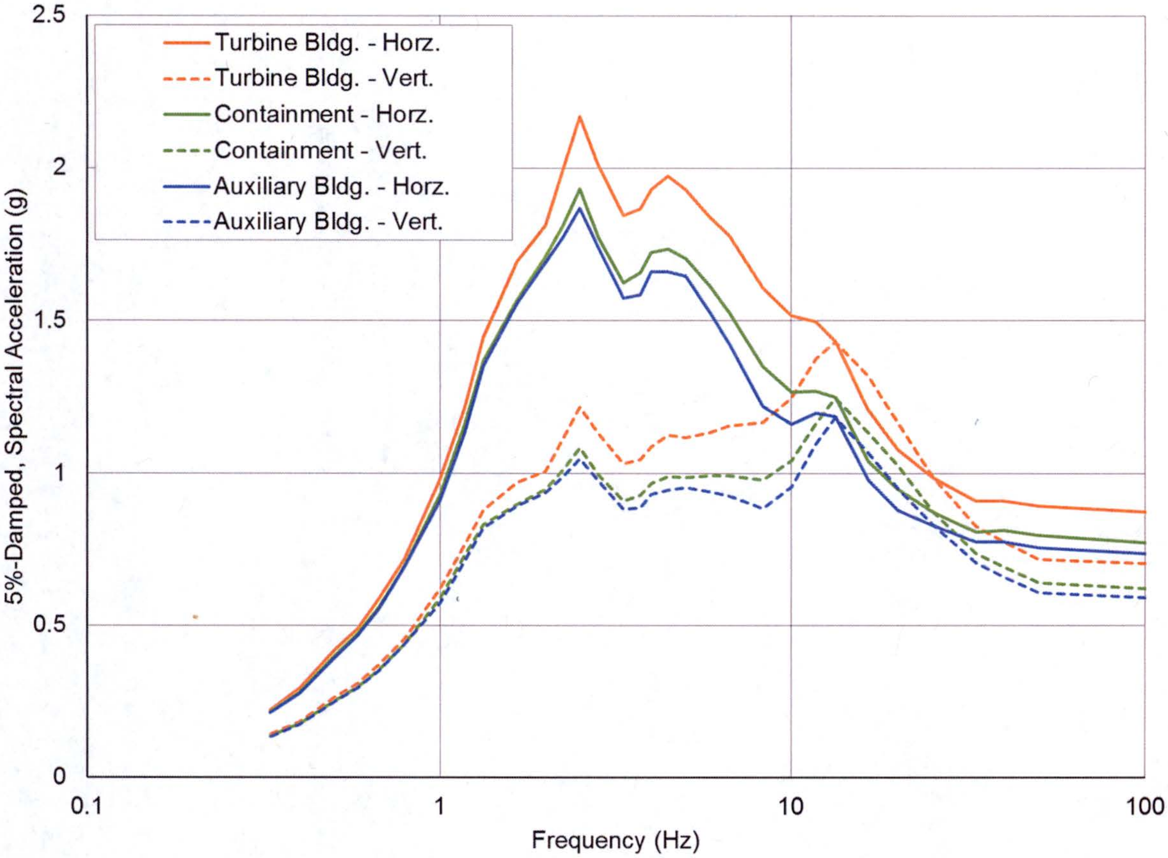

Figure 3-2 – Horizontal and Vertical FIRS for the Power-Block Structures
([41], Figure 4-12)

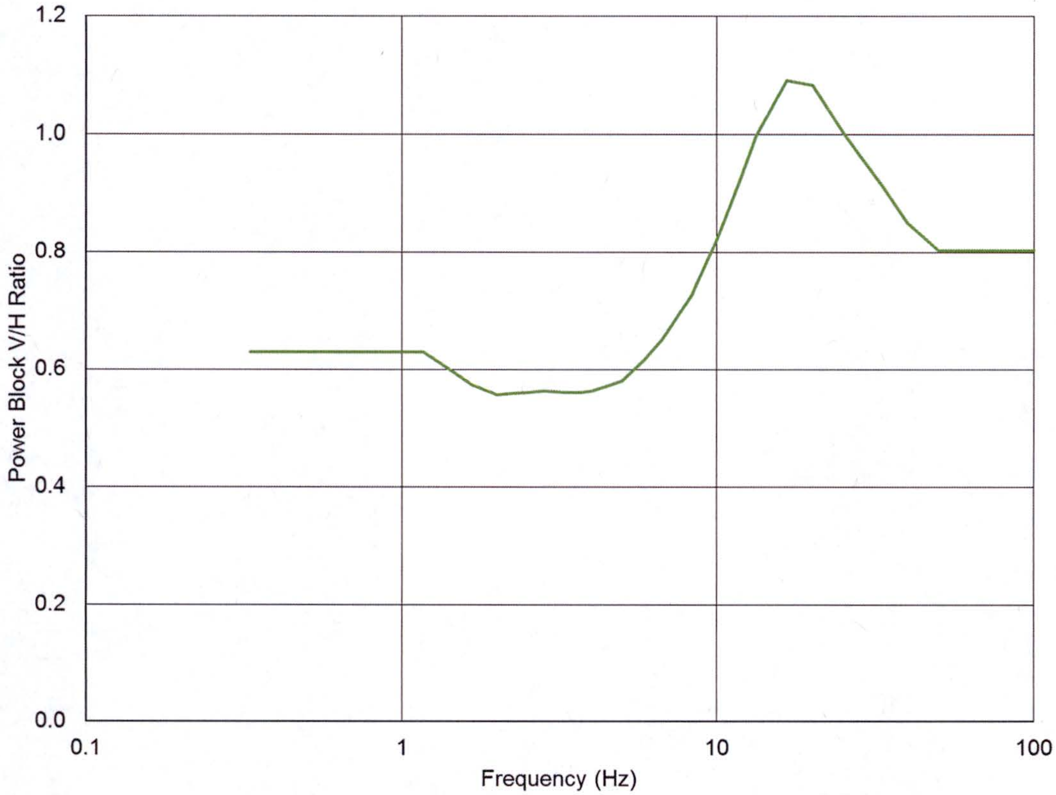

Figure 3-3 – V/H ratio for the Power-Block Structures ([41], Figure 4-21)

4. Determination of Seismic Fragilities for SPRA

This section provides a summary of the process for identifying and developing fragilities for the SSCs that participate in the plant response to a seismic event for the DCPP SPRA. The subsections provide brief summaries of these elements.

4.1 Seismic Equipment List

For the DCPP SPRA, a seismic equipment list (SEL) was developed that includes those SSCs that are important to achieving safe shutdown following a seismic event, and to mitigating radioactivity release if core damage occurs, and that are included in the SPRA model. The methodology used to develop the SEL is generally consistent with the guidance provided in EPRI Technical Report No. 3002000709 [14].

4.1.1 SEL Development

The intent of SEL development is to identify all components that could have an impact on SCDF or SLERF. The ASME/ANS PRA Standard [7] discusses the use of the internal events PRA as a starting point for developing the SPRA model. In accordance with the requirements from the ASME/ANS PRA Standard, the process for DCPP began by identifying all existing components included in the internal events and the original SPRA models. Then, through walk-downs and a comprehensive review of potential seismically-induced interactions (such as block wall failure impacting SEL components), additional SSCs, whose failure could have a unique seismic risk impact were identified. This list of components was then used to identify the scope for the fragility analysis. Because Unit 1 and 2 are essentially identical, the SEL was developed for Unit 1 but is applicable to Unit 2 as well.

The DCPP SPRA uses the PRA Software "Riskman" [55]. The DCPP Riskman event tree structure was developed to model accident progression associated with initiating events identified for seismic events (see Section 5.1). These initiators were identified through a review of the internal and existing seismic models as well as other potential seismic initiators and includes loss of offsite power (LOSP), station blackout, small loss of coolant accidents (LOCAs), excessive LOCAs (e.g., beyond design basis reactor coolant system (RCS) piping failures).

Safety functions that are required to mitigate seismic initiating events were identified based on EPRI Report No. NP 6041-SL [11] and through a review of other safety functions already included in the DCPP internal events PRA. These safety functions are:

- Reactivity Control

- RCS Inventory Control
- RCS Pressure Control
- Containment isolation
- Decay Heat Removal

The frontline and support systems required to satisfy each of these safety functions were determined by review of the DCPP internal events and the existing SPRA models. Some SSCs that were credited in the internal events model were included on the SEL, but were assumed to fail for any seismic initiator because they lack seismic qualification and are dependent on offsite power. Additional discussion on components that are assumed failed is presented in Section 5.3.2. In addition, certain SSCs are assumed to have a high seismic capacity, and were included on the SEL, but excluded from fragility analysis. These include inline filters, check valves, manual valves and strainers (more discussion on component screening is included in Section 4.4.1).

Aside from the equipment added to the SEL from the internal events PRA, the following types of SSCs were added:

- Additional electrical cabinets were identified through review of the DCPP fire PRA. These cabinets were identified because they contain wiring that could impact modeled functions.
- Seismically-induced flooding sources were identified by review of the DCPP internal flooding PRA and through the focused flooding walk-downs (see PG&E Calculation No. F.6.6 [50]).
- Contacts subject to chatter that could also impact modeled safety functions were identified through a contact chatter circuit analysis. This analysis included a review of chatter impacts that could result in spurious operation or isolation of modeled SSCs (see Section 4.1.2 for more discussion).
- Instrumentation required for operator action was identified through a review of the cues credited in the human reliability analysis (HRA).

Buildings or structures that either house SEL components or whose failure could impact PRA modeled components were selected for inclusion on the SEL. These buildings/structures are:

- containment structure
- auxiliary building

- turbine building
- intake structure
- block walls and non-load bearing concrete walls having the potential for seismic interactions

To ensure the completeness of the DCPP SEL, the SELs from two similar nuclear power plants were reviewed. No additional SSCs were added to the DCPP SEL as a result of the review of these SELs.

The resulting SEL for each unit includes approximately 1,700 component/structure entries. Approximately 1,200 of these entries required fragility evaluation. The final SEL was documented for the SPRA in PG&E Calculation No. F.6.1 [36].

4.1.2 Contact/Relay Evaluation

During a seismic event, vibratory ground motion can cause contacts to chatter. The chattering of contacts 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 contact chatter evaluation was performed for the DCPP SPRA in accordance with Section 5-2.2 of the ASME/ANS PRA Standard [7]. This evaluation involved the circuit analysis of all modeled components that could be impacted by contact chatter in the circuit analysis and identified contacts that could have an impact on modeled PRA functions. All of these contacts were dispositioned by including the impact of chatter in the SPRA model. Most of the relays are associated with the solid-state protection system (SSPS), startup transformers, 4kV vital alternating current (AC) power, or the 480V vital AC power, and are located in panels associated with these systems.

The scope of the contact chatter was defined by filtering the SEL to identify all valves and pumps. Passive components not impacted by contact chatter were removed (e.g., tanks, steam generators, containment recirculation sump strainers, heat exchangers, manual valves, etc.).

Contacts were identified that may have undesired consequences if chatter were to occur. The panels that house these contacts were then identified and the functional fragility of the panel was calculated as the bounding fragility of all the subcomponents whose function has been modeled the SPRA. The functional fragility is evaluated considering the shake table testing of the original panel and any subsequent shake table testing of replacement subcomponents.

Table 4-1 contains an abbreviated list of representative (lead) panels that house relays/contacts that were identified as having potential chatter concerns. In this table, the number of modeled relays/contacts are mapped to the lead panel. This lead analysis panel, in some cases, represents other similar panels (e.g., the control panel for emergency diesel generator (EDG) No. 1-1 (DC-1-21-E-PNL-GQD11) represents the corresponding control panels for EDG No. 1-2 (DC-1-21-E-PNL-GQD12) and EDG 1-3 (DC-1-21-E-PNL-GQD13) as well).

Table 4-1 - Modeling of Relay Chatter Impacts
([36], Table 3.9-1)

Representative (Lead) Panel No.	Description	Number of Relays/Contacts
DC-0-62-E-PNL-RU	12kV Startup Relay Panel	5
DC-1-04-E-LSB-LD30	Local Starter Board for Valve FCV-95	1
DC-1-21-E-PNL-GQD11	Diesel Generator No. 1-1 Control Panel. (Includes control panels GQD12 and GQD13)	15
DC-1-21-E-S-ES11-PB	Emergency Stop Push Button for Diesel Generator No. 1-1. (Includes Push Buttons ES12 and ES13)	3
DC-1-38-I-PNL-RNSIA	Solid State Protection System Input and Output Relay Cabinet Train A. (Includes Cabinets RNSIA, RNSIB, RNSOA and RNSOB)	14
DC-1-62-E-PNL-SVU	12kV Startup Bus Load Center	1
DC-1-62-E-XF-THU11	Standby Startup Transformer No. 1-1. (Includes control cabinet mounted on the transformer)	2
DC-1-63-E-LC-SHG	4160V (4kV) Switchgear, Bus G. (Also includes SHF and SHH)	36
DC-1-63-E-PNL-RHG	4160V (4kV) Safeguards Relay Panel, Bus G. (Includes RHF and RHH)	16
DC-1-63-E-XF-THU12	Standby Startup Transformer No. 1-2. (Includes control cabinet mounted on the transformer)	1
DC-1-64-E-LC-SPG	480V Bus G Load Center. (Includes SPG and SPH)	14
DC-1-96-E-PNL-RVHE	Station Electrical Relay Board E	5
Total		113

In addition to the contacts and panels identified by review of the SEL, contact chatter analysis was performed for valves that were identified as part of the flow diversion/flow isolation review. From this review, a number of valves were identified that could open or close spuriously and cause an undesired impact as a result of contact chatter. For each of the valves determined to have a deleterious impact, the associated panel or cabinet was identified. In each case, the cabinet was already one for which a fragility analysis had been performed. Because the impact of cabinet failure bounds the impact of contact chatter, the use of the cabinet fragility in the PRA adequately represents the impact of contact chatter. For example, the contact chatter analysis identified contact chatter-induced spurious isolation of component cooling water (CCW) flow control valves. The contacts that can cause spurious isolation are located inside the 480V switchgear for which a fragility analysis has been performed. Failure of the cabinets results in a complete failure of CCW, which has an equivalent impact to spurious isolation of the header valves.

4.2 Walk-Down Approach

This section provides a summary of the methodology and scope of the seismic walk-downs performed for the DCPP SPRA. See Report No. 128027-R-01 for additional details [35]. Walk-downs were performed by personnel with appropriate qualifications as defined in the SPID [2]. Walk-downs of those SSCs included on the SEL were performed, as part of the development of the SEL, to assess the as-installed condition of these SSCs, identify equipment or structures that are not included in the SEL, but whose structure failure may impact nearby SEL items (i.e., seismic interaction concerns), define failure mode (e.g., functionality, structural integrity, or anchorage failure) for SEL items that are not screened out and identify the type of further evaluation required.

Walk-downs were performed in accordance with guidance provided in Section 6.5 of the SPID [2] and the associated requirements in the ASME/ANS PRA Standard [7], and EPRI NP-6041-SL [11].

Note that the guidelines in the EPRI NP 6041-SL [11] permit the use of walk-down observations to screen SEL components from further review. Since the DCPP GMRS exceeds the screening levels of EPRI NP 6041-SL [11], screening was applied to only a limited subset of components that have demonstrated inherent seismic ruggedness in the earthquake experience database. Screened components included check valves, relief valves, and manual valves. Due to their similarity, components that could be screened were expanded to also include in-line strainers and filters.

The SPRA plant model is based on the configuration of Unit 1; thus, detailed walk-downs were conducted for Unit 1. However, since the SPRA is applicable to both Units 1 and 2, walk-bys of the corresponding Unit 2 components were also conducted to confirm similarity between the components in Unit 1 and 2.

The seismic walk-downs of SSCs for the SPRA were coordinated with the walk-downs associated with the resolution of NTTF Recommendation 2.3: Seismic, which are documented in the NTTF 2.3 Seismic Walk-Down Reports for Units 1 and 2 [12, 13, 48, 49]. The walk-downs for the two programs were coordinated since the equipment lists for NTTF 2.3: the seismic walk-down equipment lists (SWELs) and NTTF 2.1: SEL have many components in common and the walk-down objectives focused on similar issues. When a component common to both the SWELs and the SEL was reviewed in the field, walk-down observations were recorded for both programs. During the NTTF 2.3: Seismic effort, walk-downs were performed for selected Unit 2 components analogous to Unit 1 SEL items. For those Unit 2 components, EPRI NP 6041-SL [11] Screening and Evaluation Work Sheets (SEWS) forms were completed in parallel with the NTTF 2.3: Seismic walk-down checklist forms. Subsequently, several additional walk-downs of component that were only on the SEL were performed independently of the walk-downs of the components on the SWELs.

The composition of the seismic review team (SRT) performing the walk-downs followed the guidelines of EPRI NP 6041-SL [11]. When the SRT had a reasonable basis for assuming that a group of components were similar in configuration and anchorage, a single component in this group was selected for the performance of a detailed inspection (the lead item). The lead item was thoroughly reviewed, and the other components then briefly reviewed in the walk-by to confirm the similarity with the lead items and to ensure there were no anomalies in installation or seismic interaction sources. The similarity of a group of SEL items was established based on equipment construction, dimensions, locations, seismic qualification requirements, anchorage type, and configurations. The "similarity-basis" was confirmed through a walk-by.

4.2.1 Seismically-Induced Fire and Flooding Walk-Downs

The potential for seismically-induced fire and flooding was evaluated. An initial walk-down of the selected areas of the auxiliary building (including the fuel handling area and the containment penetration areas), the turbine building, and the intake structure was performed to identify any potential seismically-induced fire or flooding damage to equipment modeled in the SPRA after identifying the areas of interest from the fire and internal flooding PRAs (see Calculation No. F.6.6 [50]).

Results of the initial flooding walk-downs were reviewed and many of the areas were screened out from further consideration because flooding would not impact functions modeled in the SPRA. Subsequent to the initial walk-down, a focused walk-down of areas that were screened in was performed. During the walk-downs, potential spray and flooding scenarios from piping systems and other sources that could impact SEL components was reviewed. Particular emphasis was placed upon threaded or jointed piping. Flood sources, including the fire protection piping, and large tanks were evaluated. PG&E Calculation No. F.6.6 [50] documents the findings of the seismically-induced flooding walk-downs. As identified in Calculation No. F.6.6 [50], several flooding sources were identified and either: 1) determined to not have an impact on SEL components, 2) were modeled in the SPRA or 3) have been modified to preclude the flooding impact.

The potential for seismically-induced fire was also evaluated in conjunction with the seismically-induced flooding walk-downs and documented in [50]. Identified fire sources have either been dispositioned as not having an impact on the SEL components or the impact of the fire on the SEL components is modeled in the SPRA.

4.2.2 Significant Walk-Down Results and Insights

In addition to the reported observations [12, 13, 48, and 49], the following significant item was noted during subsequent walk-downs:

- The supply and exhaust ducts associated with the 480V switchgear room ventilation system, which crosses between the auxiliary building and turbine building at elevation 164 ft., was found to have insufficient flexibility necessary to accommodate seismically-induced differential movements between these adjacent buildings. This issue was entered into the corrective action program. As indicated in Table 6-1, these ducts will be modified to accommodate the seismically-induced differential movements.

Components on the SEL were evaluated for seismic anchorage and interaction effects, effects of component degradation, such as corrosion and concrete cracking, for consideration in the development of SEL fragilities. In addition, walk-downs were performed on operator pathways, and the potential for seismically-induced fire and flooding scenarios was assessed. Two seismically-induced flooding scenarios and one seismically-induced fire scenario were added to the SPRA model. Since the walk-downs were performed in accordance with the guidance provided in Section 6.5 of the SPID [2] and the associated requirements in the ASME/ANS PRA Standard [7], and EPRI NP-6041-SL [11], they were judged to be adequate for use in developing the SSC fragilities for the SPRA.

4.2.3 Seismic Equipment List and Seismic Walk-Downs Technical Adequacy

The DCPP SPRA SEL development and walk-downs were subjected to an independent peer review against the pertinent requirements in the ASME/ANS PRA Standard [7].

The peer review assessment, and subsequent disposition of peer review findings through an independent assessment, is described in Appendix A, and establishes that the DCPP SEL and seismic walk-downs are suitable for this SPRA application.

4.3 Dynamic Analysis of Structures

This section summarizes the dynamic analyses of structures that contain systems and components important to achieving a safe shutdown (i.e., structures listed on the SEL, as described in Section 4.1):

- containment structures (one per Unit): reinforced concrete with a steel liner, founded on rock
- auxiliary building (common to both Units): reinforced concrete with a structural steel superstructure, founded on rock
- turbine building (common to both Units): steel framing (combined braced frame and moment frame) with reinforced concrete shear walls, steel shear walls, reinforced concrete floor diaphragms, and steel floor diaphragms, founded on rock
- intake structure (common to both Units): reinforced concrete, embedded in rock

Dynamic analyses of these structures were performed using fixed-base and/or SSI methods (as applicable - see Table 4-2). This section discusses the methodologies used, responses at various locations within the structures and relevant outputs (e.g., in-structure response spectra (ISRS), displacements, etc.). The selection of fixed-base versus SSI analysis methodologies is consistent with Section 6.3.3 of the SPID [2].

Table 4-2 - Description of Structures and Dynamic Analysis Methods for DCPP SPRA

Structure	Foundation Condition	Type of Model	Analysis Method	Comments/Other Information
Containment Structures	Rock	3D[1] FEM[2]	Probabilistic SSI	SSI analyses for three-directions of input motion.
Auxiliary Building	Rock	3D FEM	Probabilistic SSI	SSI analyses for three-directions of input motion.
Turbine Building	Rock	3D FEM	Probabilistic SSI	SSI analyses for three-directions of input motion.

1 Three-Dimensional (3D)

2 Finite Element Model (FEM)

Table 4-2 - Description of Structures and Dynamic Analysis Methods for DCPP SPRA				
Structure	**Foundation Condition**	**Type of Model**	**Analysis Method**	**Comments/Other Information**
Intake Structure	Rock	3D FEM	Probabilistic Fixed-Base	Dynamic analysis for vertical input motion, horizontal response approximated by FIRS.

4.3.1 Soil-Structure Interaction Analyses

As indicated in Table 4-2, SSI analyses were performed for the containment structures, auxiliary building, and turbine building. These analyses are summarized in PG&E Report No. 128027-R-02 [33]. Soil properties for the SSI analysis were documented in [41]. Soil property data includes the best estimate soil profile and thirty randomized soil profiles for the containment structures, auxiliary building, and turbine building. Soil profiles for the buildings are characterized as layered viscoelastic half-spaces. Soil properties provided include: layer thicknesses, shear wave velocity, compression wave velocity, soil damping, and density. These properties are strain-compatible with each building's FIRS. The best estimate soil profiles served as input to the median-centered deterministic SSI analyses (used for model validation purposes only). The 30 randomized soil profiles are input to the probabilistic SSI response analyses.

PG&E developed earthquake ground motion input for the SSI analyses of the containment structures, auxiliary building, and turbine building based on the FIRS-compatible outcrop time histories. For these structures, the earthquake ground motion input consisted of in-column acceleration time histories at the respective foundation levels. Structure-specific seismic input time histories were developed to support the probabilistic SSI response analyses as follows:

- For analyses considering variabilities in all random variables, PG&E randomly assigned one set of the FIRS-compatible outcrop acceleration time histories to one of the thirty randomized soil profiles. PG&E computed the corresponding in-column (within) time histories through 1D site response analysis. The resulting time history sets serve as input to the SSI response analysis. Each resulting in-column time history set then becomes uniquely paired to the randomized soil profile used in site response.
- For analysis considering the randomness variability of the ground motion only, the 30 FIRS compatible outcrop acceleration time history sets are

assigned to the best estimate soil profile. The corresponding in-column time histories at the foundation level are computed.

4.3.1.1 Probabilistic SSI Analyses

Probabilistic seismic response analyses with composite variability were performed following the approach outlined in the draft of American Society of Civil Engineers (ASCE)/Structural Engineering Institute (SEI) standard No. 4-13 [44] (see [33]). These analyses considered all random variables significant to the seismic response of the structures or composite variability due to both randomness and uncertainty. The approach implemented Latin hypercube sampling (LHS). Variables defined by probability distributions were sampled according to a stratified sampling approach. The combination of the parameters for each response simulation was assembled by Latin hypercube experimental design.

In the LHS method, the number of simulations, N is first selected. N sets of earthquake time histories are developed. The variability of each parameter significant to structure response is defined by a probability distribution. These probability distributions are divided into equally probable bins. For N simulations, N equally probable bins are defined. Each bin is sampled randomly, which defines N values of the parameters. The combination of parameters for each simulation is defined by the LHS approach. Seismic response analyses are performed for each of the N simulations. Probability distributions of the response quantities of interest are characterized by the median and 84th percentile values determined from the calculated results from the N simulations.

Thirty LHS simulations were selected as sufficient to obtain stable estimates of the median and 84^{th} non-exceedance probability (NEP) ISRS. Variables considered in the probabilistic response analysis were: the earthquake ground motion, soil stiffness and damping, and structure stiffness and damping.

The seismic SSI analysis was performed for each of the 30 simulations. Probabilistic ISRS were generated at selected locations in each building. These locations cover areas of systems and components included in the SPRA. ISRS were computed at each location for each of the 30 simulations. For a given location and global coordinate direction, the spectral accelerations at each frequency were sorted and ordered. The 50 percent (i.e., median) and 84 percent NEP spectral accelerations were determined.

