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10 CFR 50.54(f)

September 26, 2019
GO2-19-136

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

Subject: COLUMBIA GENERATING STATION DOCKET NO. 50-397 SEISMIC PROBABILISTIC RISK ASSESSMENT RESPONSE TO NRC REQUEST FOR INFORMATION PURSUANT TO 10 CFR 50.54(F) REGARDING RECOMMENDATION 2.1 OF THE NEAR-TERM TASK FORCE (NTTF) REVIEW OF INSIGHTS FROM THE FUKUSHIMA DAI-ICHI ACCIDENT (MF3726, MF3727)

References:

1. Letter from E. J. Leads (NRC) to All Power Reactor Licensees and Holders of Construction Permits in Active or Deferred Status, "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 No. ML12053A340)
2. Letter from W. M. Dean (NRC) to the Power Reactors on the Enclosed List, "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 No. ML15194A015)
3. Letter GO2-18-085 from A. L. Javorik (Energy Northwest) to the NRC, "Request for Extension of Seismic Probabilistic Risk Assessment Submittal Schedule," dated September 6, 2018 (ADAMS No. ML18249A360)
4. Letter from L. Lund (NRC) to B. J. Sawatzke (Energy Northwest), "Columbia Generating Station – Response to Request for Extension of Seismic Probabilistic Risk Assessment Submittal," dated November 20, 2018 (ADAMS No. ML18291A556)

Dear Sir or Madam,

On March 12, 2012, the Nuclear Regulatory Commission (NRC) issued a request for information per 10 CFR 50.54(f) (Reference 1) to all power reactor licensees. Enclosure 1 of Reference 1 requested each addressee to reevaluate the site seismic hazard using updated seismic information and present-day regulatory guidance and methodologies and, if necessary, to perform a risk evaluation. By letter dated October 27, 2015 (Reference 2), the NRC transmitted final seismic information request tables which identified that Energy Northwest was to conduct a seismic probabilistic risk assessment (SPRA) for the Columbia Generating Station (Columbia) by March 31, 2019.

In Reference 3, Energy Northwest requested an extension of the required submittal from March 31, 2019 to September 30, 2019. That request was approved in Reference 4.

The enclosure to this letter provides the SPRA for Columbia, including the requested information in response to Item 8.B of Reference 1 associated with Near-Term Task Force (NTTF) Recommendation 2.1, Seismic Evaluation Criteria.

No new commitments are being made by this letter or enclosure. If you have any questions or require additional information, please contact Ms. D. M. Wolfgramm at (509) 377-4792.

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

Executed on this 26TH day of September, 2019.

Respectfully,

J. Kent Dittmer,
Vice President, Engineering

Enclosure: As stated

cc: NRC RIV Regional Administrator
NRC Project Manager
NRC Senior Resident Inspector / 988C
CD Sonoda – BPA/1399 (email)
WA Horn – Winston & Straw

ENCLOSURE

Columbia Generating Station
Seismic Probabilistic Risk Assessment with Regards to NTTF Recommendation 2.1

(112 pages follow)

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

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Revision 0, Submitted Date: September 13, 2019

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

Revision 0

COLUMBIA GENERATING STATION SEISMIC PROBABILISTIC RISK ASSESSMENT SUMMARY REPORT

Table of Contents

1.0 Purpose and Objective

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

A comparison between the reevaluated seismic hazard and the design basis for Columbia Generating Station (CGS) has been performed, in accordance with the guidance in Electric Power Research Institute (EPRI) 1025287, "Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic" [2], and previously submitted to NRC [3]. That comparison concluded that the ground motion response spectrum (GMRS), which was developed based on the reevaluated seismic hazard, exceeds the design basis seismic response spectrum in the 1 to 10 Hz range, and a seismic risk assessment is required. A seismic probabilistic risk assessment (SPRA) has been developed to perform the seismic risk assessment for CGS in response to the 50.54(f) letter, specifically item (8) in Enclosure 1 of the 50.54(f) letter.