The resulting probabilistic ISRS considered all variability associated with earthquake ground motion, soil stiffness and damping, and structure frequency and damping. These ISRS represent response with composite variability.

In addition to probabilistic SSI analyses with composite variability, probabilistic seismic response analyses were performed considering random variability alone. In these analyses, the structural stiffness, damping, and soil profiles for the SSI models were set to their median values, and only the earthquake ground motion was varied. Note that the in-column earthquake acceleration time histories were computed with the best estimate soil profile.

In the fragility evaluations, the probabilistic ISRS considering only ground motion variability is used to determine the logarithmic standard deviation of structure response randomness, β_R. This variability can be evaluated from the difference between the median and 84 percent NEP spectral values.

To account for coupling between horizontal and vertical responses of each structure, combined structural models were developed and the input motions in each of the 3 global directions were applied simultaneously in the probabilistic SSI analyses (see Criterion 2 from Section 6.3.1 of the SPID [2]). All modes up to a minimum of 33 Hz were included in the probabilistic SSI analyses (see Criterion 4 from Section 6.3.1 of the SPID [2]).

4.3.2 Fixed-Base Analyses

A fixed-base response analysis was performed for the intake structure, as summarized in [33]. The application of a fixed-base model was based on the structure configuration and its surrounding media. Since the intake structure is a squat, stiff structure, embedded on three sides in the cliff overlooking the Pacific Ocean (the fourth side is exposed to the Pacific Ocean) and founded on rock, SSI effects are expected to be negligible.

The modal analysis of the fixed-base model of the intake structure included modes up to 33 Hz (see Criterion 4 from Section 6.3.1 of the SPID [2]). Three median-centered fixed-base response spectrum analyses were performed, one for each global direction of input motion, which is acceptable, since there is not significant coupling between the horizontal and vertical responses of the intake structure (see Criterion 2 from Section 6.3.1 of the SPID [2]). The results of the response spectrum analyses were combined using the 100-40-40 combination rule (see Regulatory Guide 1.92 [54], Regulatory Position C.2.1) to obtain member force demands for the fragility evaluation of the structure. Seismic input for the response spectrum analyses of the intake structure was defined by the structure-specific FIRS at Elevation -38.5 ft., which is documented in [41].

As noted above, since the intake structure is low and stiff and SSI effects are expected to be minimal, horizontal in-structure responses for equipment fragility evaluations were approximated by the intake structure FIRS. A vertical fixed-base probabilistic response analysis was performed to obtain vertical ISRS for equipment fragility evaluations. This response analysis was performed to capture possible amplification associated with local vertical flexibility of the slabs and the supporting structure. The probabilistic vertical response analysis followed a LHS approach in which the random variables included the earthquake ground motion, structure stiffness, and structure damping. Thirty LHS time history response simulations were performed using the mode superposition method. Median and 84 percent NEP vertical ISRS were evaluated at selected locations in the intake structure. The results of the vertical response analysis are used only for the fragility evaluation of components in the intake structure.

4.3.3 Structure Response Models

4.3.3.1 Existing Structural Models

The design basis models of the major DCPP structures are described in Section 3.7 of [21]. Different models were used for the Double Design Earthquake (DCPP's equivalent to a Safe Shutdown Earthquake) and the Hosgri Earthquake.

These models were evaluated against the SPID [2] requirements for the modeling of structures and it was determined that new finite element models (FEMs) of all major structures were required.

4.3.3.2 Structural Models Used for Response Analysis

In order to accurately capture the seismic response of the major structures, new detailed 3D FEMs were developed for the response analyses. These 3D FEMs are described in [33]:

- Containment Structure: 3D FEM with SSI. A model of the Unit 1 Containment Structure is used to represent the nearly-identical Unit 2 Containment Structure.
- Auxiliary Building: 3D FEM with SSI. The model considers symmetry about the north-south centerline of this common structure.
- Turbine Building: 3D FEM with SSI. A model of the Unit 2 portion of the turbine building is used to represent the similar Unit 1 portion of this common structure.
- Intake Structure: fixed-base 3D FEM. A model of the entire building is used to represent this common structure.

The FEMs of the containment structure, auxiliary building, and turbine building were coupled with the SSI models of the underlying rock/soil, described in Section 4.3.1.

4.3.3.3 Structural Damping Values

For the seismic response analyses, the structures were assigned median damping ratios based on the expected response level resulting from the reference earthquake. The expected response level was based on the median structure seismic demands and the extent of concrete cracking from the fixed-base response spectrum analysis using the FIRS. The median damping ratios were assigned based on Table 3-2 of ASCE/SEI 43-05 [43].

4.3.3.4 Concrete Cracking

For reinforced concrete structures, the stiffness properties are required to be consistent with the expected response level corresponding to the reference earthquake. At each structure, the reference earthquake was defined in terms of the structure-specific FIRS. Stiffness properties of reinforced concrete elements were determined following the guidance of Section 3.3 of [44]. The extent of crack was determined by an iterative process, starting with models having the uncracked concrete stiffness properties and 4 percent damping. The extent of cracking was determined at the reference level input motion and the member stiffness properties adjusted to account for cracking. A review of the member stress distribution after each analysis was performed to assess whether additional elements required stiffness reduction.

4.3.3.5 Generation of In-Structure Response Spectra

Probabilistic ISRS were generated at selected locations in each building, from either the SSI or fixed-base analyses. These locations cover areas of systems and components included in the SPRA. Detailed discussions of the generation of ISRS are provided in Section 4.3.1.1 for the SSI analyses (containment structure, auxiliary building, and turbine building) and Section 4.3.2 for the fixed-base analysis (intake structure). Both median and 84 percent NEP spectral accelerations were determined.

4.3.4 Seismic Structure Response Analysis Technical Adequacy

The DCPP SPRA seismic structure response and SSI analyses were subjected to an independent peer review against the pertinent requirements of the ASME/ANS PRA Standard [7].

The peer review assessment, and subsequent disposition of the peer review findings through an independent assessment, is described in Appendix A, and establishes that the DCPP seismic structural response analysis is suitable for this SPRA application.

4.4 Structure, System, and Component Fragility Analyses

The SSC seismic fragility analysis considers the impact of seismic events on the probability of SSC failures at a given value of a seismic motion parameter, such as peak ground acceleration (PGA), peak spectral acceleration, floor spectral acceleration, etc. Seismic Fragilities for the DCPP SSCs are defined in terms of the 5 percent damped horizontal spectral acceleration at 5 Hz at the control point reference location. The fragilities of the SSCs that participate in the SPRA accident sequences (i.e., those included on the SEL) are addressed in the model. Seismic fragilities for the significant risk contributors, (i.e., those which have an important contribution to plant risk) are intended to be generally realistic and plant-specific based on actual current conditions of the SSCs in the plant, as confirmed through the detailed walk-downs of the plant.

See Tables 5.4-2 and 5.5-2 tabulations of the fragilities with appropriate parameters (e.g., median capacity, β_r, β_u) and the calculation method and failure modes for those SSCs determined to be sufficiently risk important, based on the final SPRA quantification (as summarized in Section 5).

4.4.1 Structure, System, and Component Screening Approach

At DCPP, based on existing fragility evaluations associated with the existing SPRA developed for the DCPP Long Term Seismic Program (LTSP) [23, 24], it was decided that the separation of variables (SOV) methodology was appropriate for the performance of the fragility evaluations for the majority of structures and components. However, it was recognized that certain components had sufficiently high seismic capacities that that the use of the SOV methodology was not beneficial.

In order to categorize high capacity components on the SEL, two definitions for high capacity components were introduced for fragility evaluations:

- Rugged: Components that have sufficiently high seismic capacity, based on industry experience and engineering judgement that they are not expected to sustain seismically-induced failures at any seismic input level. Rugged SSCs are excluded from the plant model.

  Rugged components include manual valves, check valves, and in-line filters and strainers. In addition, valves identified as other operated valves[3] are also judged to be rugged based on their characteristics.

- Robust: Components that have sufficiently high seismic capacity, based on walk-downs, engineering judgement, review of the margins in the design basis calculations, and/or simplified fragility evaluations, that are not expected to sustain seismically-induced failures at reasonably high seismic input levels. A component high confidence low probability of failure (HCLPF) screening level was established, based on guidance provided in the SPID [2], by convolving the fragility of a single element with the seismic hazard curves such that the SCDF is at most 5×10^{-7} per year. The median and HCLPF spectral acceleration that satisfies this criterion are 10.2g and 4.0g, respectively, expressed in terms of 5 percent damped spectral (5 Hz) acceleration at the site control point. The

[3] DCPP's component database includes the valve category "other operated valves" that applies to manually-operated valves with an external operator (e.g., gear box).

associated logarithmic standard deviation for composite variability, β_c, is 0.40. The contributions of randomness and uncertainty variabilities for a given component were assigned as $\beta_r = 0.24$ and $\beta_u = 0.32$, respectively, based on Table 6-2 of the SPID [2].

For example, solenoid valves, pressure regulators, and backup air supply accumulators, were categorized as robust. Approximately 7 percent of the components modeled in the SPRA were assigned the robust fragility parameters.

A sensitivity study was performed (see PG&E Calculation No. F.6.5 [16], Section 6.3.18) to assess the impact of the robust fragility parameters on the SCDF and the SLERF by including additional fictitious robust components in the SPRA model (i.e., components that are assigned the robust fragility parameters). The sensitivity study conservatively assumed that the seismic failure of any one of the five fictitious robust components leads directly to core damage and large early release. This sensitivity is of interest because it demonstrates the impact of any potential omission of robust SSCs from the model. The results of this sensitivity study show a 0 percent change in SCDF and approximately a 10 percent change in SLERF. Although the sensitivity study shows a more significant impact to SLERF, this is expected given the conservative sensitivity assumption that failure of any one of five individual, hypothetical components could lead directly to LERF.

The screening method described above is consistent with the requirements of Section 6.4.3 of the SPID [2].

4.4.2 Structure, System, and Component Fragility Analysis Methodology

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

- SOV: The SOV methodology was the primary approach used for the seismic fragility analyses of SSCs.
- Conservative Deterministic Failure Margins (CDFM): The CDFM methodology was used for obtaining fragility parameters for two components.
- Earthquake Experience and Industry Consensus Data: Earthquake experience data and industry consensus data (e.g., the EPRI SPRA Implementation Guide [42]) were used as input to the fragility evaluations for the offsite power system, firewater piping, and various non-vital electrical panels.

Consistent with the requirements in the ASME/ANS PRA Standard [7] and Section 6.4.1 of SPID [2], the guidance provided in EPRI Technical Report No. 103959 [51] is used for developing the fragilities for both SOV and CDFM approaches.

Plant specific information based on as-installed conditions of the components, confirmed by detailed plant walk-downs, as documented in PG&E Report No. 128027-R-01 [35], was used in the development of the seismic fragilities. Fragility evaluation methodologies used in the DCPP LTSP [23, 24] and previously reviewed and accepted by the NRC in Supplement No. 34 to the Safety Evaluation Report for DCPP (SSER-34) [45]. Critical failure modes of components, identified during the DCPP LTSP, were used as a starting point for the updated seismic fragility evaluations of these components.

Seismic fragilities for the DCPP structures and components are first expressed in terms of the 5 percent damped horizontal spectral acceleration at 5 Hz associated with the structure-specific FIRS. A subsequent conversion is made to express the fragility in terms of the spectral acceleration associated with the control point GMRS.

Details on the fragility evaluations of DCPP SSCs are provided in PG&E Report No. 128027-R-03 [34]. The selection of the appropriate fragility evaluation methodology is consistent with the requirements of Sections 6.4.1 and 6.4.3 of the SPID [2].

Note that the fragility analyses for SSCs are based on a full range of frequencies, and the DCPP control point GMRS, unlike Central and Eastern United States plants, does not include peaks in the high frequency range (frequencies greater than 10 Hz). Therefore, separate "high-frequency capacity" evaluations of SSCs, including relays, as described in Section 6.4.2 of the SPID [2] are not applicable to DCPP.

4.4.3 Structure, System, and Component Fragility Analysis Results and Insights

Refer to Section 5 for a tabulation of the fragilities for those SSCs (or correlated groups) determined to be sufficiently risk important, based on the final SPRA quantification. For each listed fragility, appropriate parameters (e.g., median capacity, β_r, and $\beta_{U)}$, the fragility evaluation method, and the failure mode(s) addressed in the model are also listed.

4.4.4 Structure, System, and Component Fragility Analysis Technical Adequacy

The DCPP SPRA SSC fragility evaluations were subjected to an independent peer review against the pertinent requirements in the ASME/ANS PRA Standard [7].

The peer review assessment, and subsequent disposition of peer review findings through an independent assessment, is described in Appendix A, and establishes that the DCPP SSC fragility evaluations are suitable for this SPRA application.

5. Plant Seismic Logic Model

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

5.1 Development of the SPRA Plant Seismic Logic Model

The DCPP seismic response model, described in Calculation No. F.6.2 [53], was developed/updated by starting with the DCPP seismic at power PRA model of record DC03A (which is based on the most recent internal events model). DC03A was adapted in accordance with guidance in the ASME/ANS PRA Standard [7]. Seismic fragility-related top events were added to the appropriate portions of the internal events PRA. Some parts of the internal events model that do not apply or that were screened-out were eliminated. The internal events PRA model HRA was adjusted to account for response during and following a seismic event. It should be noted that due to the LTSP [23, 24] and the DCPP Individual Plant Examination of External Events [31], DCPP had a fully developed SPRA prior to this update.

The DCPP seismic plant response model defines sixteen ranges of earthquake magnitude as the sixteen seismic initiating events. The probabilistic failures of plant SSCs in response to each seismic initiating event are then all considered within the same large event tree linking structure, separated only by the assumed earthquake magnitude. By this approach, multiple seismic failures may be considered along a single seismic plant response sequence, including those failures which may have been considered as separate initiating events in the internal events PRA (e.g., small LOCA with failure of offsite power and failure of auxiliary saltwater (ASW) cooling). The large event tree linking method is performed utilizing the Riskman PRA software [55].

Rather than combine the event sequence plant response models for all internal initiating events as is into one encompassing seismic event sequence model, some approximations were made taking advantage of the known fragilities for each SSC. It was recognized, for example, that a seismic-caused large LOCA was low in frequency compared to the more consequence limiting seismic-caused excessive LOCAs. In short, the model development started with the linked event tree set for general transients and their associated split fraction fault tree models. These models were then expanded to include seismic induced failures that were not yet represented within those trees; (e.g., excessive LOCAs). The seismic failures are grouped into separate top events from those events representing non-seismic failures. Both seismic and non-seismic (random) SSC failures, as well as operator errors, are included in the seismic plant response models. Generally, the added top events could directly make use of the plant response sequences encoded in the general transient models. Other added top events (e.g., excessive LOCA) were represented conservatively by reduced branching, so long as the final sequences accounted for the subsequent containment response and thereby captured significant accident progression sequences.

An assessment of non-vibratory hazards is documented in Report No. GEO.DCPP.TR.17.02 [61]. This assessment performed a systematic review of five seismic hazards that are not directly related to vibratory seismic inertial response of DCPP SSCs during ground shaking, but as hazards that result from the shaking.

Most, if not all, non-vibratory hazards at DCPP have been recognized, evaluated, repaired/retrofitted, and documented. Report No. GEO.DCPP.TR.17.02 [61] provides a brief summary of these hazards and references the corresponding source documentation. Calculation No. GEO.DCPP.CAL.17.05 [64] provides an assessment of secondary fault rupture impacts. The risk impact of a tsunami that is caused by a local earthquake was assessed in this report (see Section 5.7.4).

As a result of these assessments, two non-vibratory hazards were identified for risk impact evaluation:

1. Failure of the ground slope east of the intake structure was identified as having a potential impact to the buried ASW piping. The development of the fragility for the buried ASW piping considered the potential slope failure (see Report No. 128027-R-03 [34]). This fragility was included in the DCPP SPRA.
2. Failure of the ground slope east of the auxiliary building was identified as having a potential impact to the OWSTs. Calculation No. GEO.DCPP.CAL.17.01 [62] documents development of the fragility for the slope and shows that this fragility is above the "Robust" fragility screening level for a slope failure large enough to impact OWSTs that are located near the toe of the slope.

5.1.1 Seismic Initiators

The seismic initiator bins developed for the seismic plant logic modeling are shown in the table below (PG&E Calculation No. F.6.2 [53]):

Table 5.1-1 – Seismic Initiator Bins

Initiator	Lower Bound (g)	Upper Bound (g)	Frequency (/yr)
SEIS01	0.1	0.35	1.70E-02
SEIS02	0.35	0.5	1.50E-03
SEIS03	0.5	0.75	9.79E-04
SEIS04	0.75	1	3.76E-04
SEIS05	1	1.25	1.92E-04
SEIS06	1.25	1.5	1.13E-04
SEIS07	1.5	1.75	5.57E-05
SEIS08	1.75	2	3.59E-05
SEIS09	2	2.25	2.44E-05
SEIS10	2.25	2.5	1.10E-05
SEIS11	2.5	2.75	1.10E-05
SEIS12	2.75	3	4.95E-06
SEIS13	3	3.5	6.68E-06
SEIS14	3.5	4	4.01E-06
SEIS15	4	6	4.39E-06
SEIS16	6	9	6.39E-07

A seismic event that causes ground motion within the range of the modeled initiators (in excess of 0.1g) is assumed to result in a dual unit trip. The 0.1g spectral acceleration level is less than the seismic reactor trip set-point but is approximately equal to the HCLPF for LOSP (0.08g per Report No. 128027-R-03 [34], Table 7-2), which has the lowest HCLPF of any modeled SSC. Because the seismic reactor trip set-point for the reactor protection system (RPS) is 0.35g PGA at the containment structure base mat (UFSAR [21], Section 7.2.2.1.9 and DCPP Technical Specifications [69], Table 3.3.1-1), assuming a guaranteed reactor trip at 0.1g spectral is slightly conservative but accounts for the potential of a LOSP.

The seismic initiating event frequencies and acceleration ranges are derived from the Seismic Hazard Analysis. The mean hazard is discretized into sixteen different bins for the seismic initiating events. The ranges for those bins and their associated annual frequency are shown in Table 5.1-1. The annual frequency for each bin is calculated by subtracting the exceedance frequency from the upper range of the bin from the lower range of the bin. The frequency for the seismic initiators also includes the plant specific capacity factor of 0.9 ([53], Section 4.2.2).

5.1.2 Seismic Plant Logic modeling Assumption

The occurrence of a very small LOCA (VSLOCA) is considered in the SPRA by estimating the fragility for instrument tubing connected to the RCS. This fragility for instrumentation/impulse lines is judged to be robust based on the seismic walk-down report [35]. A VSLOCA sequence is assumed to eventually result in a safety injection (SI) due to low pressurizer pressure and is modeled using the same success criteria and accident sequence models as a small LOCA. This assumption was verified and validated by performing a specific modular accident analysis program thermal hydraulic analysis [59].

Small LOCAs have the same success criteria in the SPRA as they do in the internal events PRA. Medium and large LOCAs are conservatively assumed to result in core damage and are captured in the excessive LOCA top event.

Other important assumptions made in the plant logic model include the following:

- Rugged SSCs are excluded from seismic accident sequence modeling. These include check valves, manual valves, backup air tanks, inline filters/strainers, and fuses.
- No credit for repair or recovery of failed (seismic or non-seismic) EDGs is taken for seismic initiators.
- Offsite power recovery is not credited in the SPRA.
- Two recovery actions from the diverse and flexible coping strategies (FLEX) extended loss of AC power guidelines (FLEX Support Guideline (FSG) Nos. 04 [56] and 07 [57]) were utilized. These actions are to extend vital battery life and to manually control the turbine driven auxiliary feedwater (AFW) pump for extended loss of AC power scenarios. No FLEX mobile equipment was credited.
- Seismically-induced anticipated transient without scram (ATWS) sequences are not evaluated. Instead, a failure of reactor trip following a seismic event is assumed to result in core damage. Although conservative, this assumption does not impact the overall SCDF significantly.

A list of additional SPRA modeling and quantification assumptions is identified in Section 5.3.2.

5.1.3 Seismic Containment Performance

The SPRA has a separate containment event tree (CET) that evaluates plant damage states and allows for the quantification of release category frequency initiated by seismic events ([53], Section 4.3.2.8). A different CET is used for seismic events because the internal events CET contains macros and split fraction rules that were defined in terms of top events in event trees that are not present or linked in the quantification of accident sequences initiated by a seismic event. The seismic failure of the steam generators, the containment structure exterior structure, or the occurrence of a seismic interfacing system LOCA lead directly to core damage and large early release.

5.1.4 Seismic Structure, System, and Component Response Correlation

Full correlation was modeled between identical components within the same system located on the same elevation within the same building. Zero correlation was modeled between all other components. Components that are fully correlated include ([53], Section 5.3):

- EDGs – All diesel generators for Unit 1 and Unit 2 (3 per unit) are located at elevation 85 ft. in the turbine building and are aligned in the east west direction. Diesel generator 2-3 is slightly different than all other generators, but full correlation is assumed via seismic top event SDG.

- Emergency Core Cooling System (ECCS) Charging Pumps – The ECCS charging pumps are located in the same room within the auxiliary building and are essentially identical via seismic top event SCH.

- Residual Heat Removal (RHR) Pumps – RHR pumps are co-located at elevation 73 ft. in the auxiliary building and are assumed correlated via seismic top event SRH.

- Safety Injection Pumps - SI pumps are co-located at elevation 85 ft. the auxiliary building and are assumed correlated via seismic top event SSI.

- 4kV Switchgear and Buses – 4kV switchgear is located at elevation 119 ft. in the turbine building and a correlated failure is assumed via seismic top event SAC. The Bus F potential transformers and safeguard relay panels are not correlated with the Bus G and H panels because of the difference in mounting arrangements. However, this has a minor impact on the overall risk results and therefore is treated as a correlated component for modeling simplicity.

- 480V Switchgear and Buses - Vital 480V switchgear is located at elevation 100 ft. in the auxiliary building and a correlated failure is assumed via seismic top event S480.

- 125V Direct Current Switchgear and Buses - Vital DC switchgear is located at elevation 115 ft. in the auxiliary building and a correlated failure is assumed via seismic top event SDC for Buses F and G. The battery charger for Bus H exhibits a different seismic response and therefore is modeled independently.

Air-operated valves (AOVs) and motor-operated valves (MOVs) from different systems were checked to ensure that no intersystem components at the same elevation and in the same building were identical. This review determined that in all cases where different systems contained an AOV or MOV at the same elevation, the valves were not identical. Either the valve size was different, or the operator was oriented differently.

Balance of plant piping and cable trays were modeled for separate systems and not correlated. The fragility for these types of distributed components is largely dependent on their support design. Due to the large variation in support placement and orientation, zero correlation was modeled between the different systems. An analysis that assumes full correlation of all cable trays was performed to determine the sensitivity of the cable tray correlation assumption on SCDF and SLERF. The sensitivity case shows a negligible impact to SCDF and a significant impact to SLERF. Although the impact on LERF is significant, the impacts are very conservative and meant only to demonstrate the impacts of a hypothetical plant configuration.

For modeling of identical components such as AOVs, MOVs, and piping; separate fragility data variables were created and named appropriately for the system that they reside in. Those specific fragility data variables were then grouped into the appropriate seismic top event.

For containment isolation penetrations, valve failures are treated as not correlated for each penetration because they are in different buildings or at different elevations. This is modeled by including separate top events for outside containment isolation valves and inside containment isolation valves. Containment isolation function is lost when both the outside and inside valves seismically fail.

5.1.5 Seismic HRA Methodology

New seismic specific human failure events (HFEs) were developed in PG&E Calculation No. F.6.3 [52] in accordance with the same methodology as the internal events HRA as follows:

- IDENTIFY through a systematic review of the relevant procedures the set of operator responses required for each of the accident sequences.
- DEFINE HFEs that represent the impact of not properly performing the required responses, consistent with the structure and level of detail of the accident sequences.
- ASSESS the probability of each HFE using a well-defined and self-consistent process that addresses the plant-specific and scenario-specific influences on human performance, and addresses potential dependencies between HFEs in the same accident sequence
- ASSESS recovery actions (at the cut-set or scenario level) and model only if it has been demonstrated that the action is plausible and feasible for those scenarios to which they are applied. Estimates of probabilities of failure shall address dependency on prior human failures in the scenario

As in the internal events HRA, the seismic HRA uses the Cause-Based Decision Tree Method and human cognitive reliability/operator reliability experiments methods and the HRA Calculator is the tool used to implement these methods. In order to credit internal events operator actions for seismic events, it is necessary to determine what impact, if any, the seismic event will have upon the human error probability (HEP). There is no site specific data for seismic events of the magnitudes of concern; therefore, it was decided to use a combination of the EPRI guidance (EPRI Technical Report No. 1025294 [10]) and operator interviews to determine a means to assess the post-seismic HFEs. It should be noted that unlike for internal events initiators, operator experience cannot effectively account for all of the environmental impacts of an earthquake and its associated aftershocks. This is especially true for local actions. Therefore, more weight was given to the EPRI guidance [10] in assessing the post-seismic impact to the HFEs. In the future, as more operator interviews are conducted, this emphasis on the EPRI guidance may shift.

Three seismic HRA bins were defined as follows to model the impact of different seismic initiators impacting operator actions:

Table 5.1-2 – Seismic HRA Bins ([52], Table 4.1-2)

HRA Bin Name	Bin Description	Spectral Acceleration	Initiator range
Seismic Low	Equivalent to HCLPF for block wall failure that impacts DC panels. Also similar to HCLPF for turbine building shear wall failure. These SSCs have the lowest HCLPF of any safety related SSC.	≤ 1.75g	SEIS01-SEIS07
Seismic High	Upper bound acceleration based on HCLPF for annunciators and instrumentation. Also the hot shutdown panel, which is used for a FLEX steam generator level instrumentation connection has a structural HCLPF above 3g.	> 1.75g ≤ 3.0g	SEIS08-SEIS12
Guaranteed Failed	Acceleration greater than HCLPF for annunciators and instrumentation	> 3.0g	SEIS13-SEIS16

Environmental and timing seismic modifiers were applied to the seismic low and seismic high HEPs with guidance from EPRI methodology. Environmental parameters can range from human factors, such as stress or workload, to physical factors, such as accessibility to a component in the plant. The timing modifiers are based on engineering judgment using the EPRI guidance and the operator interviews as input.

5.1.6 Seismically-Induced Fire and Flooding

Seismically-induced fire impacts include the potential ignition sources from 480V non-vital high-energy arcing fault (HEAF) failures were identified by a comprehensive review of National Fire Protection Association (NFPA) 805 ignition sources documented in Calculation No. F.6.6 [50]. Convolution of the hazard with the fragilities for these non-vital switchgear indicated that their contribution to SCDF could be slightly higher than $1x10^{-06}$/yr and therefore the seismically-induced fire scenarios involving these SSCs were incorporated into the model ([53], Section 5.4.1).

The impact of a 480V non-vital HEAF fire is conservatively modeled in the SPRA plant logic model by assuming that both diesel fuel oil (DFO) transfer pumps are impacted by a fire in the applicable fire area. This assumption is conservative because both DFO pumps are only impacted if whole room burnup occurs. Whole room burnup does not occur with a single cabinet HEAF, even assuming a failure of manual suppression. For these fire scenarios, a conditional ignition frequency was used in conjunction with a seismic fragility of the 480V non-vital cabinets to quantify the impact of a seismically-induced fire in the area.

Two seismically-induced flooding scenarios were identified as having a potential impact to the SPRA ([53], Section 5.4.2). Both have the same bounding impact of a loss of RHR and depletion of the fire water storage tank (FWST) (a backup supply of water to the AFW system) if the flooding is not isolated by operator action. Therefore, this seismic-induced flooding interaction is included in the SPRA by modeling the fragility for firewater piping in seismic top event SFL and by evaluating the operator action to isolate the flooding in top event FLO. A failure of both of these top events results in a loss of RHR and the FWST.

5.2 Seismic Probabilistic Risk Assessment Plant Seismic Logic Model Technical Adequacy

The DCPP SPRA seismic plant response methodology and analysis were subjected to an independent peer review against the pertinent requirements in the ASME/ANS PRA Standard [7].

The peer review assessment and subsequent disposition of peer review findings through an independent assessment is described in Appendix A. It establishes that the DCPP SPRA seismic plant response analysis is suitable for this SPRA application.

5.3 Seismic Risk Quantification

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

SPRA quantification methodology and important modeling assumptions, as documented in Calculation No. F.6.3 [16].

5.3.1 Seismic Probabilistic Risk Assessment Quantification Methodology

For the DCPP SPRA, the following approach was used to quantify the seismic plant response model and determine SCDF and SLERF:

The quantification of the SPRA is performed using the Riskman software to generate the core damage and large early release sequences for the seismic initiating events by using the appropriate linked event trees. The Riskman software integrates the seismic hazard and component fragility from the Riskman Fragility Module with the systems analysis into the event tree quantification. Furthermore, the Riskman software integrates the accident sequences, system models, data, and HRA into the quantification process, accounting for system dependencies, to arrive at the quantitative accident sequence frequencies.

The cutsets are quantified in the systems module in Riskman using the binary decision diagram exact solution. This cutset quantification option computes the top event probability exactly and without requiring frequency or cutset order truncation.

All split fractions used in the event tree quantification are conditional on preceding events in the event tree(s), as well as the initiating event. However, the term "conditional split fractions" used in Riskman refers to the subset of all split fractions whose values are computed by an algebraic combination of other split fraction results; i.e., and not computed directly by evaluation of minimal cutsets derived from a fault tree.

5.3.2 Seismic Probabilistic Risk Assessment Model and Quantification Assumptions

The following assumptions were made as part of the SPRA quantification ([53], Section 2):

1. Full Correlation between SSCs is assumed for components that are identical and located on the same elevation of the same building. Partial correlation was not assumed for any modeled SSCs.
2. A seismic event that causes ground motion within the range of the modeled initiators (in excess of 0.1g) is assumed to result in a dual unit trip. The 0.1g spectral acceleration level is less than the seismic reactor trip set-point but is approximately equal to the HCLPF for LOSP which has the lowest HCLPF of any modeled SSC. Because the seismic reactor trip set-point for the RPS is 0.35g PGA at the containment structure base mat, assuming a guaranteed reactor trip at 0.1g is slightly conservative but accounts for the potential of a LOSP.

3. The FWST is capable of supplying a supplementary source of water for AFW for at least 24 hours for a dual unit trip [58]. The FWST is a shared water source for Unit 1 and Unit 2.
4. In the internal events PRA, credit for ASW supply from the opposite unit is modeled. Opposite unit ASW support systems (4kV and 125V DC) are also credited. However, for seismic events, full correlation is assumed between units for ASW, 4kV switchgear, EDGs, and 125V DC SSCs). No other credit for opposite unit equipment is assumed.
5. Certain non-safety/non-seismically qualified SSCs are assumed to fail for any modeled seismic event due to their low seismic capacities. These SSCs are:
    - Non-vital electrical power (except for support to the load tap changers on the startup transformers)
    - Condensate
    - Main feedwater
    - Instrument air
    - 500kV offsite power
6. Rugged SSCs are excluded from seismic accident sequence modeling. These include check valves, manual valves, backup air tanks, inline filters/strainers, and fuses.
7. Seismically-induced ATWS sequences are not evaluated. Instead, a failure of reactor trip following a seismic event is assumed to result in core damage. Although conservative, this assumption does not impact the overall SCDF significantly.
8. A loss of control room vertical boards and control consoles is assumed to result in a complete loss of instrumentation and control. Scenarios where control room vertical boards seismically fail are recoverable using FLEX actions as long as the hot shutdown panel connections are available.
9. A seismic failure of RCS piping is assumed to result in an excessive LOCA and core damage. Although conservative, this assumption does not impact the overall SCDF significantly due to the high HCLPF of the RCS system and associated piping.
10. Seismic failure of the steam generators is assumed to result in core damage and a containment bypass.
11. No credit for repair or recovery of failed EDGs is taken for seismic initiators.
12. A seismic failure of the turbine building is assumed to result in a steam line break. If main steam isolation valves (MSIVs) fail to isolate following a seismically-induced steam line break, core damage is assumed due to pressure-thermal shock of the reactor vessel.
13. The containment spray system is conservatively assumed to be unavailable for seismic events.

14. The SPRA assumes that a failure of the turbine building shear wall results in a structural failure of the building that leads to failure of all vital AC buses, EDGs, and the CCW heat exchanger.

15. For containment isolation penetrations, valve failures are treated as not correlated for each penetration because they are in different buildings or at different elevations. This modeling is achieved by having a seismic top event for the outside valves and a separate top event for the inside valves. Containment isolation function is lost when both the outside and inside valves seismically fail.

16. The input sensors to the SSPS are modeled in the SPRA. Since the sensors for any single SSPS input parameter are correlated, the "seismic" common cause failure of the SSPS input sensors is also modeled.

17. Unlike in the internal events PRA where system piping failures are very unlikely as compared with other system failures and thus can be ignored, seismic piping failures are considered in the SPRA. Depending on the piping fragility, the impact of piping failures is modeled in the SPRA.

18. Offsite power recovery is not credited in the SPRA.

19. The VSLOCA success criteria is bounded by the small LOCA success criteria and therefore, VSLOCA scenarios use the same accident sequence models as a small LOCA.

20. Two recovery actions from the FLEX extended loss of AC power guidelines were utilized. Sensitivities were performed to assess the impact of each of these actions. These sensitivities show that these FLEX actions have a significant impact to SCDF and SLERF.

21. SFP failure is bounded by failure of the auxiliary building. SFP flooding impacts are not modeled as any flooding impact is bounded by building failure.

22. Phase II FLEX equipment was not included in the SPRA model.

23. The fragility modeled in the SPRA for the 480V switchgear room ventilation system ducts was calculated assuming that the ducts have been modified to provide sufficient flexibility to accommodate seismically-induced differential movements between the turbine and auxiliary buildings (see Section 4.2.2). A sensitivity case was performed assuming the as-built condition. This sensitivity shows a small increase in SCDF/SLERF (See Section 5.7.13).

5.4 Seismic Core Damage Frequency Results

The SPRA performed for DCPP shows that the point-estimate mean SCDF is 2.8×10^{-5} per year ([16], Section 5.1). A discussion of the mean SCDF with uncertainty distribution reflecting the uncertainties in the hazard, fragilities, and model data is presented in Section 5.6. Important contributors are discussed in the following paragraphs.

The top SCDF accident sequences are documented in the SPRA quantification report (PG&E Calculation No. F.6.5 [16]). These are briefly summarized in Table 5.4-1.

Table 5.4-1 - Summary of Top SCDF Accident Sequences
([16], Table 5.3.1-1)

Initiator	Frequency	Failed and Multi-State Split Fractions	Description
SEIS15	1.1475E-006	/BYPFLF*/FBY1*/SCON15*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL1	Sequence 1 involves an earthquake between 4g and 6g that results in a seismically-induced containment structure exterior shell failure (SCON15) that results in core damage.
SEIS14	4.3396E-007	/BYPFLF*/FBY1*/SCON14*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL1	Sequence 2 involves an earthquake between 3.5g and 4g that results in a seismically-induced containment structure exterior shell failure (SCON14) that results in core damage.
SEIS16	3.3941E-007	/BYPFLF*/FBY1*/SCON16*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL1	Sequence 3 involves an earthquake over 6g that results in a seismically-induced containment structure exterior shell failure (SCON16) that results in core damage.
SEIS13	3.0728E-007	/BYPFLF*/FBY1*/SCON13*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL1	Sequence 4 involves an earthquake between 3.0g and 3.5g that results in a seismically-induced containment structure exterior shell failure (SCON13) that results in core damage.
SEIS09	2.3124E-007	/BYPFLF*/FBY1*/SID9*SOP9*/IBYP1F*/IBYP2F*/OGF*OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/ORPTF*RPF*/NRF*NMF*LAF*LBF*CSF*RFF*/REZ*/CM1*/LEVEL1	Sequence 5 involves an earthquake between 2g and 2.25g that results in failure of control room vertical boards (SID9) and offsite power (SOP9) leading to core damage.
SEIS08	1.6552E-007	/BYPFLF*/FBY1*/SID8*SOP8*/IBYP1F*/IBYP2F*/OGF*OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/ORPTF*RPF*/NRF*NMF*LAF*LBF*CSF*RFF*/REZ*/CM1*/LEVEL1	Sequence 6 involves an earthquake between 1.75 and 2g that results in failure of control room vertical boards (SID8) and offsite power (SOP8) leading to core damage.

Table 5.4-1 - Summary of Top SCDF Accident Sequences
([16], Table 5.3.1-1)

Initiator	Frequency	Failed and Multi-State Split Fractions	Description
SEIS15	1.6064E-007	/BYPFLF*/FBY1*/SSG15*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*LAF*LBF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL1	Sequence 7 involves an earthquake between 4g and 6g that results in a seismically-induced steam generator failure (SSG15) that results in core damage.
SEIS09	1.4493E-007	/BYPFLF*/FBY1*/SOP9*SDG9*/IBYP1F*/IBYP2F*/OGF*GFF*GGF*GHF*DGCF*TFF*TGF*THF*OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*FOF*/BYPF*/AMAF*AMBF*IAF*ASF*CCF*/OLP1SB*CHF*SIF*AWF*RPF*HSF*/NRF*NMF*LAF*LBF*FCF*CSF*RFF*/REZ*/CM1*/LEVEL1	Sequence 8 involves an earthquake between 2g and 2.25g that results in seismically-induced failure of the diesel generators (SDG9) and offsite power (SOP9). Operator action to shed DC load fails and control of the TDAFW pump is lost once batteries deplete (OLP1SB).
SEIS09	1.1978E-007	/BYPFLF*/FBY1*/SID9*SOP9*SFL9*/IBYP1F*/IBYP2F*/OGF*OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/ORPTF*RPF*/NRF*NMF*LAF*LBF*CSF*RFF*/REZ*/CM1*/LEVEL1	Sequence 9 involves an earthquake between 2g and 2.25g that results in failure of control room vertical boards (SID9), offsite power (SOP9), and a seismically-induced firewater piping rupture (SFL9) that leads to core damage.
SEIS10	1.1535E-007	/BYPFLF*/FBY1*/SID10*SOP10*/IBYP1F*/IBYP2F*/OGF*OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/ORPTF*RPF*/NRF*NMF*LAF*LBF*CSF*RFF*/REZ*/CM1*/LEVEL1	Sequence 10 involves an earthquake between 2.25g and 2.5g that results in failure of control room vertical boards (SID10) and offsite power (SOP10) leading to core damage.
SEIS11	9.8362E-008	/BYPFLF*/FBY1*/SCON11*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*CSF*RF4S*CPF*/REZ*/CM1*/LEVEL1	Sequence 11 involves an earthquake between 2.5g and 2.75g that results in a seismically-induced containment structure exterior shell failure (SCON11) that results in core damage.
SEIS12	9.5733E-008	/BYPFLF*/FBY1*/SCON12*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*CSF*RF4S*CPF*/REZ*/CM1*/LEVEL1	Sequence 12 involves an earthquake between 2.75g and 3g that results in a seismically-induced containment structure exterior shell failure (SCON12) that results in core damage.

Table 5.4-1 - Summary of Top SCDF Accident Sequences
([16], Table 5.3.1-1)

Initiator	Frequency	Failed and Multi-State Split Fractions	Description
SEIS08	9.5715E-008	/BYPFLF*/FBY1*/SOP8*SDG8*/IBYP1F*/IBYP2F*/OGF*GFF*GGF*GHF*DGCF*TFF*TGF*THF*OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*FOF*/BYPF*/AMAF*AMBF*IAF*ASF*CCF*/OLP1SB*CHF*SIF*AWF*RPF*HSF*/NRF*NMF*LAF*LBF*FCF*CSF*RFF*/REZ*/CM1*/LEVEL1	Sequence 13 involves an earthquake between 1.75g and 2g that results in seismically-induced failure of the diesel generators (SDG8) and offsite power (SOP8). Operator action to shed DC load fails and control of the TDAFW pump is lost once batteries deplete (OLP1SB).
SEIS10	8.9776E-008	/BYPFLF*/FBY1*/SID10*SOP10*SFL10*/IBYP1F*/IBYP2F*/OGF*OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/ORPTF*RPF*/NRF*NMF*LAF*LBF*CSF*RFF*/REZ*/CM1*/LEVEL1	Sequence 14 involves an earthquake between 2.25g and 2.5g that results in failure of control room vertical boards (SID10), offsite power (SOP10), and a seismically-induced firewater piping rupture (SFL10) that leads to core damage.
SEIS01	8.4215E-008	/BYPFLF*/FBY1*//IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*RT1SA*IAF*/ATF*/NRF*NMF*LAF*LBF*CSF*RFF*/REZ*/CM1*/LEVEL1	Sequence 15 involves an earthquake between 0.1g and 0.35g in which reactor trip fails and operator action to manually trip reactor fails (RT1SA). Seismic ATWS is assumed to result in core damage.
SEIS10	8.3927E-008	/BYPFLF*/FBY1*/SOP10*SDG10*/IBYP1F*/IBYP2F*/OGF*GFF*GGF*GHF*DGCF*TFF*TGF*THF*OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*FOF*/BYPF*/AMAF*AMBF*IAF*ASF*CCF*/OLP1SB*CHF*SIF*AWF*RPF*HSF*/NRF*NMF*LAF*LBF*FCF*CSF*RFF*/REZ*/CM1*/LEVEL1	Sequence 16 involves an earthquake between 2.25g and 2.5g that results in seismically-induced failure of the diesel generators (SDG10) and offsite power (SOP10). Operator action to shed DC load fails and control of the TDAFW pump is lost once batteries deplete (OLP1SB).
SEIS11	8.2497E-008	/BYPFLF*/FBY1*/SID11*SOP11*SFL11*/IBYP1F*/IBYP2F*/OGF*OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/ORPTF*RPF*/NRF*NMF*LAF*LBF*CSF*RFF*/REZ*/CM1*/LEVEL1	Sequence 17 involves an earthquake between 2.5g and 2.75g that results in failure of control room vertical boards (SID11), offsite power (SOP11), and a seismically-induced firewater piping rupture (SFL11) that leads to core damage.

Table 5.4-1 - Summary of Top SCDF Accident Sequences
([16], Table 5.3.1-1)

Initiator	Frequency	Failed and Multi-State Split Fractions	Description
SEIS13	8.1854E-008	/BYPFLF*/FBY1*/SAB13*SIDF*STB13*SOPF*SDCF*SDC3F*S480F*SF48F*SACF*SDGF*SFOF*SVIF*SASF*SCCF*SSVF*SCHF*SSIF*SRHF*SAWMF*SAWTF*SFLF*FLOF*SADVF*SSPSF*SMSF*SHSPF*/IBYP1F*/IBYP2F*/OGF*D2FF*D2GF*D2HF*AFF*AGF*AHF*A8FF*A8GF*A8HF*DFF*DGF*DHF*BFF*BGF*BHF*GFF*GGF*GHF*DGCF*TFF*TGF*THF*OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*FOF*/BYPF*/I1F*I2F*I3F*I4F*SAF*SBF*AMAF*AMBF*OSF*IAF*ASF*CCF*SVF*SVHF*/OLPF*CHF*SIF*AWF*ORPTF*RPF*HSF*/NRF*NMF*LAF*LBF*FCF*CSF*RFF*CI4SC*/REZ*/CM1*/LEVEL1	Sequence 18 involves an earthquake between 3.0g and 3.5g that results in a seismically-induced auxiliary building failure (SAB13) and turbine building failure (STB13) that results in a loss of all instrumentation, DC and 480V power that leads directly to core damage.
SEIS09	8.0654E-008	/BYPFLF*/FBY1*/STB9*SOPF*SACF*SDGF*SFOF*SASF*SCCF*/IBYP1F*/IBYP2F*/OGF*AFF*AGF*AHF*A8FF*A8GF*A8HF*DFF*DGF*DHF*BFF*BGF*BHF*GFF*GGF*GHF*DGCF*TFF*TGF*THF*OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*FOF*/BYPF*/AMAF*AMBF*IAF*ASF*CCF*/OLP1SB*CHF*SIF*AWF*ORPTF*RPF*HSF*/NRF*NMF*LAF*LBF*FCF*CSF*RFF*/REZ*/CM1*/LEVEL1	Sequence 19 involves an earthquake between 2g and 2.25g that results in a seismically-induced turbine building (STB9) failure which results in a loss of all AC power and CCW. Operator action to shed DC load fails and control of the TDAFW pump is lost once batteries deplete (OLP1SB). Core damage results due to loss of AFW cooling once batteries are depleted.
SEIS14	8.0644E-008	/BYPFLF*/FBY1*/SSG14*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*LAF*LBF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL1	Sequence 20 involves an earthquake between 3.5g and 4g that results in a seismically-induced steam generator failure (SSG14) that results in core damage.
SEIS15	7.9794E-008	/BYPFLF*/FBY1*/SCON15*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*CVRECB*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL1	Sequence 21 involves an earthquake between 4g and 6g that results in a seismically-induced failure of the containment structure exterior shell (SCON15) that leads to directly to core damage.

Table 5.4-1 - Summary of Top SCDF Accident Sequences
([16], Table 5.3.1-1)

Initiator	Frequency	Failed and Multi-State Split Fractions	Description
SEIS13	7.9296E-008	/BYPFLF*/FBY1*/SAB13*SIDF*SOP13*SDCF*SDC3F*S480F*SF48F*SDG13*SVIF*SCCF*SSVF*SCHF*SSIF*SRHF*SAWMF*SAWTF*SFLF*FLOF*SADVF*SSPSF*SMSF*SHSPF*/IBYP1F*/IBYP2F*/OGF*D2FF*D2GF*D2HF*AFF*AGF*AHF*A8FF*A8GF*A8HF*DFF*DGF*DHF*BFF*BGF*BHF*GFF*GGF*GHF*DGCF*TFF*TGF*THF*OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*FOF*/BYPF*/I1F*I2F*I3F*I4F*SAF*SBF*AMAF*AMBF*OSF*IAF*ASF*CCF*SVF*SVHF*/OLPF*CHF*SIF*AWF*ORPTF*RPF*HSF*/NRF*NMF*LAF*LBF*FCF*CSF*RFF*CI4SC*/REZ*/CM1*/LEVEL1	Sequence 22 involves an earthquake between 3g and 3.5g that results in a seismically-induced failure of the auxiliary building (SAB13), offsite power (SOP13), and the EDGs (SDG13) which leads to core damage.
SEIS11	7.5216E-008	/BYPFLF*/FBY1*/SOP11*SDG11*SFL11*/IBYP1F*/IBYP2F*/OGF*GFF*GGF*GHF*DGCF*TFF*TGF*THF*OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*FOF*/BYPF*/AMAF*AMBF*IAF*ASF*CCF*/OLP1SB*CHF*SIF*AWF*RPF*HSF*/NRF*NMF*LAF*LBF*FCF*CSF*RFF*/REZ*/CM1*/LEVEL1	Sequence 23 involves an earthquake between 2.5g and 2.75g that results in failure of offsite power (SOP11) and EDGs (SDG11) with a seismically-induced firewater piping rupture (SFL11). The operator action to shed DC load fails and control of the TDAFW pump is lost once batteries deplete (OLP1SB) which leads to core damage.
SEIS09	7.5073E-008	/BYPFLF*/FBY1*/SOP9*SDG9*SFL9*/IBYP1F*/IBYP2F*/OGF*GFF*GGF*GHF*DGCF*TFF*TGF*THF*OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*FOF*/BYPF*/AMAF*AMBF*IAF*ASF*CCF*/OLP1SB*CHF*SIF*AWF*RPF*HSF*/NRF*NMF*LAF*LBF*FCF*CSF*RFF*/REZ*/CM1*/LEVEL1	Sequence 24 involves an earthquake between 2g and 2.25g that results in failure of offsite power (SOP9) and EDGs (SDG9) with and seismically-induced firewater piping rupture and the operator action to shed DC load fails and control of the TDAFW pump is lost once batteries deplete (OLP1SB) which leads to core damage.
SEIS13	7.4614E-008	/BYPFLF*/FBY1*/SSG13*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*LAF*LBF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL1	Sequence 25 involves an earthquake between 3g and 3.5 g that results in a seismically-induced failure of the steam generators (SSG13) which leads to core damage.

Table 5.4-1 - Summary of Top SCDF Accident Sequences ([16], Table 5.3.1-1)			
Initiator	**Frequency**	**Failed and Multi-State Split Fractions**	**Description**
SEIS11	7.4467E-008	/BYPFLF*/FBY1*/SID11*SOP11*/IBYP1F*/IBYP2F*/OGF*OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/ORPTF*RPF*/NRF*NMF*LAF*LBF*CSF*RFF*/REZ*/CM1*/LEVEL1	Sequence 26 involves an earthquake between 2.5g and 2.75g that results in failure of all instrumentation and control (SID11) and offsite power (SOP11) which leads to core damage.
SEIS11	6.9226E-008	/BYPFLF*/FBY1*/SAB11*SIDF*SOP11*SDCF*SDC3F*S480F*SF48F*SVIF*SCCF*SSVF*SCHF*SSIF*SRHF*SAWMF*SAWTF*SFLF*FLOF*SADVF*SSPSF*SMSF*SHSPF*/IBYP1F*/IBYP2F*/OGF*D2FF*D2GF*D2HF*AFF*AGF*AHF*A8FF*A8GF*A8HF*DFF*DGF*DHF*BFF*BGF*BHF*GFF*GGF*GHF*DGCF*TFF*TGF*THF*OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*FOF*/BYPF*/I1F*I2F*I3F*I4F*SAF*SBF*AMAF*AMBF*OSF*IAF*ASF*CCF*SVF*SVHF*/OLPF*CHF*SIF*AWF*ORPTF*RPF*HSF*/NRF*NMF*LAF*LBF*FCF*CSF*RFF*CI4SB*/REZ*/CM1*/LEVEL1	Sequence 27 involves an earthquake between 3g and 3.5g that results in a seismically-induced failure of the auxiliary building (SAB11) and offsite power (SOP11) which leads to core damage.
SEIS11	6.8702E-008	/BYPFLF*/FBY1*/SAB11*SIDF*SOP11*SDCF*SDC3F*S480F*SF48F*SVIF*SCCF*SSVF*SCHF*SSIF*SRHF*SAWMF*SAWTF*SFLF*FLOF*SADVF*SSPSF*SMSF*SHSPF*/IBYP1F*/IBYP2F*/OGF*D2FF*D2GF*D2HF*AFF*AGF*AHF*A8FF*A8GF*A8HF*DFF*DGF*DHF*BFF*BGF*BHF*GFF*GGF*GHF*DGCF*TFF*TGF*THF*OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*FOF*/BYPF*/I1F*I2F*I3F*I4F*SAF*SBF*AMAF*AMBF*OSF*IAF*ASF*CCF*SVF*SVHF*/OLPF*CHF*SIF*AWF*ORPTF*RPF*HSF*/NRF*NMF*LAF*LBF*FCF*CSF*RFF*/REZ*/CM1*/LEVEL1	Sequence 28 involves an earthquake between 3g and 3.5g that results in a seismically-induced failure of the auxiliary building (SAB11) and offsite power (SOP11) which leads to core damage.

Table 5.4-1 - Summary of Top SCDF Accident Sequences ([16], Table 5.3.1-1)			
Initiator	**Frequency**	**Failed and Multi-State Split Fractions**	**Description**
SEIS11	6.7894E-008	/BYPFLF*/FBY1*/SOP11*SDG11*/IBYP1F*/IBYP2F*/OGF*GFF*GGF*GHF*DGCF*TFF*TGF*THF*OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*FOF*/BYPF*/AMAF*AMBF*IAF*ASF*CCF*/OLP1SB*CHF*SIF*AWF*RPF*HSF*/NRF*NMF*LAF*LBF*FCF*CSF*RFF*/REZ*/CM1*/LEVEL1	Sequence 29 involves an earthquake between 3g and 3.5g that results in a seismically-induced failure of offsite power (SOP11), EDGs (SDG11), and the operator action to shed DC load fails and control of the TDAFW pump is lost once batteries deplete (OLP1SB) which leads to core damage.
SEIS10	6.5319E-008	/BYPFLF*/FBY1*/SOP10*SDG10*SFL10*/IBYP1F*/IBYP2F*/OGF*GFF*GGF*GHF*DGCF*TFF*TGF*THF*OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*FOF*/BYPF*/AMAF*AMBF*IAF*ASF*CCF*/OLP1SB*CHF*SIF*AWF*RPF*HSF*/NRF*NMF*LAF*LBF*FCF*CSF*RFF*/REZ*/CM1*/LEVEL1	Sequence 30 involves an earthquake between 2.25g and 2.5g that results in seismically-induced failure of the diesel generators (SDG10) and offsite power (SOP10). Operator action to shed DC load fails and control of the TDAFW pump is lost once batteries deplete (OLP1SB).

SSCs with the most significant seismic failure contributions to SCDF are listed in Table 5.4-2, sorted by F-V Importance.

The seismic fragilities for each of the significant contributors is also provided in Table 5.4-2, along with the corresponding limiting seismic failure mode and method of fragility calculation.

Among the top SCDF contributors are:

- The condensate storage tank (CST) and FWST because of their function to supply the AFW system with water as a primary source and supplemental source, respectively.

- The main control room vertical boards, process control and protection system, and nuclear instrumentation regulating transformer because a loss of these components is conservatively assumed to result in a complete loss of instrumentation and control and results in core damage if FLEX actions to control AFW are not successful.

- The failure of auxiliary building, containment structure, and turbine building which results in a loss of the important safety related equipment they house.

- Firewater piping failure results in a drain-down of the FWST and loss of a supplemental water supply for AFW, as well as a potential flooding scenario in the auxiliary building if operator action to isolate the firewater piping break is not successful.

- Loss of EDGs via a direct seismic failure of the EDGs or a failure of the walls surrounding them that result in a loss of the EDGs.

Table 5.4-2 - SCDF Importance Measures Ranked by F-V ([16], Table 5.1.3-1 and Appendix M)							
Component	**Description**	**F-V**	**Median Capacity (g)**	**βr**	**βu**	**Failure Mode**	**Fragility Method**
CSTRW	CST/Refueling water storage tank (RWST)	4.32E-02	4.05	0.14	0.23	Structural - Tangential shear failure of the tank at elevation 125 ft-6 in.	SOV
MCRVB	Main control board vertical board	3.03E-02	6.72	0.29	0.43	Functional	SOV
FWST	Firewater storage tank (FWST)	2.85E-02	4.05	0.14	0.23	Structural - Tangential shear failure of the tank at elevation 125 ft-6 in.	SOV
AXBLDG	Auxiliary building (common structure)	2.81E-02	4.74	0.16	0.27	Structural - Diagonal Shear Failure	SOV
PCAPS	Process control and protection system (Eagle 21)	2.80E-02	7.82	0.28	0.46	Functional	SOV
NW2B	EDG room interior NLBW (between EDG 1-1 and 1-2)	2.27E-02	3.90	0.17	0.30	Structural - Out-of-plane flexural failure	SOV
TBSHR	Turbine building Unit 1 portion (shear wall failure mode)	2.20E-02	4.00	0.12	0.31	Structural - Excessive story drifts in EW shear walls	SOV
IACTX	Nuclear instrumentation regulating transformer	1.64E-02	7.00	0.27	0.42	Functional	SOV
FLOOD	Firewater piping in auxiliary building	1.58E-02	2.54	0.22	0.39	Structural - leakage	Experience based
BW9B	East-West concrete block wall between 4kV cable spreading room Bus G and H	1.24E-02	4.58	0.17	0.34	Structural - Out-of-plane flexural failure	SOV
ASPIPB	ASW piping (buried)	1.14E-02	3.40	0.08	0.04	Structural - pull out from dresser coupling	SOV
CNBLDG	Containment exterior structure	1.08E-02	5.22	0.23	0.19	Structural -	SOV

Table 5.4-2 - SCDF Importance Measures Ranked by F-V ([16], Table 5.1.3-1 and Appendix M)							
Component	**Description**	**F-V**	**Median Capacity (g)**	**βr**	**βu**	**Failure Mode**	**Fragility Method**
						Tangential Shear Failure	
DGENGN	EDG engine (includes jacket water and radiator)	9.13E-03	3.85	0.18	0.20	Structural – shear failure of skid end seismic stays	SOV
DGGNTR	EDG Generator	9.13E-03	3.85	0.18	0.20	Structural – shear failure of skid end seismic stays	SOV
MOVAWT	TDAFW MOVs	6.68E-03	13.83	0.23	0.67	Structural - Binding of valve stem	SOV

The significant non-seismic SSC failures (e.g., random failures of modeled components during the SPRA mission time) are listed in Table 5.4-3. The definition of significant SSCs is defined as having a F-V of greater than or equal to $5x10^{-3}$. The components are ranked by F-V, but the RAW values are displayed as well.

Table 5.4-3 - SCDF Importance Measures Ranked by F-V for Non-Seismic SSC Failures
([16], Table 5.1.5-1)

Basic Event	Description	F-V	RAW
[DABK3FS DABK3GS DABK3HS]	Common cause failure of all three 125VDC batteries	6.85E-02	2.94E+04
BB-T3CB	Common cause failure of three vital 4KV AC circuit breakers	1.72E-02	7.62E+02
SDS_RCPSDS_FTA	RCP shutdown seal fails to actuate and initially seal	7.06E-03	1.84E+00
GX1H1_DG1_FR2	Diesel generator 1-1 fails to run more than 1 hour	6.59E-03	1.32E+00
AWZ_TDP11_FS	TDAFW Pump 11 fails to start	6.26E-03	2.12E+00
GX1F1_DG3_FR2	Diesel generator 1-3 fails to run more than 1 hour	5.97E-03	1.29E+00
GXTLCV	Common cause triple failure of diesel generator fuel oil day tank LCVs	5.82E-03	1.33E+01

A summary of the SCDF results for each seismic hazard interval is presented in Table 5.4-4. Truncation sensitivities were performed to ensure that the quantification of SCDF is adequately converged.

Table 5.4-4 - Contribution to SCDF by Acceleration Interval
([16], Table 5.1.1-1)

Initiator	Lower Bound (g)	Upper Bound (g)	Frequency	Truncation	Quantified SCDF	CCDP	Percent of Total	Unaccounted Freq.
SEIS01	0.1	0.35	1.7E-02	2E-14	6.6E-07	3.9E-05	2.4%	3.9E-09
SEIS02	0.35	0.5	1.5E-03	2E-14	1.4E-07	9.2E-05	0.5%	2.0E-09
SEIS03	0.5	0.75	9.8E-04	2E-14	1.2E-07	1.2E-04	0.4%	1.7E-09
SEIS04	0.75	1	3.8E-04	2E-14	5.8E-08	1.6E-04	0.2%	1.4E-09
SEIS05	1	1.25	1.9E-04	2E-14	4.4E-08	2.3E-04	0.2%	1.3E-09
SEIS06	1.25	1.5	1.1E-04	2E-14	4.2E-08	3.7E-04	0.2%	1.2E-09
SEIS07	1.5	1.75	5.6E-05	2E-14	1.2E-07	2.2E-03	0.4%	1.4E-09
SEIS08	1.75	2	3.6E-05	2E-14	1.1E-06	3.0E-02	3.8%	3.6E-09
SEIS09	2	2.25	2.4E-05	2E-14	2.2E-06	9.0E-02	7.9%	8.1E-09
SEIS10	2.25	2.5	1.1E-05	2E-14	2.1E-06	1.9E-01	7.6%	1.2E-08
SEIS11	2.5	2.75	1.1E-05	2E-14	3.5E-06	3.2E-01	12.6%	3.7E-08
SEIS12	2.75	3	5.0E-06	2E-14	2.4E-06	4.8E-01	8.6%	7.7E-08
SEIS13	3	3.5	6.7E-06	2E-14	6.6E-06	9.9E-01	23.8%	1.7E-07
SEIS14	3.5	4	4.0E-06	2E-14	3.9E-06	9.7E-01	14.0%	2.6E-07
SEIS15	4	6	4.4E-06	2E-14	4.2E-06	9.7E-01	15.3%	2.9E-07
SEIS16	6	9	6.4E-07	2E-14	6.2E-07	9.7E-01	2.2%	2.5E-08
Total Reported Frequencies of the Group:			2.0E-02		2.8E-05			8.9E-07

The significant operator actions, including FLEX actions, are listed in Table 5.4-5.

The operator actions are ranked by F-V, but the RAW values are displayed as well.

The FLEX actions credited in the SPRA only include operator actions that manipulate permanently installed plant equipment using PG&E FSG Nos. FSG 04 [56] and FSG 07 [57]. These two FLEX actions include shedding battery loads to extend battery life for an extended loss of AC power and starting the TDAFW pump and monitoring and control of steam generator levels during a loss of AC and DC power.

Table 5.4-5 - SCDF Importance Measures Ranked by F-V for Top Operator Actions
([16], Table 5.1.4-1)

BE ID	BE Description	Basic Event	F-V	RAW
ZH2DC3	FLEX Action: Operator Fails to Shed Battery Loads on Extended Loss of AC Power-	OLP_OPER_S1B	1.11E-01	1.32

Table 5.4-5 - SCDF Importance Measures Ranked by F-V for Top Operator Actions ([16], Table 5.1.4-1)				
BE ID	**BE Description**	**Basic Event**	**F-V**	**RAW**
	SEISMIC-HIGH			
ZH2AW7	FLEX Action: Operators fail to prevent Steam Generator overfill during a loss of all power (AC & DC) - SEIS-HIGH	OLP_TDFEED_S2B	2.44E-02	1.18
ZH2RP1	Operators fail to trip RCPs from the Control Room on Loss of CCW (13 MIN)	ORPT_OP_F1SB	1.73E-02	1.13
ZH1DC3	FLEX Action: Operator Fails to Shed Battery Loads on Extended Loss of AC Power - SEISMIC-LOW	OLP_OPER_S1A	5.54E-03	1.23
ZH1RP3	Operators fail to trip RCPs locally on Loss of CCW (30 MIN) - SEIS-LOW	ORPS_OPER_F2S	5.49E-03	1.01

5.5 Seismic Large Early Release Frequency Results

The SPRA performed for DCPP shows that the point-estimate mean SLERF is $5.4x10^{-6}$ ([16], Section 5.2). A discussion of the mean SLERF with uncertainty distribution reflecting the uncertainties in the hazard, fragilities, and model data is presented in Section 5.6. Important contributors are discussed in the following subsections.

The top SLERF accident sequences are documented in the [16]. These are briefly summarized in Table 5.5-1.

The evaluation of SLERF is consistent with the requirements of Section 6.5.1 and Table 6-2 of the SPID [2].

Table 5.5-1 - Summary of Top SLERF Accident Sequences
([16], Table 5.3.3-1)

Initiator	Frequency	Failed and Multi-State Split Fractions	Description
SEIS15	1.0554E-006	/BYPFLF*/FBY1*/SCON15*/IBYP1F*/IBYP2F* /OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F *YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/O LPF*ORPTF*ORPSF*/NRF*NMF*CSF*RFF* CPF*/REZ*/CM1*/LEVEL2*ADCET0*P1CET0 *P2CET1*P3CET0*PECET4*C2CT0*L2CT0	Sequence 1 involves an earthquake between 4g and 6g that results in a seismically-induced containment structure exterior shell failure (SCON15) that results directly in SLERF.
SEIS14	3.9914E-007	/BYPFLF*/FBY1*/SCON14*/IBYP1F*/IBYP2F* /OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F *YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/O LPF*ORPTF*ORPSF*/NRF*NMF*CSF*RFF* CPF*/REZ*/CM1*/LEVEL2*ADCET0*P1CET0 *P2CET1*P3CET0*PECET4*C2CT0*L2CT0	Sequence 2 involves an earthquake between 3.5g and 4g that results in a seismically-induced containment structure exterior shell failure (SCON14) that results directly in SLERF.
SEIS16	3.1217E-007	/BYPFLF*/FBY1*/SCON16*/IBYP1F*/IBYP2F* /OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F *YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/O LPF*ORPTF*ORPSF*/NRF*NMF*CSF*RFF* CPF*/REZ*/CM1*/LEVEL2*ADCET0*P1CET0 *P2CET1*P3CET0*PECET4*C2CT0*L2CT0	Sequence 3 involves an earthquake over 6g that results in a seismically-induced containment structure exterior shell failure (SCON16) that results directly in SLERF.
SEIS13	2.8263E-007	/BYPFLF*/FBY1*/SCON13*/IBYP1F*/IBYP2F* /OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F *YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/O LPF*ORPTF*ORPSF*/NRF*NMF*CSF*RFF* CPF*/REZ*/CM1*/LEVEL2*ADCET0*P1CET0 *P2CET1*P3CET0*PECET4*C2CT0*L2CT0	Sequence 4 involves an earthquake between 3g and 3.5g that results in a seismically-induced containment structure exterior shell failure (SCON13) that results directly in SLERF.
SEIS15	1.6060E-007	/BYPFLF*/FBY1*/SSG15*/IBYP1F*/IBYP2F*/ OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F* YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/O LPF*ORPTF*ORPSF*/NRF*NMF*LAF*LBF*C SF*RFF*CPF*/REZ*/CM1*/LEVEL2*ADCET0* C2CT0*L2CT0	Sequence 5 involves an earthquake between 4g and 6g that results in seismic-induced failure of the steam generators (SSG15) which results directly in SLERF.
SEIS15	9.1775E-008	/BYPFLF*/FBY1*/SCON15*/IBYP1F*/IBYP2F* /OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F *YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/O LPF*ORPTF*ORPSF*/NRF*NMF*CSF*RFF* CPF*/REZ*/CM1*/LEVEL2*ADCET0*P1CET0 *P2CET1*P3CET0*C2CT0*L2CT0	Sequence 6 involves an earthquake between 4g and 6g that results in a seismically-induced containment structure exterior shell failure (SCON15) that results directly in SLERF.

Table 5.5-1 - Summary of Top SLERF Accident Sequences
([16], Table 5.3.3-1)

Initiator	Frequency	Failed and Multi-State Split Fractions	Description
SEIS11	9.0469E-008	/BYPFLF*/FBY1*/SCON11*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*CSF*RF4S*CPF*/REZ*/CM1*/LEVEL2*ADCET0*P1CET0*P2CET1*P3CET0*PECET4*C2CT0*L2CT0	Sequence 7 involves an earthquake between 2.5g and 2.75g that results in a seismically-induced containment structure exterior shell failure (SCON11) that results directly in SLERF.
SEIS12	8.8050E-008	/BYPFLF*/FBY1*/SCON12*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*CSF*RF4S*CPF*/REZ*/CM1*/LEVEL2*ADCET0*P1CET0*P2CET1*P3CET0*PECET4*C2CT0*L2CT0	Sequence 8 involves an earthquake between 2.75g and 3g that results in a seismically-induced containment structure exterior shell failure (SCON12) that results directly in SLERF.
SEIS14	8.0622E-008	/BYPFLF*/FBY1*/SSG14*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*LAF*LBF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL2*ADCET0*C2CT0*L2CT0	Sequence 9 involves an earthquake between 3.5g and 4g that results in seismic-induced failure of the steam generators (SSG14) which results directly in SLERF.
SEIS13	7.4594E-008	/BYPFLF*/FBY1*/SSG13*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*LAF*LBF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL2*ADCET0*C2CT0*L2CT0	Sequence 10 involves an earthquake between 3g and 3.5g that results in seismic-induced failure of the steam generators (SSG13) which results directly in SLERF.
SEIS15	7.3391E-008	/BYPFLF*/FBY1*/SCON15*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*CVRECB*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL2*ADCET0*P1CET0*P2CET1*P3CET0*PECET4*C2CT0*L2CT0	Sequence 11 involves an earthquake between 4g and 6g that results in a seismically-induced containment structure exterior shell failure (SCON15) that results directly in SLERF.

Table 5.5-1 - Summary of Top SLERF Accident Sequences ([16], Table 5.3.3-1)			
Initiator	**Frequency**	**Failed and Multi-State Split Fractions**	**Description**
SEIS10	3.6138E-008	/BYPFLF*/FBY1*/SCON10*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*CSF*RF4S*CPF*/REZ*/CM1*/LEVEL2*ADCET0*P1CET0*P2CET1*P3CET0*PECET4*C2CT0*L2CT0	Sequence 12 involves an earthquake between 2.25g and 2.5g that results in a seismically-induced containment structure exterior shell failure (SCON10) that results directly in SLERF.
SEIS14	3.4708E-008	/BYPFLF*/FBY1*/SCON14*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL2*ADCET0*P1CET0*P2CET1*P3CET0*C2CT0*L2CT0	Sequence 13 involves an earthquake between 3.5g and 4g that results in a seismically-induced containment structure exterior shell failure (SCON14) that results directly in SLERF.
SEIS12	3.0922E-008	/BYPFLF*/FBY1*/SSG12*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*LAF*LBF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL2*ADCET0*C2CT0*L2CT0	Sequence 14 involves an earthquake between 2.75g and 3g that results in seismic-induced failure of the steam generators (SSG12) which results directly in SLERF.
SEIS16	3.0622E-008	/BYPFLF*/FBY1*/SSG16*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*LAF*LBF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL2*ADCET0*C2CT0*L2CT0	Sequence 15 involves an earthquake over 6g that results in seismic-induced failure of the steam generators (SSG16) which results directly in SLERF.
SEIS14	2.7755E-008	/BYPFLF*/FBY1*/SCON14*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*CVRECB*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL2*ADCET0*P1CET0*P2CET1*P3CET0*PECET4*C2CT0*L2CT0	Sequence 16 involves an earthquake between 3.5g and 4g that results in a seismically-induced containment structure exterior shell failure (SCON14) that results directly in SLERF.

Table 5.5-1 - Summary of Top SLERF Accident Sequences
([16], Table 5.3.3-1)

Initiator	Frequency	Failed and Multi-State Split Fractions	Description
SEIS16	2.7146E-008	/BYPFLF*/FBY1*/SCON16*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL2*ADCET0*P1CET0*P2CET1*P3CET0*C2CT0*L2CT0	Sequence 17 involves an earthquake over 6g that results in a seismically-induced containment structure exterior shell failure (SCON16) that results directly in SLERF.
SEIS13	2.4576E-008	/BYPFLF*/FBY1*/SCON13*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL2*ADCET0*P1CET0*P2CET1*P3CET0*C2CT0*L2CT0	Sequence 18 involves an earthquake between 3.0g and 3.5g that results in a seismically-induced containment structure exterior shell failure (SCON13) that results directly in SLERF.
SEIS16	2.1708E-008	/BYPFLF*/FBY1*/SCON16*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*CVRECB*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL2*ADCET0*P1CET0*P2CET1*P3CET0*PECET4*C2CT0*L2CT0	Sequence 19 involves an earthquake over 6g that results in a seismically-induced containment structure exterior shell failure (SCON16) that results directly in SLERF.
SEIS11	2.0766E-008	/BYPFLF*/FBY1*/SSG11*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*LAF*LBF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL2*ADCET0*C2CT0*L2CT0	Sequence 20 involves an earthquake between 2.5g and 2.75g that results in seismic-induced failure of the steam generators (SSG11) which results directly in SLERF.
SEIS13	1.9653E-008	/BYPFLF*/FBY1*/SCON13*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*CVRECB*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL2*ADCET0*P1CET0*P2CET1*P3CET0*PECET4*C2CT0*L2CT0	Sequence 21 involves an earthquake between 3.0g and 3.5g that results in a seismically-induced containment structure exterior shell failure (SCON13) that results directly in SLERF.

Table 5.5-1 - Summary of Top SLERF Accident Sequences
([16], Table 5.3.3-1)

Initiator	Frequency	Failed and Multi-State Split Fractions	Description
SEIS15	1.5532E-008	/BYPFLF*/FBY1*/SCON15*/IBYP1F*/IBYP2F*/BF1S*OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL2*ADCET0*P1CET0*P2CET1*P3CET0*PECET4*C2CT0*L2CT0	Sequence 22 involves an earthquake between 4g and 6g that results in a seismically-induced containment structure exterior shell failure (SCON15) and Unit 2 4kV Train F (BF1S) that results directly in SLERF.
SEIS15	1.5532E-008	/BYPFLF*/FBY1*/SCON15*/IBYP1F*/IBYP2F*/BH1S*OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL2*ADCET0*P1CET0*P2CET1*P3CET0*PECET4*C2CT0*L2CT0	Sequence 23 involves an earthquake between 4g and 6g that results in a seismically-induced containment structure exterior shell failure (SCON15) and Unit 2 4kV Train H (BH1S) that results directly in SLERF.
SEIS15	1.5460E-008	/BYPFLF*/FBY1*/SCON15*/IBYP1F*/IBYP2F*/BG1S*OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL2*ADCET0*P1CET0*P2CET1*P3CET0*PECET4*C2CT0*L2CT0	Sequence 24 involves an earthquake between 4g and 6g that results in a seismically-induced containment structure exterior shell failure (SCON15) and Unit 2 4kV Train G (BG1S) that results directly in SLERF.
SEIS15	1.1168E-008	/BYPFLF*/FBY1*/SSG15*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*CVRECB*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*LAF*LBF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL2*ADCET0*C2CT0*L2CT0	Sequence 25 involves an earthquake between 4g and 6g that results in a seismic-induced failure of the steam generators (SSG15) which results directly in SLERF.
SEIS15	1.0186E-008	/BYPFLF*/FBY1*/SCON15*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/OLPF*SDS1*ORPTF*ORPSF*/NRF*NMF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL2*ADCET0*P1CET0*P2CET1*P3CET0*PECET4*C2CT0*L2CT0	Sequence 26 involves an earthquake between 4g and 6g that results in a seismically-induced containment structure exterior shell failure (SCON15) that results directly in SLERF.

Table 5.5-1 - Summary of Top SLERF Accident Sequences ([16], Table 5.3.3-1)			
Initiator	**Frequency**	**Failed and Multi-State Split Fractions**	**Description**
SEIS11	7.8669E-009	/BYPFLF*/FBY1*/SCON11*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*CSF*RF4S*CPF*/REZ*/CM1*/LEVEL2*ADCET0*P1CET0*P2CET1*P3CET0*C2CT0*L2CT0	Sequence 27 involves an earthquake between 2.5g and 2.75g that results in a seismically-induced containment structure exterior shell failure (SCON11) that results directly in SLERF.
SEIS12	7.6565E-009	/BYPFLF*/FBY1*/SCON12*/IBYP1F*/IBYP2F*/OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*/BYPF*/AMAF*AMBF*IAF*/OLPF*ORPTF*ORPSF*/NRF*NMF*CSF*RF4S*CPF*/REZ*/CM1*/LEVEL2*ADCET0*P1CET0*P2CET1*P3CET0*C2CT0*L2CT0	Sequence 28 involves an earthquake between 2.75g and 3g that results in a seismically-induced containment structure exterior shell failure (SCON12) that results directly in SLERF.
SEIS15	7.3563E-009	/BYPFLF*/FBY1*/SAB15*SIDF*STB15*SOPF*SDCF*SDC3F*S480F*SF48F*SACF*SDGF*SFOF*SVIF*SASF*SCB15*SCCF*SCPAF*SCPBF*SSVF*SRW15*SCHF*SSIF*SRHF*SFW15*SAWMF*SAWTF*SFLF*FLOF*SADVF*SSPSF*SMSF*SHSPF*/IBYP1F*/IBYP2F*/OGF*D2FF*D2GF*D2HF*AFF*AGF*AHF*A8FF*A8GF*A8HF*DFF*DGF*DHF*BFF*BGF*BHF*GFF*GGF*GHF*DGCF*TFF*TGF*THF*OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*FOF*/BYPF*/I1F*I2F*I3F*I4F*SAF*SBF*AMAF*AMBF*OSF*IAF*ASF*CCF*SVF*SVHF*/OLPF*RWF*AWF*ORPTF*RPF*HSF*/NRF*NMF*LAF*LBF*FCF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL2*ADCET0*P1CET0*P2CET1*P3CET0*PECET4*C2CT0*L2CT0	Sequence 29 involves an earthquake between 4g and 6g that results in a seismically-induced failure of the auxiliary building (SAB15) and turbine building (STB15) that results in a loss of vital power. Seismic containment bypass occurs (SCB15) which contributes to SLERF.
SEIS15	7.2376E-009	/BYPFLF*/FBY1*/SEL15*SAB15*SIDF*STB15*SOPF*SDCF*SDC3F*S480F*SF48F*SACF*SDGF*SFOF*SVIF*SASF*SCB15*SCCF*SCPAF*SCPBF*SSVF*SRW15*SCHF*SSIF*SRHF*SFW15*SAWMF*SAWTF*SFLF*FLOF*SADVF*SSPSF*SMSF*SHSPF*/IBYP1F*/IBYP2F*/OGF*D2FF*D2GF*D2HF*AFF*AGF*AHF*A8FF*A8GF*A8HF*DFF*DGF*DHF*BFF*BGF	Sequence 30 involves an earthquake between 4g and 6g that results in an excessive LOCA (SEL15) and a failure of the auxiliary building (SAB15) and turbine building (STB15). Seismic containment bypass occurs which contributes to SLERF (SCB15).

Table 5.5-1 - Summary of Top SLERF Accident Sequences ([16], Table 5.3.3-1)			
Initiator	**Frequency**	**Failed and Multi-State Split Fractions**	**Description**
		*BHF*GFF*GGF*GHF*DGCF*TFF*TGF*THF*OGRF*OGAF*NVF*X5F*XGF*XFF*XHF*Y5F*YGF*YFF*YHF*FOF*/BYPF*/I1F*I2F*I3F*I4F*SAF*SBF*AMAF*AMBF*OSF*IAF*ASF*CCF*SVF*SVHF*/OLPF*RWF*AWF*ORPTF*RPF*HSF*/NRF*NMF*LAF*LBF*FCF*CSF*RFF*CPF*/REZ*/CM1*/LEVEL2*ADCET0*C2CT0*L2CT0	

SSCs with the most significant seismic failure contribution to SLERF are listed in Table 5.5-2, sorted by F-V. The seismic fragilities for each of the significant contributors are also provided in Table 5.5-2, along with the corresponding limiting seismic failure mode and method of fragility calculation.

Among the top SLERF contributors are:

- The containment structure failure and steam generator failure(s) which are assumed to result directly to SLERF.
- Failure of the CFCUs or failure of the containment mechanical penetrations is assumed to result in a containment bypass. This failure of the CFCU is conservatively assume to fail the CCW piping that supplies the CFCUs and results in a possibly containment bypass pathway.
- The CST and FWST because of their function to supply the AFW system with water as a primary source and supplemental source, respectively.
- The main control room vertical boards, process control and protection system, and nuclear instrumentation regulating transformer because a loss of these components is conservatively assumed to result in a complete loss of instrumentation and control and results in core damage if FLEX actions to control AFW are not successful.
- Firewater piping failure results in a drain-down of the FWST and loss of a supplemental water supply for AFW, as well as a potential flooding scenario in the auxiliary building if operator action to isolate the firewater piping break is not successful.
- Failure of SSPS which results in a failure of containment isolation functions.

Table 5.5-2 - SLERF Importance Measures Ranked by F-V
([16], Table 5.2.3-1 and Appendix M)

Component	Description	F-V	Median Capacity (g)	β_r	β_u	Failure Mode	Fragility Method
CNBLDG	Containment exterior structure	4.41E-01	5.22	0.23	0.19	Structural - Tangential Shear Failure	SOV
STMGN	Steam generators	7.79E-02	9.77	0.26	0.43	Structural - Upper Support Ring Band in bending	SOV
CFCU	Containment fan cooler (piping containment bypass)	6.75E-02	7.34	0.23	0.26	Structural - Housing column footplate weld Failure	SOV
IACBK	120V AC instrument breaker panel	4.25E-02	10.73	0.28	0.55	Functional	SOV
SSPSI	SSPS input relay panel	4.06E-02	5.94	0.13	0.44	Functional	SOV
SSPST	SSPS test cabinet	2.02E-02	5.94	0.13	0.44	Functional	SOV
FLOOD	Firewater piping in auxiliary building	1.94E-02	2.54	0.22	0.39	Structural - leakage	Experience based
PCAPS	Process control and protection system (Eagle 21)	1.88E-02	7.82	0.28	0.46	Functional	SOV
MCRVB	Main control Board vertical board	1.60E-02	6.72	0.29	0.43	Functional	SOV
CSTRW	CST/RWST	1.49E-02	4.05	0.14	0.23	Structural - Tangential shear failure of the tank at elevation 125 ft-6 in.	SOV
CMEPEN	Containment mechanical penetration	1.48E-02	10.2	0.24	0.32	Structural	SOV
ASPIPB	ASW piping (buried)	1.30E-02	3.40	0.08	0.04	Structural - pull out from dresser	SOV

Table 5.5-2 - SLERF Importance Measures Ranked by F-V
([16], Table 5.2.3-1 and Appendix M)

Component	Description	F-V	Median Capacity (g)	β_r	β_u	Failure Mode	Fragility Method
						coupling	
TBSHR	Turbine building Unit 1 portion (shear wall failure mode)	1.15E-02	4.00	0.12	0.31	Structural - Excessive story drifts in EW shear walls	SOV
IACTX	Nuclear instrumentation regulating transformer	1.11E-02	7.00	0.27	0.42	Functional	SOV
FWST	FWST	8.94E-03	4.05	0.14	0.23	Structural - Tangential shear failure of the tank at elevation 125 ft-6 in.	SOV

The significant non-seismic SSC SLERF contributor (e.g., random failures of modeled components during the SPRA mission time) is listed in Table 5.5-3. The definition of significant SSCs is defined as having a F-V of greater than or equal to $5x10^{-3}$.

The components are ranked by F-V, but the RAW values are displayed as well.

Table 5.5-3 - SLERF Importance Measures Ranked by F-V for Non-Seismic SSC Failures
([16], Table 5.2.5-1)

Basic Event	Description	F-V	RAW
[DABK3FS DABK3GS DABK3HS]	Common cause failure of all three 125VDC batteries	1.86E-02	7.98E+03

A summary of the SLERF results for each seismic hazard interval is presented in Table 5.5-4. Truncation sensitivities were performed to ensure that the quantification of SLERF is adequately converged.

Table 5.5-4 - Contribution to SLERF by Acceleration Interval
([16], Table 5.2.1-1)

Initiator	Lower Bound (g)	Upper Bound (g)	Frequency	Truncation	Quantified SLERF	CLERP	Percent of Total	Unaccounted Freq.
SEIS01	0.1	0.35	1.7E-02	2E-14	7.3E-08	4.3E-06	1.4%	3.9E-09
SEIS02	0.35	0.5	1.5E-03	2E-14	8.4E-09	5.6E-06	0.2%	2.0E-09
SEIS03	0.5	0.75	9.8E-04	2E-14	6.0E-09	6.2E-06	0.1%	1.7E-09
SEIS04	0.75	1	3.8E-04	2E-14	2.8E-09	7.5E-06	0.1%	1.4E-09
SEIS05	1	1.25	1.9E-04	2E-14	2.1E-09	1.1E-05	0.0%	1.3E-09
SEIS06	1.25	1.5	1.1E-04	2E-14	2.0E-09	1.7E-05	0.0%	1.2E-09
SEIS07	1.5	1.75	5.6E-05	2E-14	4.5E-09	8.0E-05	0.1%	1.4E-09
SEIS08	1.75	2	3.6E-05	2E-14	3.9E-08	1.1E-03	0.7%	3.6E-09
SEIS09	2	2.25	2.4E-05	2E-14	9.1E-08	3.7E-03	1.7%	8.1E-09
SEIS10	2.25	2.5	1.1E-05	2E-14	1.9E-07	1.7E-02	3.5%	1.2E-08
SEIS11	2.5	2.75	1.1E-05	2E-14	3.6E-07	3.3E-02	6.7%	3.7E-08
SEIS12	2.75	3	5.0E-06	2E-14	2.9E-07	5.9E-02	5.4%	7.7E-08
SEIS13	3	3.5	6.7E-06	2E-14	8.9E-07	1.3E-01	16.6%	1.7E-07
SEIS14	3.5	4	4.0E-06	2E-14	9.2E-07	2.3E-01	17.1%	2.6E-07
SEIS15	4	6	4.4E-06	2E-14	2.0E-06	4.5E-01	36.8%	2.9E-07
SEIS16	6	9	6.4E-07	2E-14	5.2E-07	8.1E-01	9.7%	2.5E-08
Total Reported Frequencies of the Group:			2.0E-02		5.4E-06			8.9E-07

The significant operator actions, including FLEX actions, are listed in Table 5.5-5.

The operator actions are ranked by F-V, but the RAW values are displayed as well.

Table 5.5-5 - SLERF Importance Measures Ranked by F-V for Top Operator Actions ([16], Table 5.2.4-1)				
BE ID	**BE Description**	**Basic Event**	**F-V**	**RAW**
ZH2DC3	FLEX Action: Operator Fails to Shed Battery Loads on Extended Loss of AC Power - SEISMIC-HIGH	OLP_OPER_S1B	2.53E-02	1.07
ZH1RP3	Operators fail to trip RCPs locally on Loss of CCW (30 MIN) - SEIS-LOW	ORPS_OPER_F2S	7.06E-03	1.01
ZH2OS1	Operators fail to manually actuate ESF equipment with SSPS failures - SEISMIC-HIGH	OSMANUALSB	6.79E-03	1.24

5.6 Seismic Probabilistic Risk Assessment Quantification Uncertainty Analysis

The Big Loop Monte Carlo (BLMC) feature in Riskman was used to estimate the parametric uncertainty in the SCDF and SLERF results. This feature can be used to perform a full uncertainty analysis by sampling individual data variables, hazard fractiles and fragility split fractions. The full event tree model is then quantified multiple times using the sampled values. Table 5.6-1 summarizes the results of this analysis.

For the uncertainty analysis, all 16 seismic initiators and all split fractions were included. Although all initiators and split fractions were included in the analysis, only those initiators and split fractions that were important to the analysis were selected for Monte Carlo sampling. The uncertainty case sampled 10 of the 16 seismic initiators representing about 98 percent of the SCDF contribution. In addition, split fractions were included for sampling only if the split fraction had either a RAW value greater than 2 or an F-V value greater than 1E-02. Importance metrics for both SCDF and SLERF were used to identify the risk significant split fractions and initiators.

Table 5.6-1 Results of Parametric Uncertainty Analysis ([16], Table 6.1-1)							
Group	**Iterations**	**Point Estimate (at 1E-11 Truncation)**	**% Difference between PE and BLMC Result**	**Sample Mean**	**Median**	**95th**	**5th**
SCDF	1182	2.40E-05	18%	2.82E-05	1.42E-05	1.02E-04	2.81E-06
SLERF	1182	4.47E-06	17%	5.22E-06	1.86E-06	2.16E-05	1.60E-07

The uncertainty analysis results reflect a broad distribution of SCDF and SLERF. The 95th percentile for SCDF is nearly an order of magnitude above the mean SCDF and the 5th percentile SCDF is nearly an order of magnitude below the mean. These results are similarly broad for SLERF. As expected, most of the uncertainty present in these results originates from the seismic hazard.

A review of generic uncertainty topics provided in EPRI Technical Report No. 1016737 [18] was performed to identify if these topics were applicable to DCPP.

Section 4.3 of [18] provides guidance on performing a sensitivity analysis in the licensing application space as a way of understanding the impact of the source of KEY model uncertainty or related assumption. This list serves as a starting point to identify the set of plant-specific sources of model uncertainty and related assumptions and helped to identify topics for sensitivity cases. The review of generic uncertainties is documented in [16].

Modeling assumptions associated with the DCPP SPRA were collected and reviewed. Of these, most were related to one of the following subject areas:

1. Degree of correlation between components.
2. Consequences of failures (i.e. failure of reactor trip is assumed to result in core damage)
3. Degree of credit for operator recovery actions

Section 5.7 contains a summary of the select sensitivity analyses performed to address assumptions/uncertainties present in the DCPP SPRA.

5.7 Seismic Probabilistic Risk Assessment Quantification Sensitivity Analyses

A set of more than 40 sensitivity cases were performed and documented in [16], Section 6.3, to gain further insight into the inputs and assumptions for the SPRA model. These sensitivities were identified by review of generic pressurized water reactor uncertainties, DCPP SPRA specific assumptions, and through review of significant accident sequences. The sensitivities were performed at an increased truncation for quantification efficiency. The following subsections provide a summary description of a select set of the sensitivity cases that were performed.

Note that the SCDF and SLERF were calculated for sensitivities using a higher quantification truncation frequency in order to expedite the analysis ($1x10^{-11}$/yr. for sensitivities vs. $2x10^{-14}$/yr. for the base model). The base SCDF and SLERF for the sensitivity cases are $2.40x10^{-05}$/yr. and $4.47x10^{-06}$/yr., respectively. Since the increased truncation frequency impacts both the base case and sensitivity case, the relative difference will be similar to a sensitivity analysis using the full truncation frequency.

5.7.1 Seismic HEP distribution 5th and 95th Percentile Values

Sensitivity Case 2 ([16], Section 6.3.2): The SPRA was quantified with the 5th percentile and 95th percentile distribution values. The three statistical values (mean, 5th, and 95th) are plotted below in order or increasing mean value for simplicity. There are several HEPs that have a uniform value of 1.0 and they are not included in this sensitivity. The use of the 5th percentile results in approximately a 10 percent reduction in SCDF and 2 percent reduction in SLERF. The use of the 95th percentile results in approximately a 21 percent increase in SCDF and a 7 percent increase SLERF.

Case	Description	SCDF	SLERF	% Change in SCDF	% Change in SLERF
2	Set HEPs to 5% value	2.17E-05	4.38E-06	-9.6%	-2.0%
	Set HEPs to 95% value	2.90E-05	4.80E-06	20.8%	7.4%

5.7.2 Remove Credit for DC Load-Shedding Operator Action

Sensitivity Case 3 ([16], Section 6.3.3): A sensitivity case was performed on the impact of the DC load-shedding operator action. This operator action was set to fail and the SPRA was re-quantified. This operator action has a significant impact to the SCDF that demonstrates the importance of the FLEX DC load shedding action.

Case	Description	SCDF	SLERF	% Change in SCDF	% Change in SLERF
3	Remove credit for DC load shedding action	3.00E-05	4.68E-06	25.0%	4.7%

5.7.3 Removal of Very Small LOCA Assumption

Sensitivity Case 4 ([16], Section 6.3.4): The impact of the VSLOCA modeling in the PRA was evaluated by removing it from the model and re-quantifying. VSLOCA has a negligible impact to the overall seismic risk.

Case	Description	SCDF	SLERF	% Change in SCDF	% Change in SLERF
4	Removal of VSLOCA accident scenario	2.40E-05	4.48E-06	0.0%	0.2%

5.7.4 Tsunami Impact on ASW

Sensitivity Case 10 ([16], Section 6.3.10): The conditional failure probability of ASW failure due to a locally generated landslide causing a tsunami with a wave height greater than 14 m which would impact the intake structure is assessed below. This sensitivity shows that inclusion of a conditional tsunami impact to the intake structure has an insignificant impact on the SCDF and SLERF.

Case	Description	SCDF	SLERF	% Change in SCDF	% Change in SLERF
10	Evaluation of tsunami impact on ASW	2.40E-05	4.47E-06	0.0%	0.0%

The conditional probability of a tsunami that could reach the 85 ft. level of the site and impact other plant equipment inside the turbine building was also assessed. For all accelerations, the conditional core damage probability is several orders of magnitude higher for vibratory impacts than for the tsunami impact. In this case, the tsunami impact also has an insignificant contribution to risk.

5.7.5 No Credit for Offsite Power Recovery

Sensitivity Case 14 ([16], Section 6.3.14): A sensitivity case was performed on the impact of crediting offsite power for seismic events. This recovery was set to fail and the SPRA was re-quantified. This recovery action has no impact on the SCDF and SLERF.

Case	Description	SCDF	SLERF	% Change in SCDF	% Change in SLERF
14	Removal of credit for offsite power	2.40E-05	4.47E-06	0.0%	0.0%

5.7.6 SSPS Refined Fragility

Sensitivity Case 27 ([16], Section 6.3.27): An improvement in the SSPS fragility was shown to result in a slight decrease of 2 percent in SLERF. LERF is reduced because the change in SSPS fragility results in a lower seismically induced automatic containment isolation failure probability.

Case	Description	SCDF	SLERF	% Change in SCDF	% Change in SLERF
27	Refinement/increase in SSPS fragility by 20%	2.40E-05	4.37E-06	0.0%	-2.2%

5.7.7 48-Hour Mission Time

Sensitivity Case 30 ([16], Section 6.3.30): A sensitivity case was performed on the impact of changing the mission time from 24 hours to 48 hours for important systems. The following systems were modeled with a 48 hour mission time:

1. AFW
2. EDGs and DFO
3. SI, Charging, and RHR ECCS Injection
4. ASW and CCW
5. SSPS
6. Containment isolation
7. 480V switchgear ventilation
8. 480V vital buses

The mission time does not have a significant impact to the SCDF or SLERF, which is to be expected because the seismic failures dominate.

Case	Description	SCDF	SLERF	% Change in SCDF	% Change in SLERF
30	Change mission time to 48 hours	2.50E-05	4.48E-06	4.2%	0.2%

5.7.8 No Credit for any FLEX Actions

Sensitivity Case 33 ([16], Section 6.3.33): The FSG-04 [56] load shed action and FSG-07 [57] recovery action for earthquakes with ground motion between 0.1 and 3g was assumed to fail. The SCDF increased by about 83 percent and SLERF by 17 percent. This analysis case shows that the model is very sensitive to the assumed reliability value for these operator actions.

Case	Description	SCDF	SLERF	% Change in SCDF	% Change in SLERF
33	Removal of credit for any FLEX actions	4.40E-05	5.21E-06	83.3%	16.6%

5.7.9 Hazard Bins for Conditional Large Early Release Probability

Sensitivity Case 35 ([16], Section 6.3.35): Because the conditional large early release probability (CLERP) is calculated as the average over the range of earthquake size for a seismic bin, bins with a large range, such as the SEIS16 initiator that spans from 6g to 9g, could report an underestimated maximum CLERP value for the initiator which can be used to examine if the seismic hazard is fully developed. As shown in Table 5.5-4, the calculated CLERP for the SEIS16 initiator, which represents earthquakes from 6g to 9g, is approximately 0.81 and accounts for almost 10 percent of the total SLERF.

To examine the tail end of the hazard curve definition, a sensitivity case was performed to change the last seismic initiator SEIS16 to represent earthquakes from 8.99g to 9g. The SEIS15 initiator was modified to represent earthquakes from 4g to 8.99g. The results are shown in the table below and show that the tail end of the hazard curve definition accounts for a CLERP of 0.97. This shows that the hazard curve is more developed than the previous SEIS16 initiator definition suggests due to the averaging over the bin range. Further, the exceedance frequency at the highest acceleration initiating event, which corresponds to 9g, is approximately 3E-08 /yr. This low exceedance frequency represents the most any additional frequency could be added to the seismic hazard in the SPRA model.

Initiator	Lower Bound	Upper Bound	Frequency	Truncation	Quantified SLERF	CLERP	Percent of Total	Unaccounted Freq.
SEIS01	0.1	0.35	1.70E-02	2.50E-15	5.18E-08	3.05E-06	0.8%	1.01E-09
SEIS02	0.35	0.5	1.50E-03	2.50E-15	7.47E-09	4.98E-06	0.1%	5.19E-10
SEIS03	0.5	0.75	9.79E-04	2.50E-15	5.81E-09	5.93E-06	0.1%	4.51E-10
SEIS04	0.75	1	3.76E-04	2.50E-15	2.82E-09	7.51E-06	0.0%	3.84E-10
SEIS05	1	1.25	1.92E-04	2.50E-15	2.12E-09	1.10E-05	0.0%	3.79E-10
SEIS06	1.25	1.5	1.13E-04	2.50E-15	2.07E-09	1.83E-05	0.0%	4.29E-10
SEIS07	1.5	1.75	5.57E-05	2.50E-15	6.57E-09	1.18E-04	0.1%	6.01E-10
SEIS08	1.75	2	3.59E-05	2.50E-15	5.03E-08	1.40E-03	0.8%	1.57E-09
SEIS09	2	2.25	2.44E-05	2.50E-15	1.15E-07	4.69E-03	1.8%	3.68E-09
SEIS10	2.25	2.5	1.10E-05	2.50E-15	1.82E-07	1.64E-02	2.8%	6.96E-09
SEIS11	2.5	2.75	1.10E-05	2.50E-15	3.87E-07	3.51E-02	6.0%	2.48E-08
SEIS12	2.75	3	4.95E-06	2.50E-15	3.21E-07	6.48E-02	5.0%	5.09E-08
SEIS13	3	3.5	6.68E-06	2.50E-15	9.59E-07	1.44E-01	15.0%	1.08E-07
SEIS14	3.5	4	4.01E-06	2.50E-15	1.19E-06	2.97E-01	18.6%	1.47E-07
SEIS15	4	8.99	5.03E-06	2.50E-15	3.13E-06	6.21E-01	48.8%	1.76E-07
SEIS16	8.99	9	1.10E-10	2.50E-15	1.07E-10	9.72E-01	0.0%	3.09E-12
Total Reported Frequencies of the Group:			2.03E-02					

5.7.10 Main Steam Line Break on Turbine Building Collapse

Sensitivity Case 36 ([16], Section 6.3.36): The impact of the main steam line break (MSLB) modeling in the PRA was evaluated by removing it from the model and re-quantifying. A seismic event resulting in a turbine building collapse with a failure to close the MSIVs which leads directly to core damage has a negligible impact to the overall seismic risk.

Case	Description	SCDF	SLERF	% Change in SCDF	% Change in SLERF
36	Removal of MSLB on turbine building collapse that results in core damage	2.40E-05	4.47E-06	0.00%	0.00%

5.7.11 Correlation of Containment Isolation Valves

Sensitivity Case 38 ([16], Section 6.3.38): This sensitivity investigates the importance of uncorrelating the containment penetration values. Modeling full correlation between the inside and outside containment isolation valves results in a 6 percent increase in SLERF.

Case	Description	SCDF	SLERF	% Change in SCDF	% Change in SLERF
38	Assume full correlation of containment isolation valves	2.40E-05	4.72E-06	0.0%	5.6%

5.7.12 Top Fragility Components Correlation

Sensitivity Case 39 ([16], Section 6.3.39): This sensitivity investigates the impact of modeling correlation among the top 10 fragility components for both SCDF and SLERF. Some of the top components do not have correlation, but the table below summarizes the impact for those top fragility components that do have correlation assumed and modeled. In all cases shown in the table below, removing correlation in the seismic failures of components results in a reduction in risk. These insights will be considered for future updates.

SCDF Top Contributing Fragilities								
Case	Rank by F-V	Fragility Basic Event	Seismic Top Event	Base case Correlation	SCDF	SLERF	% Change in SCDF	% Change in SLERF
39a	1	CSTRW	SRW	The CST and RWST are assumed correlated.	2.26E-05	4.35E-06	-5.8%	-2.7%
39b	2	MCRVB	SID	Correlation among the vertical boards is assumed.	2.30E-05	4.32E-06	-4.2%	-3.4%
N/A	3	FWST	SFW	No correlation assumed. The FWST is one component.	N/A	N/A	N/A	N/A

SCDF Top Contributing Fragilities								
Case	**Rank by F-V**	**Fragility Basic Event**	**Seismic Top Event**	**Base case Correlation**	**SCDF**	**SLERF**	**% Change in SCDF**	**% Change in SLERF**
N/A	4	AXBLDG	SAB	No correlation assumed. The auxiliary building is one structure.	N/A	N/A	N/A	N/A
39c	5	PCAPS	SID	Correlation among the instrumentation is assume and is modeled directly to core damage.	2.10E-05	4.38E-06	-12.5%	-2.0%
39d	6	NW2B	SDG	Correlation among the walls is assumed and impacts all of the EDGs.	2.10E-05	4.17E-06	-12.5%	-6.7%
N/A	7	TBSHR	STB	No correlation assumed. The turbine building is one structure.	N/A	N/A	N/A	N/A
N/A	8	IACTX	SID	Correlation among the instrumentation is assumed and is modeled directly to core damage.	Same impact as PCAPS. See Sensitivity Case 39c.			
N/A	9	FLOOD	SFL	No correlation assumed. The firewater piping is one representative component.	N/A	N/A	N/A	N/A
39e	10	BW9B	SAC	Correlation among the walls is assumed and impacts all of trains of 4kV AC power.	2.20E-05	4.27E-06	-8.3%	-4.5%

SLERF Top Contributing Fragilities								
Case	**Rank by F-V**	**Fragility Basic Event**	**Seismic Top Event**	**Base case Correlation**	**SCDF**	**SLERF**	**% Change in SCDF**	**% Change in SLERF**
N/A	1	CNBLDG	SCON	No correlation assumed. The containment structure is one structure.	N/A	N/A	N/A	N/A
N/A	2	STMGN	SSG	Not applicable, the failure of 1 steam generator has the same impact to SLERF as failing 2, 3, or all 4 steam generators.	N/A	N/A	N/A	N/A

SLERF Top Contributing Fragilities								
Case	**Rank by F-V**	**Fragility Basic Event**	**Seismic Top Event**	**Base case Correlation**	**SCDF**	**SLERF**	**% Change in SCDF**	**% Change in SLERF**
N/A	3	CFCU	SCB	Not applicable, the bypass failure of 1 CFCU has the same impact to SLERF as failing 2, 3, 4, or all 5 CFCUs.	N/A	N/A	N/A	N/A
39f	4	IACBK	SVI	The 120V AC instrument breaker panels are assumed correlated.	2.30E-05	4.31E-06	-4.2%	-3.6%
39g	5	SSPSI	SSPS	Correlation among the SSPS cabinets is assumed.	2.30E-05	4.35E-06	-4.2%	-2.7%
N/A	6	SSPST	SSPS	Correlation among the SSPS cabinets is assumed.	Same impact as SSPSI. See Sensitivity Case 39g.			
N/A	7	FLOOD	SFL	No correlation assumed. The firewater piping is one representative component.	N/A	N/A	N/A	N/A
N/A	8	PCAPS	SID	Correlation among the instrumentation is assumed and is modeled directly to core damage.	See Sensitivity Case 39c.			
N/A	9	MCRVB	SID	Correlation among the vertical boards is assumed.	See Sensitivity Case 39b.			
N/A	10	CSTRW	SRW	The CST and RWST are assumed correlated.	See Sensitivity Case 39a.			

Furthermore, the input from the fragility team identified select fragilities that have a more uncertain basis for correlation. Those fragilities and a description of the uncertainty in correlation are described below. In some instances, the fragilities identified by the fragility team were already evaluated because it was a top contributor to SCDF or SLERF. The fragility review focused on the following items:

- Fragility of mechanical components is primarily governed by the horizontal ground motion. Therefore, seismic inputs for similar mechanical components in different trains are expected to see similar seismic input.

- Generic components are not considered because a bounding fragility was computed for such components.

- For electrical components, SSCs with more than one train/bus are checked to identify if the components have different seismic input, failure mechanism, and interaction concerns.

Component	Reason for removing correlation
DC-1-65-E-XF-TRY11	TRY12 is oriented perpendicular to the rest. The functional failure of the remaining 3 transformers could be at higher acceleration.
DC-1-65-E-PNL-PY11	PY11A and PY13A are smaller panels and are mounted perpendicular to the remaining panels.
DC-1-67-E-BTC-BTC11	Failure of BTC132 could be influenced by failure of adjacent masonry wall.
DC-1-96-E-PNL-1VB1	Functional failure of the some of the vertical boards could be higher than the others.

Fragility Team Input Sensitivities							
Rank by F-V	**Fragility Basic Event**	**Seismic Top Event**	**Correlation**	**SCDF**	**SLERF**	**% Change in SCDF**	**% Change in SLERF**
N/A	INVTR	SVI	Correlation among the PY cabinets is assumed.	Same impact as IACBK. See Sensitivity Case 39f.			
	IACTX	SID	Correlation among the IY cabinets is assumed.	Same impact as PCAPS. See Sensitivity Case 39c.			
	BATCHG	SDC	Correlation among the battery chargers is assumed and impacts all of trains of DC power.	2.30E-05	4.40E-06	-4.2%	-1.6%
	MCRVB	SID	Correlation among the vertical boards is assumed.	See Sensitivity Case 39b.			

5.7.13 Degraded 480V Switchgear Room Ventilation System Ducts

Sensitivity Case 43 ([16], Section 6.3.43): During the walk-downs associated with the SPRA update, it was found that the supply and exhaust ducts associated with the 480V switchgear room ventilation system is routed between the turbine building and auxiliary building without provisions to accommodate the seismically-induced differential movements between these buildings (see Section 4.2.2 for discussion of this issue). This configuration could result in failure of the ducts during a seismic event. This sensitivity uses fragility parameters calculated for this non-conforming configuration of the 480V switchgear room ventilation system ducts and assesses the impact on SCDF and SLERF.

Case	Description	SCDF	SLERF	% Change in SCDF	% Change in SLERF
43	Include degraded 480V Switchgear Room Ventilation System duct fragility	2.50E-05	4.55E-06	4.2%	1.8%

5.7.14 ASW Buried Piping Fragility

Sensitivity Case 37a ([16], Section 6.3.37): The purpose of this sensitivity case was to assess the impact of changes in the ASW buried piping fragility as a result of a minor change to the hazard analysis. The change was required to resolve comments from the independent assessment associated with the closure of the SPRA peer review findings described in Appendix A. This sensitivity shows that SCDF and SLERF are not impacted by the hazard update.

Case	Description	SCDF	SLERF	% Change in SCDF	% Change in SLERF
37a	Use of updated fragility for ASW buried piping.	2.40E-05	4.47E-06	0.0%	0.0%

5.8 Seismic Probabilistic Risk Assessment Logic Model and Quantification Technical Adequacy

The DCPP SPRA risk quantification and results interpretation methodology were subjected to an independent peer review against the pertinent requirements in the ASME/ANS PRA Standard [7].

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

6. Conclusions

The DCPP SPRA has been updated in accordance with the guidance provided in the ASME/ANS PRA Standard [7]. The DCPP SPRA shows that the SCDF is 2.8×10^{-5} per year ([16], Section 5.1) and the SLERF is 5.4×10^{-6} per year ([16], Section 5.2). .

Although no seismic hazard vulnerabilities were identified, a deficiency in the installation of the ducts associated with the 480V switchgear room ventilation system was identified during the walk-downs performed in support of the SPRA update (see Section 4.2.2). The commitments to address this deficiency are summarized in Table 6-1.

Table 6-1 - Actions to be Performed as a Result of the SPRA Update

Action No.	Component ID	Component Description	Action Description	Completion Date
1	N/A	480V switchgear room ventilation ducts and supports (Unit 1)	Modify ducts and duct supports to accommodate differential movements between turbine and auxiliary buildings	End of Unit 1 Refueling Outage No. 21 (3/2019)

Table 6-1 - Actions to be Performed as a Result of the SPRA Update				
Action No.	Component ID	Component Description	Action Description	Completion Date
2	N/A	480V switchgear room ventilation ducts and supports (Unit 2)	Modify ducts and duct supports to accommodate differential movements between turbine and auxiliary buildings	End of Unit 2 Refueling Outage No. 21 (12/2019)

7. References

1. NRC Letter, "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3 and 9.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," dated March 12, 2012 (ADAMS Accession No. ML12053A340).

2. EPRI Technical Report No. 1025287, "Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," dated February 2013.

3. PG&E Report, "Seismic Hazard and Screening Report - Diablo Canyon Power Plant, Units 1 and 2," Enclosure 1 to PG&E Letter DCL-15-035, "Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding the Seismic Aspects of Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident: Seismic Hazard and Screening Report," dated March 11, 2015 (ADAMS Accession No. ML15070A607).

4. PG&E Report, "Response to NRC Request for Additional Information dated October 1, 2015 and November 12, 2015 Regarding DCPP Seismic Hazard and Screening Report," Enclosure to PG&E Letter DCL-15-154, "Response to NRC Request for Additional Information dated October 1, 2015, and November 12, 2015, Regarding Recommendation 2.1 of the Near-Term Task Force: Seismic Hazard and Screening Report," dated December 21, 2015 (ADAMS Accession Nos. ML15355A550 and ML15355A551).

5. NRC Letter, "Diablo Canyon Power Plant, Unit Nos. 1 and 2 - Staff Assessment of Information Provided under Title 10 of the Code of Federal Regulations, Part 50, Section 50.54(f), Seismic Hazard Reevaluation for Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," dated December 21, 2016 (ADAMS Accession No. ML16341C057).

6. PG&E Report, "Site-Specific Spent Fuel Pool Criteria for the Diablo Canyon Power Plant," Enclosure to PG&E Letter DCL-17-108, "Spent Fuel Pool

Evaluation, Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," dated December 18, 2017 (ADAMS Accession No. ML17352A703).

7. ASME/ANS Standard No RA-Sb-2013, "Addenda B to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," dated June 2013.

8. NEI Document No. NEI 12-13, "External Hazards PRA Peer Review Process Guidelines," Revision 0, dated August 2012.

9. PWROG Report No. PWROG-17022-P, "Peer Review of the Diablo Canyon Units 1 & 2 Seismic Probabilistic Risk Assessment," Revision 0, dated September 2017.

10. EPRI Technical Report No. 1025294, "A Preliminary Approach to Human Reliability Analysis for External Events with a Focus on Seismic," dated December 12, 2012.

11. EPRI Technical Report No. NP 6041-SL, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," Revision 1, dated August 1991.

12. PG&E Report, "Response to Recommendation 2.3 Seismic, Diablo Canyon Power Plant Unit 1," Enclosure 1 to PG&E Letter DCL-12-118, "Response to Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.3 Seismic Unit 1," dated November 27, 2012 (ADAMS Accession No. ML12333A268).

13. PG&E Report, "Response to Recommendation 2.3 Seismic, Diablo Canyon Power Plant Unit 2," Enclosure 1 to PG&E Letter DCL-12-119, "Response to Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.3 Seismic Unit 2," dated November 27, 2012 (ADAMS Accession No. ML12333A270).

14. EPRI Technical Report No. 3002000709, "Seismic PRA Implementation Guide," dated December 2013.

15. NRC Regulatory Guide No. 1.200, "An Approach for Determining The Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, dated March 2009.

16. PG&E Calculation No. F.6.5, "DCPP Seismic PRA Quantification," Revision 3.

17. NRC NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making", Revision 0, dated March 2009.

18. EPRI Technical Report No. 1016737, "Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments," dated December 2008.

19. PG&E Report, "Seismic Source Characterization for the Diablo Canyon Power Plant, San Luis Obispo County, California," dated March 2015.

20. GeoPentech, "Technical Report - Southwestern United States Ground Motion Characterization SSHAC Level 3," Revision 2, dated March 2015.

21. PG&E Report, "Diablo Canyon Power Plant Units 1 & 2 - Final Safety Analysis Report Update," Revision 23, dated December 2016.

22. PG&E Report, "Diablo Canyon Independent Spent Fuel Storage Installation – Updated Final Safety Analysis Report," Revision 6, dated March 2016.

23. PG&E Report, "Long Term Seismic Program Final Report," Enclosure to PG&E Letter DCL-88-192, "Long Term Seismic Program Completion," dated July 31, 1988 (ADAMS Accession No. ML16342C203).

24. PG&E Report, "Addendum to the 1988 Final Report of the Diablo Canyon Long Term Seismic Program," Enclosure to PG&E Letter DCL-91-027, "Addendum to Long Term Seismic Program Final Report", dated February 13, 1991 (ADAMS Accession No. ML16341F963).

25. PG&E Report, "Report to the California Public Utilities Commission - Central Coastal California Seismic Imaging Project," Enclosure to PG&E Letter DCL-14-081, "Central Coastal California Seismic Imaging Project, Shoreline Fault Commitment," dated September 10, 2014 (ADAMS Accession Nos. ML14253A491 and ML14260A387).

26. Lawrence Livermore National Laboratory, NUREG/CR-6372, "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts," dated April 1997.

27. NRC NUREG-2117, "Practical Implementation Guidelines for SSHAC Level 3 and 4 Hazard Studies," dated February 2012.

28. NRC Regulatory Guide No. 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion," Revision 0, dated March 2007.

29. NRC NUREG/CR-2015, "Seismic Safety Margins Research Program Phase I Final Report - Overview," dated April 1981.

30. ASCE/SEI Standard No. ASCE 4-13, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary," dated July 2013.

31. PG&E Report, "Individual Plant Examination of External Events Report for Diablo Canyon Power Plant Unit 2 in Response to Generic Letter 88-20 Supplement 4," Enclosure to PG&E Letter DCL-94-133, "Response to Generic Letter 88-20, Supplement 4, Individual Plant Examination of External Events for Severe Accident Vulnerabilities" dated June 27, 1994 (ADAMS Accession No. ML073600731).

32. PG&E Document No. 128027-CD-01, "Criteria Document for the Seismic Fragility Evaluation of the Diablo Canyon Power Plant," Revision 3.

33. PG&E Report No. 128027-R-02, "Seismic Response Analysis of Diablo Canyon Power Plant," Revision 2.

34. PG&E Report No. 128027-R-03, "Seismic Fragility Evaluation of the Diablo Canyon Power Plant," Revision 1.

35. PG&E Report No. 128027-R-01, "Seismic Walkdown of the Diablo Canyon Power Plant," Revision 3.

36. PG&E Calculation File No. F.6.1, "DCPP Seismic Equipment List Development," Revision 4.

37. NEI Document, "Close-Out of Facts and Observations (F&Os)," Appendix X to Document Nos. 05-04/07-12/12-13, Revision 0, dated February 2017.

38. NRC Letter, "U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 07-1,2 and 12-13, Close-Out of Facts & Observations (F&Os)," dated May 3, 2017 (ADAMS Accession No. ML17079A427).

39. PWROG Report No. PWROG-17078-P, "Independent Assessment of Facts & Observations Closure and Focused Scope Peer Review of the Diablo Canyon Units 1 & 2 Seismic Probabilistic Risk Assessment," Revision 0.

40. NEI Document No. 05-04, "Process for Performing Internal Events PRA Peer Reviews Using ASME/ANS PRA Standard," Revision 2, dated November 2008.

41. PG&E Report No. GEO.DCPP.TR.16.01, "Summary of Ground Motions for Use in Seismic PRA," Revision 4.

42. EPRI Technical Report No. 3002000709, "Seismic Probabilistic Risk Assessment Implementation Guide," dated December 2013.

43. ASCE/SEI Standard No. ASCE 43-05, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities," dated 2005.

44. ASCE/SEI Standard No. ASCE 4-13 (Draft), "Seismic Analysis of Nuclear Safety-Related Structures and Commentary," dated April 2013.

45. NRC NUREG-075, Supplement No. 34, "Safety Evaluation Report Related to the Operation of Diablo Canyon Nuclear Power Plant, Units 1 and 2, Docket Nos. 50-275 and 50-323," dated June 1991.

46. Earthquake Spectra, 27(4), "Site-Specific Design Spectra for Vertical Ground Motion", 1023-1047, accepted January 13, 2011.

47. PG&E Calculation No. GEO.DCPP.15.02, "Updated DCPP GMRS Using the Analytical and Empirical Site Term Approaches," Revision 4.

48. PG&E Report, "Amended Response to Recommendation 2.3 Seismic, Diablo Canyon Power Plant Unit 2," Enclosure 2 to PG&E Letter DCL-13-054, "Response Amendment to Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.3 Seismic," dated May 22, 2013 (ADAMS Accession No. ML13143A168).

49. PG&E Report, "Updated Response to Recommendation 2.3 Seismic, Diablo Canyon Power Plant Unit 1," Enclosure 2 to PG&E Letter DCL-14-041, "Response Update to Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.3 Seismic Unit 1," dated May 8, 2014 (ADAMS Accession No. ML14129A001).

50. PG&E Calculation No. F.6.6, "DCPP Seismic Induced Flooding and Fire Impacts," Revision 2.

51. EPRI Technical Report No. 103959, "Methodology for Developing Seismic Fragilities," dated June 1994.

52. PG&E Calculation No. F.6.3, "DCPP Seismic PRA Human Reliability Analysis," Revision 2.

53. PG&E Calculation F.6.2, "DCPP Seismic PRA Plant Logic Model," Revision 3.

54. NRC Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," Revision 3, dated October 2012."

55. ABS Consulting Computer Program, "Riskman for Windows" Version 14.3, dated 2014

56. PG&E FLEX Support Guideline No. FSG-04, "ELAP DC Load Shed and Management," Revision 1 (Unit 1) and Revision 0 (Unit 2).

57. PG&E FLEX Support Guideline No. FSG-07, "Loss of Vital Instrumentation or Control Power," Revision 1A (Unit 1) and Revision 0A (Unit 2).

58. PG&E Calculation No. RE-20111111, "Coping Time Estimates for IERL L1-11-4," Revision 5.

59. PG&E Calculation No. MAAP 17-05, "Very Small LOCA (VSLOCA) Success Criteria," Revision 0.

60. PG&E Report No. GEO DCPP.TR.17.01, "Screening of Liquefaction Hazards," Revision 1.

61. PG&E Report No. GEO.DCPP.TR.17.02, "Systematic Review of Non-Vibratory Hazard at DCPP from Seismic Events," Revision 1.

62. PG&E Calculation No. GEO.DCPP.CAL.17.01, "Screening of Seismic Hazards Other Than Vibratory Ground Motion," Revision 1.

63. PG&E Calculation No. GEO.DCPP.CAL.17.03, "Tsunami and Ground Motion Vector Hazard," Revision 2.

64. PG&E Calculation No. GEO.DCPP.CAL.17.05, "Vector Hazard Analysis for the Shoreline Fault Secondary Rupture and Ground Motion," Revision 2.

65. PG&E Calculation No. GEO.DCPP.CAL.16.01, "Development of FIRS for Unit 1 and Unit 2 Containment, Turbine Building, and Aux. Building," Revision 3.

66. PG&E Departmental Administrative Procedure TS3.NR1, "Probabilistic Risk Assessment (PRA)," Revision 8.

67. PG&E Administrative Work Procedure E-028, "PRA Model Maintenance and Upgrades," Revision 3.

68. PG&E Interdepartmental Administrative Procedure CF3.ID9, "Design Change Development," Revision 54.

69. "Technical Specifications, Diablo Canyon Nuclear Power Plant, Units 1 and 2," Docket Nos. 50-275 and 50-323, Appendix "A" to License Nos. DPR-80 and DPR-82

8. Acronyms

1D	One-Dimensional
3D	Three-Dimensional
AC	Alternating Current
AFW	Auxiliary Feedwater
ANS	American Nuclear Society
AOV	Air-Operated Valve
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASW	Auxiliary Saltwater
ATWS	Anticipated Transient Without Scram
BLMC	Big Loop Monte Carlo
BVPS	Beaver Valley Power Station
CCW	Component Cooling Water
CDFM	Conservative Deterministic Failure Method
CET	Containment Event Tree
CLERP	Conditional Large Early Release Probability
CFCU	Containment Fan Cooler Unit
CST	Condensate Storage Tank
DC	Direct Current
DCPP	Diablo Canyon Power Plant
DFO	Diesel Fuel Oil
DOE	Department of Energy
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EPRI	Electric Power Research Institute
F-V	Fussell-Vesely
F&O	Fact and Observation
FEM	Finite Element Model
FENOC	FirstEnergy Nuclear Operating Company
FIRS	Foundation Input Response Spectra
FLEX	Diverse and Flexible Coping Strategies
FPIE	Full Power Initiating Event
FSG	FLEX Support Guideline

ft.	foot or feet
FWST	Fire Water Storage Tank
GMPE	Ground Motion Prediction Equation
GMRS	Ground Motion Response Spectra
HCLPF	High Confidence Low Probability of Failure
HEAF	High-Energy Arcing Fault
HEP	Human Error Probability
HFE	Human Failure Event
HLR	High Level Requirements
HRA	Human Reliability Analysis
IAT	Independent Assessment Team
ISRS	In-Structure Response Spectra
JCNRM	Joint Committee on Nuclear Risk Management
km	Kilometers
LHS	Latin Hypercube Sampling
LOCA	Loss of Coolant Accident
LOSP	Loss of Offsite Power
LTSP	Long Term Seismic Program
m	Meter
m/s	Meters per Second
mi.	Mile
MOV	Motor-Operated Valve
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MU	Maintenance and Update
NEI	Nuclear Energy Institute
NEP	Non-Exceedance Probability
NFPA	National Fire Protection Association
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NTTF	Near-Term Task Force
OWST	Outdoor Water Storage Tanks
PGA	Peak Ground Acceleration
PG&E	Pacific Gas and Electric

PPRP	Participatory Peer Review Panel
PRA	Probabilistic Risk Assessment
PRT	Peer Review Team
PSHA	Probabilistic Seismic Hazard Analysis
PWROG	Pressurized Water Reactor Owners Group
RAW	Risk Achievement Worth
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RPS	Reactor Protection System
RWST	Refueling Water Storage Tank
SCDF	Seismic Core Damage Frequency
SEI	Structural Engineering Institute
SEL	Seismic Equipment List
SEWS	Screening Evaluation Work Sheet
SFP	Spent Fuel Pool
SFR	Seismic Fragility Element within the ASME/ANS PRA Standard
SHA	Seismic Hazard Analysis Element within the ASME/ANS PRA Standard
SI	Safety Injection
SLERF	Seismic Large Early Release Frequency
SOV	Separation of Variables
SPID	Screening, Prioritization and Implementation Details
SPR	Seismic PRA Modeling Element within the ASME/ANS PRA Standard
SPRA	Seismic Probabilistic Risk Assessment
SR	Supporting Requirement
SRT	Seismic Review Team
SSC	Structure, System, and Component
SSPS	Solid State Protection System
SSER-34	Supplement No. 34 to the Safety Evaluation Report for the Diablo Canyon Power Plant [45]
SSHAC	Senior Seismic Hazard Advisory Committee
SSI	Soil-Structure Interaction
SSSI	Structure-Soil-Structure Interaction
SWEL	Seismic Walk-Down Equipment List

SWUS	Southwestern United States
TDAFW	Turbine Driven Auxiliary Feedwater
TI Team	Technical Integration Team
UFSAR	Updated Final Safety Analysis Report
UHRS	Uniform Hazard Response Spectrum
USGS	United States Geological Survey
Vs_{30}	Average Shear Wave Velocity to a Depth of 30 meters
VSLOCA	Very Small Loss of Coolant Accident

Appendix A

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

A1. Introduction

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

A2. Peer Review of DCPP Seismic Probabilistic Risk Assessment

The DCPP PRA was subjected to an independent peer review against the pertinent requirements in Part 5 of the current ASME/ANS PRA Standard (RA-Sb-2013) [7]. The peer review assessment (Pressurized Water Reactor Owners Group (PWROG) Report No. PWROG-17022-P [9]) and subsequent closure of the peer review findings by independent assessment (PWROG Report No. PWROG-17078-P [39]), is summarized in this appendix. The scope of the review encompassed the set of technical elements and SRs for the seismic hazard analysis element within the ASME/ANS PRA Standard (SHA), seismic fragility element within the ASME/ANS PRA Standard (SFR), SPRA modeling element within the ASME/ANS PRA Standard (SPR), and SMU (configuration control) elements for SCDF and SLERF. The peer review therefore addressed the set of SRs identified in Tables 6-4 through 6-6 of the SPID [2].

The information presented here establishes that the SPRA has been peer reviewed by a team with adequate credentials to perform the assessment, establishes that the peer review process followed meets the intent of the peer review characteristics and attributes in Table 16 of NRC Regulatory Guide1.200, Revision 2 [15] and the requirements in Section 1-6 of the ASME/ANS PRA Standard [7], and presents the significant results of the peer review.

The DCPP SPRA peer review was conducted during the week of June 19, 2017, at the PG&E Emergency Response Facility in San Luis Obispo, California. As part of the peer review, a walk-down of portions of DCPP Units 1 and 2, located 7 mi. northwest of Avila Beach, California, was performed on June 20, 2017, by three members of the peer review team (PRT) who have the appropriate Seismic Qualification Utilities Group training.

A2.1. Summary of the DCPP SPRA Peer Review Process

The peer review was performed against the requirements in Part 5 (Seismic) of the ASME/ANS PRA Standard [7], using the peer review process defined in NEI 12-13 [8], as required by Section 6.7 of the SPID [2]. The review was conducted over a four-day period, with a summary and exit meeting on the morning of the fifth day.

The SPRA peer review process defined in NEI 12-13 [8] involves an examination by each reviewer of their assigned PRA technical elements against the requirements in the ASME/ANS PRA Standard [7] to ensure the robustness of the model relative to all of the requirements.

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

For each area, i.e., SHA, SFR, SPR, a team of two or three peer reviewers were assigned, one having lead responsibility for that area. For each SR reviewed, the responsible reviewers reached consensus regarding which of the capability categories defined in the ASME/ANS PRA Standard [7] 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 PRT. The ASME/ANS PRA Standard [7] also specifies high level requirements (HLR). Consistent with the guidance in the ASME/ANS PRA Standard [7], 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 PRT's assessment of capability categories, F&Os are prepared. There are three types of F&Os defined in NEI 12-13 [8]:

- findings, which identify issues that must be addressed in order for an SR (or multiple SRs) to meet capability category II;
- suggestions, which identify issues that the reviewers have noted as potentially important but not requiring resolution to meet the SRs; and
- best practices, which reflect the reviewers' opinion that a particular aspect of the review exceeds normal industry practice.

The focus in this Appendix is on findings and their disposition relative to this submittal.

A2.2. Peer Review Team Qualifications

The members of the PRT were Dr. Andrea Maioli of Westinghouse Electric Company LLC, Dr. Martin McCann of Jack Benjamin & Associates, Dr. Richard Quittmeyer of Paul C. Rizzo & Associates, Mr. Jeffery Kimball of Paul C. Rizzo & Associates, Dr. Ram Srinivasan, an independent consultant, Mr. Parthasarathy Chandran of Southern Nuclear Operating Company, Mr. Daniel J. Vasquez of Dominion Resources Services, Mr. Douglas Rapp of FirstEnergy Nuclear Operating Company (FENOC), and Mr. Robert Kirschner of JENSEN HUGHES.

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

Dr. Martin McCann was the lead for the review of the SHA technical element. He has over 35 years of experience in engineering seismology including site response analysis and specification of ground motion. Dr. McCann has served as SHA lead reviewer for a number of recent SPRAs. He was assisted in the hazard review by Dr. Richard Quittmeyer and Mr. Jeffrey Kimball.

Dr. Quittmeyer has over 35 years of experience in project studies, mostly in the nuclear regulatory environment, for site characterization and hazard studies for nuclear waste disposal, nuclear power facilities, reservoirs, and major infrastructure constructions. Dr. Quittmeyer recently served as peer reviewer for the Vogtle Units 1 & 2 SPRA probabilistic seismic hazard analysis (PSHA).

Mr. Kimball has 36 years of experience with the evaluation and characterization of natural phenomena hazards and the design of critical facilities to resist these hazards. He led the preparation of Department of Energy (DOE) standards and guides to define requirements and procedures to complete assessment of natural phenomena hazards. Mr. Kimball has recently defended the seismic hazard review for all the FENOC SPRAs and is a member of the External Events Working Group for the ASME/ANS JCNRM.

Dr. Ram Srinivasan was the lead reviewer for the SFR technical element. Dr. Srinivasan has 42 years of experience in the nuclear industry, principally in the design, analysis (static and dynamic, including seismic), and construction of nuclear power plant structures, spent fuel cask systems including design of independent spent fuel storage installation. Dr. Srinivasan 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. He is also actively involved in the ASME/ANS JCNRM Working Group 5 (External Hazards) responsible for the maintenance of the ASME/ANS PRA Standard being used in the SPRA for Seismic Events. He participated in the SPRA peer review of the Peach Bottom Plant as a working observer and supported the SPRA Peer Review of TVA Watts Bar Plant. He was assisted in fragility review by Mr. Parthasarathy Chandran and Mr. Daniel J. Vasquez.

Mr. Chandran is the SPRA lead for Southern Nuclear Operating Company, with overall responsibility of the Vogtle and Hatch SPRA. He has defended the Vogtle Units 1 & 2 and Hatch SPRA peer reviews and participated as a reviewer for the Fermi, Watts Bar, and Indian Point SPRA peer reviews. He is a member of the ASME/ANS JCNRM Working Group maintaining Part 5 of the ASME/ANS PRA Standard.

Mr. Vasquez also supported the review of the SFR technical element. Mr. Vasquez has over 16 years of diverse nuclear engineering experience covering all areas within the Engineering Mechanics field. His areas of specialties include: pipe stress analysis, pipe and equipment support analysis, pressure vessel design and analysis, seismic qualification of mechanical and electrical equipment, seismic margins assessment and fragility analyses, and fracture mechanics. Mr. Vasquez has lead responsibility for the seismic fragility analysis of the North Anna SPRA and served as working observer for the Peach Bottom SPRA peer review.

Mr. Douglas C. Rapp was the lead reviewer for the SPR technical element as well as for the configuration control. Mr. Rapp has nine years of experience in the nuclear industry and seven years of experience in the areas of PRA of FENOC. He is presently leading the FENOC Beaver Valley Power Station (BVPS) issuance of updated Internal Events PRA models for both units, acting as backup Supervisor for Analytical Methods (Fleet Design Engineering), mentoring new PRA analysts, supporting the BVPS NFPA 805 transition effort. He is the Project Manager and PRA Lead for the FENOC Fleet Other External Hazards PRA Project, the BVPS SPRA Project, and the BVPS Internal Flooding PRA Project. He is serving on the following: NEI Risk-Informed Operations Working Group, ASME/ANS JCNRM PRA Standard Working Group 3 (Internal Flooding), EPRI HRA Users Group, Riskman Technology Group, EPRI External Hazards Technical Advisory, and BVPS Maintenance Rule Expert Panel and Steering Committee. He has served on the NEI ROP Task Force, Riskman Technology Group Steering Committee (previous Chairman), and the PWROG Risk-Informed Regulations Core Team. Mr. Rapp has served as peer reviewer for the Watts Bar SPRA peer review and as Working Observer for the H.B. Robinson High Winds PRA peer review. He was assisted in the SPR technical element review by Mr. Robert Kirchner.

Mr. Kirchner has 28 years of experience in PRA with extensive experience in leading reliability and risk assessment projects and developing process improvement initiatives. He has covered virtually all areas of reliability engineering and risk assessment, including seismic and other hazard PRA for domestic and international plants.

Mr. Colter Somerville from Southern Nuclear Company supported the review of the SFR technical element as working observer. Any observations and findings that he generated were given to the PRT for their review and "ownership." As such, Mr. Somerville assisted with the review but was not a formal member of the PRT.

The PRT members met the peer reviewer independence criteria in NEI 12-13 [8].

A2.3. Summary of the Peer Review Conclusions

The PRT's assessment of the SPRA elements is excerpted from the DCPP SPRA peer review report [9] and summarized as follows. Where the PRT identified issues, these are captured in the PRT findings, for which the dispositions are summarized in Section A4 of this appendix.

A2.3.1. Seismic Hazard Analysis

As required by the ASME/ANS PRA Standard, the frequency of occurrence of earthquake ground motions at the site was based on a PSHA. To support the DCPP PSHA two major, site-specific senior seismic hazard advisory committee (SSHAC) Level 3 studies were conducted:

- Seismic Source Characterization for the Diablo Canyon Power Plant, San Luis Obispo County, California (March, 2015) [19], and
- Southwestern United States Ground Motion Characterization SSHAC Level 3, Technical Report (March, 2015) [20].

The DCPP PSHA is an excellent scientific study that is unique in a number of respects. The quality and uniqueness is founded in the cumulative benefits of decades of geologic, seismologic and geophysical site and regional investigations that have been carried out to support an understanding of the seismic hazard at the plant. PG&E has invested considerable resources and partnered with others (e.g., the United States Geological Survey (USGS)) to gather and evaluate data related to the seismic hazard potential at DCPP.

The data and interpretations from these studies result in the site being well characterized from a seismic hazard perspective. The seismic source characterization for the DCPP documents a SSHAC Level 3 study performed for DCPP[19]. This report summarizes the DCPP seismic, geologic, geophysical, and geotechnical database includes information from:

- The DCPP Final Safety Analysis Report (FSAR) Update [21] and the FSAR for the DCPP Independent Spent Fuel Storage Installation [22].

- The DCPP LTSP Final Report [23] and Addendum to the LTSP Final Report [24], including high- resolution and deep-penetration seismic reflection surveys, analysis of seismic reflection data obtained from others, mapping of marine terraces, logging and interpretation of trenches and natural exposures, 240 boreholes, examination of more than 300 water and oil well records, offshore rock samples, and drop core samples.

- The PG&E Central Coast Seismic Network consisting of 18 high-gain telemetered stations installed as part of the DCPP LTSP.

- PG&E/USGS Cooperative Research and Development Agreement studies, including onshore and offshore geophysical data and geologic mapping supporting the Shoreline fault zone study, integration of multiple offshore data sets, analysis of seismicity data, and analyses of aeromagnetic data.

- Geophysical data collected in response to California Assembly Bill 1632, including the report on the Central Coastal California Seismic Imaging Project [25].

- Ongoing DCPP LTSP studies, including acquisition and analysis of geodetic data, offshore seismic stratigraphy studies, and analyses of fluvial and marine terraces.

These data allow PG&E to characterize the center, body, and range of technically defensible interpretations better than at most sites in the United States and reduce the uncertainty that otherwise would need to be included in the PSHA and in the site response analyses. These programs also allow the DCPP staff to work with a wide range of experts in government, academia, and the private sector, and this leads to a better understanding of the seismic hazard at the DCPP site.

The SSHAC process (NUREG/CR-6372 [26] and NUREG-2117 [27]) satisfies the requirements of the ASME/ANS PRA Standard with respect to conducting a PSHA study, evaluation of aleatory variability and epistemic uncertainties, and the use experts. The SSHAC Level 3 studies included the establishment of a Technical Integration Team (TI Team) and the convening of three formal workshops in which the TI Team interacted with subject experts who served as resource and/or proponent experts on selected PSHA input issues (i.e., how the earthquake occurrence rate for fault seismic sources should be modeled, etc.). The SSHAC process is a structured approach for the evaluation of data, models and methods, and the integration of this information to represent the center, body, and range of technically defensible interpretations. In this manner the seismic source characterization and ground motion characterization models reflect a representation of aleatory variability and epistemic uncertainty in PSHA inputs.

The SSHAC Level 3 seismic source characterization and ground motion characterization projects included participatory peer review panels (PPRP) (one for each project). The PPRP followed each project from the planning stage, through each of the formal workshops, to review of final project documentation. In addition to attending the project workshops, the PPRP also observed many of the working meetings of the TI Team as part of understanding how the TI Team ensured the breadth of data, models, and methods were appropriately considered.

In addition to the PPRP observing the two SSHAC studies, other observers included the NRC staff and their consultants.

In March 2015, PG&E submitted PG&E Letter DCL-15-035, "Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding the Seismic Aspects of Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident: Seismic Hazard and Screening Report," [3]). As part of the response, PG&E provided an estimate of the site GMRS that was based on the results of the DCPP PSHA, including the seismic source characterization, ground motion characterization and site response inputs. As a result of its evaluation, the NRC staff concluded these parts of the PSHA were an acceptable basis to estimate the seismic hazard at the plant. It is worth emphasizing the NRC acceptance of the PSHA is a product of their ongoing observation of the SSHAC projects and their own independent evaluation of the final models.

For the seismic source characterization model, the first step involved the compilation of up-to-date data. In the case of the DCPP, this included the recognition that extensive work, carried out over several decades, has focused on local fault sources. The seismic source characterization model included "Primary and Connected" faults, assessments of uncertainty in fault geometry, fault slip rate, how different faults or fault segments might rupture together, slip rate allocation among faults, magnitude distribution, and the time-dependency of earthquake recurrence. Sensitivity analyses were used to inform the degree to which uncertainty was characterized. Epistemic uncertainties were modeled using a logic tree approach.

The ground motion characterization model was based on an extensive evaluation of existing published ground motion models and data, and integration of this information by the TI Team. The TI Team assembled an extensive set of empirical strong motion data and models, and supplemented this data with finite-fault simulations to better represent larger earthquakes at closer rupture distances where observations are limited. The ground motion characterization model is represented by a suite of alternative median and standard deviation models for a given seismic source. The median models account for style of faulting. The suite of median models was based on an extensive evaluation completed by the TI Team using the visualization approach referred to as Sammon's map representation. The suite of median models represents the epistemic uncertainty in median models; each individual model was assigned a weight based on consistency with data and from the distribution of the candidate published ground motion models.

The seismic source characterization and ground motion characterization inputs were used to derive seismic hazard curves for a reference rock site condition represented by a V_{S30} = 760 m/s (time-averaged shear wave velocity over the upper 30m of depth). The reference rock PSHA results were based on the concept of single-station sigma to represent ground motion aleatory variability. The reference rock PSHA results are presented in a suite of seismic hazard curves at different spectral frequencies including PGA. Because both the seismic source characterization and ground motion characterization models include a wide range of alternative inputs, there is a wide range of hazard curves. The PSHA quantifies the mean hazard and fractiles of hazard, typically represented by the 5th, 16th, 84th, and 95th fractile curves.

The reference rock PSHA results were modified to account for the effects of surficial geology (local site materials) based on site response analyses, which were completed using empirical and analytical approaches. The empirical site response analysis was based on earthquake recordings at two seismic stations located at the DCPP site. Recordings from two earthquakes were used to assess the path and source effects for each event, which were then used to derive site terms for each of the two stations. The available data is from lower ground motions, and thus may not reflect the site response at larger ground motions that are included in the PSHA model. Epistemic uncertainty in the site response was assessed taking into account the limited number of recordings and the low ground motion levels on which it is based. The use of empirical data avoids having to use an analytical site response model, which requires multiple parameters to be estimated and depends on several modeling assumptions. Recorded data at the plant site directly reflects the site surficial geology and material properties at depth that control how ground motions are modified relative to the reference rock assumptions.

In response to the review undertaken by the NRC of the original PG&E submittal [3], which was based on the empirical site response analysis only, an analytical site response analysis was performed following the guidance in the SPID [2] which recommends a 1D equivalent linear analysis. The analytical site response included the derivation of three base-case shear-wave velocity profiles for the general site, and for several structure-specific locations. The DCPP site profiles were developed from a 3D compression-wave velocity model that was derived from inversions of active seismic data collected at the site, which were then used to create a 3D shear wave velocity model. The analytical site response accounted for epistemic uncertainties in site profile, site material properties (shear modulus and damping versus shear strain), and upper crustal damping as reflected in the ground motion modeling parameter kappa. Consistent with the guidance in the SPID, analytical amplification factors were derived for a wide range of input ground motions. The DCPP site response was updated based on a combination of the empirical and analytical methods, and submitted to the NRC in PG&E Report "Response to NRC Request for Additional Information dated October 1, 2015 and November 12, 2015 Regarding DCPP Seismic Hazard and Screening Report" [4].

Review of the site response analysis identified three issues that need to be addressed. First, implementation of the analytical approach in addition to the empirical approach appears to have reduced the epistemic uncertainty in the estimate of the seismic hazard. PG&E should clarify and confirm their technical support for this outcome. Second, given that extensive site studies have characterized the site very well, shear-wave velocity profiles generated to represent the aleatory variability in site velocity properties should be compared to the 3D interpreted velocity model to demonstrate that the suite of randomized profiles is reasonable. Third, implementation of Approach 3 to combine the reference rock seismic hazard with the site response should be reviewed to confirm that it results in the appropriate distribution of seismic hazard at the control point.

The spectral shape used in the SPRA reflects the PSHA results for the reference rock condition using the SWUS/DCPP ground motion characterization model as modified by the results from the site response analysis. The reference rock spectral shape appropriately reflects the dominant seismic sources and earthquake magnitudes and distances that contribute to ground motions at the site, given the set of ground motion models developed as part of the SSHAC Level 3 ground motion characterization model. The reference rock spectral shape gets modified as a result of the site response, as indicated by both the empirical and the analytic site response analyses. The DCPP site GMRS and FIRS are generated from the control point UHRS with 1E-4 and 1E-5 mean annual frequency of exceedance and the associated design factors consistent with NRC Regulatory Guide 1.208 [28]. The relative change in spectral shape between the reference rock site condition and the DCPP control point site condition is consistent with the change in spectral shape expected going from a V_{S30} = 760 m/s to the stiffer (higher V_{S30}) DCPP site condition. The spectral shapes for the vertical GMRS and FIRS are developed using an appropriate relationship for deriving a spectral frequency-dependent vertical-to-horizontal ratio function and using that to adjust the horizontal GMRS and FIRS.

As part of the assessment of seismic hazards, the ASME/ANS Standard requires evaluation of other seismic hazards be considered. The review determined the assessment was limited in scope and failed to adequately evaluate the potential threat to plant SSCs.

Lastly, the ASME/ANS Standard places considerable importance on documentation of the seismic hazard analysis. The documentation must be sufficiently detailed to support PRA applications, peer review, and future model updates. In this respect, the DCPP PSHA has not met the documentation requirements of the ASME/ANS PRA Standard; documentation is missing and supporting calculations are, in places, incomplete and require editorial attention.

A2.3.2. Seismic Fragility Analysis

Seismic response analysis was performed using the appropriate GMRS/FIRS derived from the PSHA as the input for structures housing the SSCs. A qualitative evaluation was made for a higher reference level earthquake that was 1.63 times the GMRS, corresponding to 50 percent cumulative SCDF (based on preliminary quantification results). PG&E judged that the seismic responses of the top contributors to the SCDF and SLERF would not be significantly different at the higher ground motion levels considering the higher structural damping. It was also observed that the shapes of the UHRS with mean annual frequencies of exceedance of 1.0E-04 and 1.0E-05 (used to develop the GMRS) are similar in the frequency range of 1 to 10 Hz).

Probabilistic response analysis was performed for the containment structure, auxiliary building, and turbine buildings, using the guidance provided in NUREG/CR-2015 [29] and ASCE 4-13 [30]. New finite element structural models were developed and used in the development of structural responses. LHS approach was used with 30 trials, the minimum number specified in ASCE/SEI Standard No. ASCE 4-13 [30] to achieve stable responses. The variables included: structural stiffness, damping, soil/rock profiles and ground motions (time histories). The median 50 percent and 84 percent NEP spectral accelerations were determined. Forces and moments for the determination of fragilities of major structural components were also based on the probabilistic seismic response analyses.

SSI analysis was performed for the containment structure, auxiliary building, and turbine buildings. The effects of structure-soil-structure Interaction (SSSI) between the major structures (containment structures, auxiliary building, and turbine building) were judged to be insignificant based on the results from the DCPP LTSP evaluations. However, the SSSI effects on the seismic ground motion input to RWST, given its proximity to the auxiliary building, was determined to be significant.

Seismic walk-downs were performed and the results of the walk-downs were utilized in the fragility evaluations. Walk-downs results informed fragility evaluations with respect to anchorage and load paths, functional considerations, and seismic interactions (spatial, seismic induced fire, and flood). Walk-downs were documented on EPRI Report No. NP-6041-SL [11] SEWS.

Fragilities were calculated using a SOV approach for all but two components, other than those deemed rugged or robust. The exceptions being the 230 kV offsite power systems and firewater piping system in the auxiliary building, that were evaluated using earthquake performance data. Fragility values were expressed in terms of the 5 percent damped spectral acceleration at the Control

Point GMRS (Elevation 85 ft. of the reactor building foundation). PG&E leveraged on an ongoing engineering project for the qualification of the Nuclear Steam Supply System (NSSS) and derived detailed fragilities for the SPRA. This practice is rather unique and PG&E should be commended for it. The plant leveraged extensive site-specific qualification data for evaluation of risk significant SSCs including relays and those contributing to SLERF sequences.

Considering the use of plant specific PSHA based seismic response spectra, development of new 3D finite element building models, use of probabilistic seismic response analysis, detailed walk-downs, and the use of SOV approach to derive fragility values for all but two SSCs of the SEL (other than rugged and robust components). The PRT judged that the fragility values used in the SPRA are indeed realistic.

In general, critical failure modes were identified and evaluated for SSCs based on input from design documents and plant walk-downs. However, as documented in review of the SHA I-2, slope stability requires additional review and disposition. In addition, the PRT concluded that in some cases, no effort was consistently made to revisit other failure modes for SSCs (e.g., seismic spatial interactions) when high median acceleration capacities were calculated to ensure median capacities for those other failure modes were not lower than those calculated for what was identified to be controlling. On this basis SFR D-2 was considered not met.

Documentation of the structural seismic response analysis, walk-down, and fragility evaluations was organized in a hierarchical manner. The top-level requirements were included in a Criteria Document (PG&E Document No. 128027-CD-01 [32]). The next level included summary reports for the structural response analysis (PG&E Report No. 128027-R-02 [33]), walk-down (PG&E Report No. 128027-R-01 [35]), and fragility evaluations (PG&E Report No. 128027-R-03 [34]). The third level included the detailed calculations.

A2.3.3. Seismic Plant Response Analysis

The DCPP SPRA linked event tree model was systematically developed from the at-power, internal events PRA systems models and logic structures using overlaying seismic failure modes. The seismic pre-tree provides the integration of the seismic hazard curves, seismic fragilities, and fragile component correlations. 16 initiating events arranged in seismic magnitude bins ranging from 0.1g to 9g, 140 fragile components, and 39 fragile top events demonstrates the comprehensive scope, level of detail, and significant analysis that has been completed in the DCPP SPRA model.

The DCPP SPRA systems model included seismic-caused initiating events and other failures including seismically-induced SSC failures, non-seismically-induced unavailabilities, and human errors that give rise to significant accident sequences and/or significant accident progression sequences. The hazard spectrum explicitly modeled begins at 0.1g (5 percent damped spectral acceleration at 5 Hz), which is bounded by the Design Earthquake (DCPP's Operating Basis Earthquake- equivalent) as defined in the DCPP FSAR [21], which has a PGA of 0.2g (0.7g 5 percent damped spectral acceleration at 5 Hz). Initiating event definitions, event tree logic, split fraction rules, and end states were "walked" through the entire linked-event tree quantification for a few select, significant accident sequences. This review focused on accident sequence development and contributors to split fractions, including the inclusion of appropriate fragilities and random failures. The accident sequence logic is judged to be appropriate and correctly represents plant success criteria.

The DCPP SPRA model has adapted the at-power, internal-events PRA systems models to incorporate seismic-analysis aspects which generally satisfy the criteria of the ASME/ANS PRA Standard [7] Part 2. The SPRA systems model included seismic impacts on performance shaping factors for the control room and ex-control room post-initiator actions, the addition of post-earthquake recovery actions, an assessment of local operator pathways to identify potential vulnerabilities, full correlation between identical SSCs located on the same elevation of the same building, and implicit modeling of relay chatter. Some issues with VSLOCA modeling and walk-down for seismically-induced fires and floods resulted in two NOT MET assessments.

Reasonable controls in place to assure the as-built, as-operated plant are being reflected in the DCPP, e.g., the DCPP PRA configuration control procedures. The significance of some noteworthy conservatism, including mitigation of ATWS at low seismic levels as well as the treatment of relays, is not fully characterized and the level of distortion in results, while judged minor, must be determined. A more complete listing of sources of model uncertainty and related assumptions would be beneficial towards identifying any further potential model conservatisms.

The development of the SEL is comprehensive and includes the appropriate treatment of relevant inputs. The SEL Notebook (PG&E Calculation No. F.6.1 [36]) includes explicit treatment of full power initiating event (FPIE) SSCs, SSCs included in the fire PRA, FPIE Flooding Sources, annunciators, tanks, walls, relays which can chatter, and the SELs of similar industry SPRAs. While some moderately significant completeness issues were identified, the overall development of the SEL is adequate to support the SPRA.

The DCPP SPRA model appropriately integrates the hazard, fragilities, and the systems analysis model to quantify core damage frequency and large early release frequency. Full correlation was used for identical components within the same system located on the same elevation to capture the significant correlations that affect the results. Plant-specific correlation values were not used. Overall, the quantification process (including uncertainty analysis) used for the SPRA is judged to be appropriate and MET most of the ASME/ANS PRA

Standard [7] Part 2 requirements. Findings were established to address fragility cutoff at the HCLPF capacity and correlation of all bottom-mounted instrument tubing as an excessive LOCA.

With some minor exceptions, the DCPP SPRA documents were assessed adequate to facilitate PRA applications, upgrades, and peer review.

A2.3.4. PRA Configuration Control (SMU)

The DCPP PRA Configuration Control Program was assessed for the following key elements:

- a process for monitoring PRA inputs and collecting new information;
- a process that maintains and upgrades the PRA to be consistent with the as-built, as operated plant;
- a process that ensures that the cumulative impact of pending changes is considered when applying the PRA;
- a process that maintains configuration control of computer codes used to support PRA quantification; and
- documentation of the Program.

The assessment is based on Section 1.5 of the ASME/ANS PRA Standard [7]. Five maintenance and update (MU) high-level requirements (HLR A-E) and nine SRs (SRs MU A1-A2, B1-B4, C1, D1, and E1) are defined in Section 1.5 of the ASME/ANS PRA Standard [7]for assessing the five key elements above. The MU assessment is now a standard assessment for every SPRA Peer Review.

The HLR and SR assessments were completed during the DCPP SPRA Peer Review. The following two DCPP procedures establish the basis for this assessment:

- Departmental Administrative Procedure No. TS3.NR1 [66]
- Administrative Work Procedure AWP No. E-028 [67]

Limited, related DCPP procedures were also reviewed to ensure the PRA configuration control process includes monitoring of changes in design, operation, and maintenance that could affect the PRA, (e.g., Interdepartmental Administrative Procedure No. CF3.ID9 [68]). Other submitted DCPP PRA procedures were not reviewed for configuration control as they were controlling documents for specific programs. Finally, PRA configuration control evidence was reviewed during the DCPP SPRA peer review, (e.g., Software Quality Assurance Plan for Riskman for Windows, PRA Tracking Database, etc.).

DCPP PRA configuration control procedures adequately address: the monitoring of PRA inputs and the collection of new information, the maintenance and upgrade of the PRA model(s) of record, the evaluation of cumulative impacts of pending changes on risk applications, the control of computer codes used to support PRA quantification, and the documentation for PRA configuration control. The health of the PRA configuration control process was demonstrated by various sources of evidence at the peer review, including the PRA Action Tracking Database, the SAP Notification System, formal Software Quality Assurance plans for RISKMAN and HRA Calculator, design review for PRA coordination, cumulative impact assessments, and inherent ties of PRA to the LTSP. Numerous improvements were identified to clarify and enhance the procedures.

No findings and suggestions defined any underlying major issues with the DCPP PRA configuration control program. All findings and suggestions center about updating, clarifying, and expanding for completeness the two PRA configuration control procedures. The findings address the clarity and completeness of the DCPP PRA configuration control across multiple facets of the program including:

- cumulative impacts of pending plant changes or model improvements on all in-progress risk-informed applications,
- use and control of PRA software and hardware,
- design change process integration with PRA, and
- qualifications and roles and responsibilities of PRA personnel.

The suggestion focuses on the consideration of conducting PRA data updates every second refueling cycle (rather than every fourth refueling cycle) to better reflect DCPP operational performance, and better support DCPP's future risk-informed applications.

A2.3.5. Peer Review Findings

The DCPP SPRA was peer reviewed against the requirements of the ASME/ANS PRA Standard [7] following the review process described in NEI-12-13 [8]. A number of comments and observations have been provided to the DCPP SPRA team. Because of the nature of PRA peer reviews and the process documented in NEI-12-13 [8], such comments and observations are to be considered as examples and, when and as appropriate, they should be investigated for extent of condition and for any systematic approach used in the development of the SPRA.

The limitation observed in the other (i.e., non-vibratory) seismic-induced hazards, when addressed, is currently not expected to result in a change to the plant risk profile from what is currently documented. The limitations observed in the assessment of a complete set of failure modes for some of the fragilities may reduce the actual fragility value used for some of the components and potentially change SCDF and SLERF. Preliminary sensitivities performed during the peer review to address questions on these items indicate that the increase in SCDF and SLERF should be minimal. A more systematic evaluation of seismic-induced fires and floods may result in additional scenarios to be explicitly modeled, which can also change the SCDF and SLERF of the plant. Other minor limitations and suggestions are discussed in the Appendices of peer review report [9].

In summary, the PRT concluded that the DCPP SPRA meets most of the ASME/ANS PRA Standard requirements. The amount of data available for the site results in a seismic hazard evaluation that is unique in the industry and well beyond the state of practice at any other U.S. site. The fragility analysis is performed in a fully probabilistic manner and on a site specific and plant specific basis, using abundant testing and qualification information as well as NSSS specific information that is also well beyond the state of practice in the industry. As such, the review team concluded that the DCPP SPRA realistically reflects the seismic risk profile of the plant, with no evident bias or conservatism. Uncertainties are appropriately characterized in the seismic hazard, fragility, and plant response modeling technical elements. As a result of the above, the DCPP SPRA is judged by the review team to be technically adequate for supporting risk-informed applications and risk-informed decision making.

However, the PRT identified specific areas for improving the technical adequacy of the SPRA. These areas are documented as F&Os. At the conclusion of the Peer Review, there were 47 open finding-level F&Os "findings", distributed between the SHA, SFR, SPR, and SMU areas.

A3. Revision of SPRA Model and Documentation

Following the peer review, the DCPP SPRA model and documentation were updated to address each of the 47 findings. In addition, PG&E generated closure documentation for each of the findings from the peer review against the ASME/ANS PRA Standard [7] of the DCPP SPRA.

Subsequently, the updated DCPP SPRA model and documentation were subjected to an independent assessment for the closure of the findings. This assessment is described in Section A4.

Following the independent assessment for the closure of the findings, the DCPP SPRA was once again updated to incorporate other optional changes recommended by the independent assessment team (IAT). These included de-correlation of containment isolation valves and other changes to improve realism in the model.

A4. Finding Closure by Independent Assessment and Focused Scope Peer Review

In order to close-out the 47 findings of record from the DCPP SPRA peer review [9], an independent assessment of PG&E's resolution of each finding was performed. In addition, concurrent focused-scope peer reviews were performed, where necessary, for the close-out of a specific finding which was identified as a PRA Upgrade. 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 [37], which has been accepted by NRC, with two conditions (NRC Letter dated May 3, 2017 ([38]):

(i) Use of New Methods: *A PRA method is new if it has not been reviewed by the NRC staff. There are two ways 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 the 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).*

(ii) 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 in 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 DCPP license amendment request submittals, other regulatory interactions, risk-informed applications, and risk-informed decision making. Finding resolutions 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 not be carried forward nor discussed in such future activities.

A4.1. Selection of Independent Assessment Team Members

The IAT consisted of six team members, all of whom have extensive qualifications and many years of experience in the pertinent areas of SPRA and peer review. Since the 47 findings were distributed between the SHA, SFR, and SPR (including SMU) areas, 2 team members were selected for each review area. Note that the members of the IAT also participated in the concurrent focused-scope peer reviews.

Team member qualifications were reviewed by the team lead and by PG&E to assure consistency with the requirements defined in Section 1-6.2 of the ASME/ANS RPRA Standard [7] and the NEI review guidance (Appendix X of NEI 12-13 [37]), including the NRC's clarification on Appendix X (NRC Letter dated May 3, 2017 [38]). Reviewer independence was established, approved, and documented in the independent assessment report [39]. The overall IAT experience is such that there were two qualified reviewers for each F&O. Ms. Sara Lyons, a process observer from the NRC, attended the review but was not involved in the independent assessment or focused-scope peer reviews.

The names of the members of the IAT, their professional affiliations, and their review responsibilities are summarized in Table A-1. The resume for each team member is included in Appendix F of the independent assessment report [39].

- Dr. Andrea Maioli was the team lead for the initial peer review of the DCPP SPRA in June 2017. See Section A2.2 for a summary of Dr. Maioli's qualifications.

- Dr. Gabriel Toro has over 35 years of experience geological and seismic hazard studies, most recently as a Senior Principal Engineer with Lettis Consultants International, Inc., an Earth Science firm specializing in the assessment of geologic and seismic hazards for site-critical facilities. Dr. Toro has extensive experience in the development of probabilistic models for natural hazards, particularly earthquakes, working in close cooperation with earth scientists that are experts in the natural phenomenon being evaluated. His experience includes participation in TI Teams, resource expert roles, and PPRP for SSHAC ground motion studies, support of the SPRA studies for Vogtle Units 3 & 4, and performance of seismic hazard evaluations for numerous nuclear plant sites, and other facilities in the United States and throughout the world.

- Dr. Robert Youngs has over 35 years of consulting experience, with primary emphasis in hazard and decision analysis, most recently as a principal engineer with AMEC Foster Wheeler. He has pioneered approaches for incorporating earth sciences data, and their associated uncertainties, into probabilistic hazard analyses. The focus of this work has been on developing quantitative evaluations of hazard by combining statistical data and expert judgment. Dr Youngs has extensive experience performing studies for nuclear facilities, which include PSHAs and ground motion studies for Washington Public Power Supply System's facilities at Hanford and Satsop, Washington; Portland General Electric's Trojan Power Plant; PG&E's facilities at DCPP and Humboldt Bay; U.S. DOE nuclear facilities at Hanford, Idaho National Engineering ,the

Environmental Laboratory, Savannah River, Rocky Flats, Los Alamos National Laboratory, and the Nevada Test Site, the proposed commercial nuclear waste repository at Yucca Mountain, Nevada, and applications for new power plants in Illinois, North Carolina, Florida, Michigan, and Louisiana. He is a member of the SSHAC Level 3 TI Teams developing updated seismic hazard assessments for the Hanford Washington DOE site and the SWUS project developing ground motion models for post-Fukushima response for the DCPP and Palo Verde nuclear power plants.

- Dr. Ram Srinivasan participated in the initial peer review of the DCPP SPRA in June 2017. See Section A2.2 for a summary of Dr. Srinivasan's qualifications.

- Mr. Daniel J. Vasquez participated in the initial peer review of the DCPP SPRA in June 2017. See Section A2.2 for a summary of Mr. Vasquez's qualifications.

- Mr. Ken Kiper is currently a Technical Manager at Westinghouse Electric Company, LLC after a 31-year career at Seabrook Station. He has experience in virtually every aspect of PRA modeling and applications, including upgrading and maintaining the Seabrook SPRA.

Table A-1 - IAT Member Names, Affiliations, and Responsibilities

Name (affiliation)	**Responsibilities**
Andrea Maioli(1) *(Westinghouse Electric Company, LLC)*	**Team Lead.** Lead Reviewer for F&Os 26-15, 26-19, 26-22, 26-23, 26-24, 26-26, 27-1, 27-3, 27-7, 27-9, 27-11, 27-12 Lead Reviewer for SR SPR-A1, SPR-B9
Gabriel Toro(2) *(Lettis Consultants International)*	Lead Reviewer for F&Os 20-4, 20-5, 20-6, 20-11 Lead Reviewer for SR SHA-I2
Robert Youngs(2) *(AMEC Foster Wheeler)*	Lead Reviewer for F&Os 20-3, 20-7, 20-8, 20-9 Support Reviewer for SR SHA-I2
Ram Srinivasan(1) *(Independent Consultant)*	Lead reviewer for F&Os 23-1, 23-2, 24-41, 24-2, 24-3, 24-4, 24-5, 24-9, 25-1, 25-5 Lead Reviewer for SR SFR-D1, SFR-A2 Support reviewer for SR SFR-D2
Daniel J. Vasquez(1) *(Dominion)*	Lead reviewer for F&Os 26-4, 26-8, 25-2, 25-4, 25-6, 25-7, 25-9 Lead Reviewer for SR SFR-D2 Support Reviewer for SR SFR-D1, SFR-A2

Table A-1 - IAT Member Names, Affiliations, and Responsibilities	
Name (affiliation)	**Responsibilities**
Kenneth Kiper(2) *(Westinghouse Electric Company, LLC)*	Lead Reviewer for F&Os 26-1, 26-5, 26-7, 26-17, 26-18, 26-20, 26-21, 26-25, 27-13, 27-17 Support Reviewer for SR SPR-A1, SPR-B9

Notes:

(1) Participated in the original June 2017 peer review of the DCPP SPRA [9].

(2) Did not participate in the original June 2017 peer review of the DCPP SPRA [9], and, thus, was not responsible for any of the HLRs described in Appendix A of the independent assessment report [39], the SRs in Appendix B of the independent assessment report, and the F&Os in Appendix C of the independent assessment report, except as covered by the Independent Assessment and Focused Scope Peer Review.

A4.2. Pre-Review Activities

In preparation for the independent assessment associated with the close-out of the findings, PG&E and the IAT performed the activities described in the following subsections.

A4.2.1. Host Utility Preparation

PG&E provided the complete and relevant review material to the IAT on Friday, October 20, 2017, which was more than two-weeks in advance of the start of the review (Tuesday, November 7, 2017) to allow the reviewers to prepare and conduct a more efficient technical review. As input to the review, PG&E provided the following documentation:

- A PG&E-prepared written summary for each finding, which included:
    - The exact wording of the original findings from the SPRA peer review [9], which included the following:
        - Identification of the SRs impacted for which the finding was written against, including the SRs that reference another SR
        - A description of why the SR was not met
        - A summary description of how each finding was dispositioned
        - A self-assessment on whether the finding closure involved an upgrade or a maintenance activity, based on the definitions of

upgrade and maintenance documented in the ASME/ANS PRA Standard [7]

- A list of documents that were revised to reflect the above mentioned activities

- Copies of each PG&E document associated with the disposition of the finding (e.g., calculations, reports, SPRA model)

A4.2.2. Off-Site Reviews

The IAT started the review and familiarization of the documentation provided by PG&E prior to the commencement of the on-site review.

A4.3. On-Site Review and Consensus

The DCPP SPRA finding close-out independent assessment, including the concurrent focused-scope peer reviews, was performed at the PG&E General Office in San Francisco, CA, on November 7 and 8, 2017. This independent assessment process was conducted as described in the following subsections.

A4.3.1. Conduct of Reviews

During the on-site review and consensus session, the IAT performed a review of the bases for closure of each of the findings documented by PG&E. The DCPP PRA team, composed of PG&E staff members directly involved in the development of the closure basis documents, and supporting consultants, were present at the PG&E General Office during the duration of the independent assessment for interactive questions and answers. Access to the DCPP PRA model was also provided on a computer with the required software. Since the closure of none of the findings required access to DCPP, an on-site visit was not required.

A lead reviewer and supporting reviewer were assigned for each technical element. The lead and supporting reviewer associated with each finding made the initial determination regarding adequacy of resolution of each finding within their scope.

The IAT reviewed the DCPP SPRA Peer Review Report [9] and the associated SRs from the ASME/ANS PRA Standard [7] in order to assure a good understanding of the findings from the initial peer review. The IAT decided if the findings had been adequately addressed with appropriate and acceptable assumptions, and evaluated if the relevant changes had been incorporated into the PRA and appropriate plant configuration programs to ensure that the model represents the as-built, as-operated plant. Based on these assessments, the team determined if the finding could be closed-out via consensus, referencing the appropriate SRs of the ASME/ANS PRA Standard for the review criteria.

The relevant PRA documentation was reviewed to assure that it had been completed and appropriately incorporated into the PRA model and other supporting documentation prior to closing the finding. In cases where the original finding identified error(s), the IAT verified, via a sampling review, that identified errors, including those specifically described in the original peer review, are fixed throughout the model.

A consensus process was followed during which the full team present on the day of the associated consensus session considered and reached the following:

- Consensus on the status of the finding (i.e., CLOSED, OPEN or PARTIALLY CLOSED). This conclusion was reached through a review of the original basis and description of the finding and on the technical work and documentation provided by PG&E to resolve the originally identified issue.

- Consensus on whether the activities performed to close the finding are to be considered maintenance or upgrade, per the appropriate definition in Non-Mandatory Appendix 1-A, "PRA Maintenance, PRA Upgrade, and Advisability of Peer Review," of the ASME/ANS PRA Standard [7].

- If the finding was associated with an SR that was originally judged as Not Met or Met at Capability Category less than II, upon confirming closure of the associated finding(s), the SR has been re-assessed to reach consensus on whether the intent of the SR is now Met or Met at Capability Category II or higher.

Note that prior to the independent assessment, PG&E self-identified that the PRA changes associated with the dispositions of three of the findings constituted PRA upgrades and during the course of the on-site review, the IAT determined that the PRA changes associated with the disposition of one additional finding constituted a PRA Upgrade. The findings for which their disposition constituted a PRA Upgrade are identified in Table A-2. The IAT conducted concurrent focused-scope reviews during the on-site review to assess the adequacy of each PRA Upgrade, following the applicable guidance from the main body of NEI 12-13 [8] and the ASME/ANS PRA Standard [7]. This process requires the performance of a re-peer review for each SR associated with the finding and allows for the possible generation of new F&Os during this review. However, the focused scope peer reviews for DCPP did not identify any new F&Os.

A4.3.2. Treatment of "New Methods"

All of the changes to the DCPP SPRA were classified as either PRA Maintenance or PRA Upgrade by the IAT. Therefore, no new methods were identified during the independent assessment.

A4.3.3. Use of Remote Reviewers

All members of the IAT were present for the portions of the independent assessment activities associated with their area of review. Therefore, remote reviewers were not utilized for the performance of the independent assessment.

A4.3.4. Status of Findings at End of On-Site Review

At the end of the on-site review, the statuses of the 47 findings from the initial peer review were:

- Closed: 40
- Partially Closed (requiring additional documentation): 4
- Open: 3

The close-out of the seven remaining findings was subsequently addressed during the post-review activities, described in Section A4.4.1, below.

A4.4. Post-Review Activities

Subsequent to the completion of the on-site review, the independent assessment included the activities described in the following subsections.

A4.4.1. Closure After the On-Site Review

As indicated in Section A4.3.4, the disposition of 4 findings required additional documentation for close-out and 3 remained open.

During the time between the end of the on-site review (November 8, 2017) and the finalization of the IAT report, PG&E prepared updates to the documentation associated with the resolution of these remaining findings (e.g., descriptions of the basis for close-out, revised calculations, and revised reports) and incorporated this information, where appropriate, into the DCPP SPRA Model of record. This information was transmitted to the IAT for their review.

The IAT performed a re-review of the remaining findings, using the information provided by PG&E, to determine the adequacy of the resolution. The IAT performed consensus sessions in support of their re-review. At the completion of this process, the IAT determined that all remaining findings had been adequately resolved and were assigned a status of closed.

A4.4.2. Final Report

At the end of the independent assessment, the IAT provided a final report, documenting all the activities and results [39]. This report included the following information:

- Descriptions of the F&O independent assessment process - see Section 2 of [39].
- Description of the scope of the independent assessment (i.e., identification and description of the findings being reviewed for closure) - see Table 1-1 of [39].
- Identification of the SRs which the findings were written against, including SRs that reference another SR. Include the basis for the SR assessment from the peer review of record - see Table 1-1 and Appendix C of [39].
- 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 or inadequacy for closure of each finding in relation to the affected portions of the associated SR. If multiple SRs are referenced by a single finding, the affected portions of all associated SRs were addressed - see Appendix E of [39].
- 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 - see Appendix E of [39].
- No new significant issues were identified by the IAT that are directly related to the findings being closed; so there were none included in the report.

The report categorized each in-scope finding as "closed." For each finding, the basis for the decision on closure was documented.

The final report included each of the IAT members' resumes and summary of their experience as it applies to qualification guidelines of NEI guidance documents and the ASME/ANS PRA Standard - see Appendix F of [39].

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

A4.5. Summary of Independent Assessment Team Conclusions

For all of the findings that were in the scope of the independent assessment, the IAT concurred that they can be considered closed.

As a result of the closure of the associated findings, six SRs, that were judged to be not met in the original peer review [9], were judged to be now met.

Four of the findings were judged to be closed with an upgrade, which originated in a focused scope peer review of the two associated SRs. Both the SRs were reassessed as met with no additional F&Os. None of the upgrades involved the use of a new method.

The IAT concluded that, as a result of the closure of the associated findings, the DCPP SPRA more realistically reflects the current seismic risk at the site and identified no evident conservatism or bias that would question the technical adequacy of the SPRA in support to risk-informed decision making. Table A-2 presents a summary of the SRs graded as not met or not Capability Category II during the June 2017 peer review and the current disposition for each.

Table A-2 - Summary of SRs Graded as Not Met or Capability Category I for Supporting Requirements Covered by the DCPP SPRA Peer Review

SR	Assessed Capability Category	Associated Finding F&Os	Disposition to Achieve Met or Capability Category II
SHA			
SHA-I1	Not Met	20-4	F&Os have been resolved utilizing the process given in Appendix X of NEI-12-13 [37]. SRs are judged to be Met.
SHA-I2	Not Met	20-5(1), 20-6(1), 20-11(1)	
SHA-J1	Not Met	20-8	
SFR			
SFR-D2	Not Met	26-15, 27-11(2)	F&Os have been resolved utilizing the process given in Appendix X of NEI-12-13 [37]. SR is judged to be Met.
SPR			
SPR-B8	Not Met	26-19	F&Os have been resolved utilizing the process given in Appendix X of NEI-12-13 [37]. SRs are judged to be Met.
SPR-B9	Not Met	26-15, 27-11[4]	

Notes:

(1) The disposition of this F&O was self-identified as a PRA Upgrade by PG&E.

(2) The disposition of this F&O was identified as a PRA Upgrade by the IAT.

A4.6. Compliance of Independent Assessment with NRC Conditions

As indicated in Section A4, the NRC's acceptance of the F&O closure process described in Appendix X to NEI 12-13 [37], as documented in [38], includes two conditions. The independent assessment of the finding closure for the DCPP SPRA satisfies the conditions as follows:

(i) Use of New Methods: As indicated in Section A4.3.2, new methods were not employed in the resolution of the finding associated with the DCPP SPRA. Therefore, this condition does not apply.

(ii) Use of Appendix X in Its Entirety: The finding closure process, as outlined in Sections A4.1 through A4.4.2, 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 DCPP SPRA peer review is in conformance with the NRC's requirements.

A5. Summary of SPRA Capability Relative to SPID Tables 6-4 through 6-6

The PWROG performed a peer review of the DCPP SPRA in June 2016. The SPRA was peer reviewed relative to Capability Category II for the full set of requirements in the AMSE/ANS PRA Standard [7]. Table A-1 provides a summary of the disposition of SRs judged by the PRT to be not met, or not meeting Capability Category II.

After the completion of the subsequent independent closure assessment in November 2017, which utilized the process given in Appendix X of NEI 12-13 [37], the full set of SRs were met and all findings have been closed.

A6. Identification of Key Assumptions and Uncertainties Relevant to the SPRA Results

The ASME/ANS PRA Standard [7] 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 [17] and EPRI Technical Report No. 1016737 [18] provide guidance on assessment of uncertainty for applications of a PRA. As described in [17], 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 DCPP SPRA model quantification (see Section 5 of this submittal).
- Modeling uncertainties are considered in both the base internal events PRA and the SPRA. Assumptions are made during the PRA development as a way to address a particular modeling uncertainty because there is not

a single definitive approach. Plant-specific assumptions made for each of the DCPP SPRA technical elements are noted in the SPRA documentation that was subject to peer review, and a summary of important modeling assumptions is included in Section 5.

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

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

Table A-3 - Summary of Potentially Important Sources of Uncertainty

PRA Element	Summary of Treatment of Sources of Uncertainty per Peer Review	Potential Impact on SPRA Results
Seismic Hazard	The DCPP SPRA PRT 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 DCPP site.	The seismic hazard reasonably reflects sources of uncertainty.
Seismic Fragilities	DCPP fragilities were obtained by SOV method with site specific uncertainties. See Sections 5.7.12 through 5.7.14 for relevant sensitivities.	The insight gained is that the fragilities are generally realistic and reasonable.
SPRA Model	The PRT identified [9] two documentation related issues associated with the uncertainty analysis. The first issue was associated with a lack of detail in the description of the parametric analysis. The other was that a complete listing of sources of modeling uncertainty and related assumptions was not documented.	As part of the finding closure process, PG&E updated documentation of the parametric analysis to include a detailed discussion of analysis inputs as well as insights obtained. Also, a more complete listing of model assumptions was documented and reviewed to help identify potential sources of uncertainty. More than 40 sensitivity cases were performed based, in part, on this list of

Table A-3 - Summary of Potentially Important Sources of Uncertainty		
PRA Element	**Summary of Treatment of Sources of Uncertainty per Peer Review**	**Potential Impact on SPRA Results**
		model assumptions. A summary of these sensitivity cases is provided in Section5.7

A7. Identification of Plant Changes Not Reflected in the SPRA

Except for the planned modifications to the supply and exhaust ducts associated with the 480V switchgear room ventilation system (see Table 6-1), the SPRA model reflects the as-built configuration of the plant as of the cut-off dates for the SPRA, which were:

- Unit 1: End of Refueling Outage 1R19 (November 7, 2015)
- Unit 2: End of Refueling Outage 2R19 (June 2, 2016)

These planned modifications to the 480V ventilation system were included in the DCPP SPRA for this submittal in order to support the Seismic Mitigating Strategies Assessment. See Section 5.7.13 for the results of a sensitivity study relating to the risk impact associated with these planned modifications.

Table A-3 lists significant plant changes implemented subsequent to the cut-off dates and provides a qualitative assessment of the likely impact of those changes on the SPRA results and insights.

Table A-3 - Summary of Significant Plant Changes Since SPRA Cut-off Dates	
Description of Plant Change	**Impact on SPRA Results**
Replacement of Unit 1 Nuclear Instrumentation Regulating Transformer Output Circuit Breakers.	The initial assessment indicates that the replacement of the output circuit breakers does not result in a reduction of the fragility for this component. Therefore, there is no impact on the SPRA results.
Replacement of Unit 2 Nuclear Instrumentation Regulating Transformer Output Circuit Breakers.	The initial assessment indicates that the replacement of the output circuit breakers does not result in a reduction of the fragility for this component. Therefore, there is no impact on the SPRA results.
Replacement of cooling coils in Unit 1 Containment Fan Cooler No. 1-5.	Fragility of the containment fan coolers is governed by capacity of anchorage welds, not the cooling coils, so replacement will not impact fragility parameters used in SPRA.

Table A-3 - Summary of Significant Plant Changes Since SPRA Cut-off Dates	
Description of Plant Change	**Impact on SPRA Results**
Installation of new 480V DC power boards and batteries associated with the FLEX communication system in the auxiliary building.	The FLEX communication system is not in-scope for the SPRA, but the weight associated with the additional power boards and batteries affects the seismic evaluation of the auxiliary building. An assessment indicates that the impact on the auxiliary building fragility is insignificant and will not impact the SPRA.
Installation of new security defensive positions in the turbine building.	The security defensive positions are not in-scope for the SPRA, but the weight associated with the defensive positions affects the seismic evaluation of the turbine building. An assessment indicates that the impact on the turbine building fragility is insignificant and will not impact the SPRA.
Increase in the capacity of the new fuel assembly storage racks in the auxiliary building.	The new fuel assembly storage racks are not in-scope for the SPRA, but the weight associated with the additional fuel assemblies affects the seismic evaluation of the auxiliary building. An assessment indicates that the impact on the auxiliary building fragility is insignificant and will not impact the SPRA.

Revised Regulatory Commitment

1. In PG&E Letter DCL-91-178, PG&E stated the following:

"Future additions and modifications to the plant will be designed and constructed in accordance with this existing seismic qualification basis. In addition, certain future plant additions and modifications as specified in enclosed Table 1 will be checked against insights and knowledge gained from the LTSP to verify that the plant 'high-confidence-of low-probability-of-failure' values remain acceptable."

PG&E is replacing the previous commitment with the following:

"PG&E consistent with its LTSP commitments (DCL-91-178) and SSER-34 conclusions, will continue to assess the future plant additions and modifications to verify that the plant seismic risk remains acceptable (i.e. lowered or no significant increases), using the insights from the updated hazard and SPRA."

New Regulatory Commitment

1. The deficiency in the installation of the ducts associated with the 480V switchgear room ventilation system, identified during the walk-downs performed in support of the SPRA update (see Section 4.2.2), will be addressed as summarized in Table 6-1.

Table 6-1 - Actions to be Performed as a Result of the SPRA Update

Action No.	Component ID	Component Description	Action Description	Completion Date
1	N/A	480V switchgear room ventilation ducting (Unit 1)	Modify ducts and duct supports to accommodate differential movements between turbine and auxiliary buildings	End of Unit 1 Refueling Outage No. 21 (3/2019)
2	N/A	480V switchgear room ventilation ducting (Unit 2)	Modify ducts and duct supports to accommodate differential movements between turbine and auxiliary Buildings	End of Unit 2 Refueling Outage No. 21 (12/2019)