This report describes the SPRA developed for CGS 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 CGS, 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 CGS SPRA.

2.0 Information Provided in This Report

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

1. The list of the significant contributors to seismic core damage frequency (SCDF) for each seismic acceleration bin, including importance measures (e.g., Risk Achievement Worth, Fussell-Vesely (F-V) and Birnbaum),
2. A summary of the methodologies used to estimate the SCDF and seismic large early release frequency (SLERF), including the following:
    i. Methodologies used to quantify the seismic fragilities of SSCs, together with key assumptions,
    ii. SSC fragility values with reference to the method of seismic qualification, the dominant failure mode(s), and the source of information,
    iii. Seismic fragility parameters,
    iv. Important findings from plant walkdowns and any corrective actions taken,
    v. Process used in the seismic plant response analysis and quantification, including the specific adaptations made in the full power internal events (FPIE) probabilistic risk assessment (PRA) model to produce the SPRA model and their motivations,
    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, and
4. Identified plant-specific vulnerabilities and actions that are planned or taken.

Note that 50.54(f) letter Enclosure 1 paragraphs 1 through 6, regarding the seismic hazard evaluation reporting, also apply, but have been satisfied through the previously submitted CGS Seismic Hazard Submittal [3]. Further, 50.54(f) letter Enclosure 1 paragraph 9 requests information on the spent fuel pool. This information is being submitted separately.

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 CGS SPRA has been developed and documented in accordance with the SPID. The main elements of the SPRA performed for CGS in response to the 50.54(f) Seismic letter correspond to those described in Section 6.1.1 of the SPID, that is:

- 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 CGS SPRA and associated documentation has been peer reviewed against the PRA Standard [4] in accordance with the process defined in Nuclear Energy Institute (NEI) 12-13 [5], as documented in the CGS SPRA peer review report [6]. The CGS SPRA, complete SPRA documentation, and details of the peer review are available for NRC review.

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

The content of this report is organized as follows:

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

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

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

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

Section 7 provides references.

Section 8 provides a list of acronyms used.

Appendix A provides an assessment of SPRA Technical Adequacy for Response to NTTF 2.1 Seismic 50.54(f) Letter, including a summary of the CGS 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, 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 fragilities (Am and beta), failure mode information, and method of determining fragilities for the top risk significant SSCs based on standard importance measures such as Fussell-Vesely (F-V). Seismic qualification reference is not provided as it is not relevant to development of SPRA.
2iii	Seismic fragility parameters	Tables 5.4-2 and 5.5-2 provide fragilities (Am and beta) information for the top risk significant SSCs based on standard importance measures such as F-V.
2iv	Important findings from plant walkdowns and any corrective actions taken	Section 4.2
2v	Process used in the seismic plant response analysis and quantification, including specific adaptations made in the internal events PRA model to produce the seismic PRA model and their motivation	Sections 5.1 and 5.2
2vi	Assumptions about containment performance	Sections 5.1 and 5.5
3	Description of the process used to ensure that the SPRA is technically adequate, including the dates and findings of any peer reviews	App. A describes the assessment of SPRA technical adequacy for the 50.54(f) submittal and results of the SPRA peer review

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

50.54(f) Letter Reporting Item	Description	Location in this Report
4	Identified plant-specific vulnerabilities and actions that are planned or taken	Section 6

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

SPID Section 6.8 Item (1) Description	Location in this Report
A report should be submitted to the NRC summarizing the SPRA inputs, methods, and results.	Entirety of the report.
The level of detail needed in the submittal should be sufficient to enable NRC to understand and determine the validity of all input data and calculation models used	Entirety of the report addresses this and identifies key methods of analysis and referenced codes and standards.
The level of detail needed in the submittal should be sufficient to assess the sensitivity of the results to all key aspects of the analysis	Entirety of the report addresses this. Sensitivity evaluations are discussed in Section 5.7.
The level of detail needed in the submittal should be sufficient to make necessary regulatory decisions as a part of NTTF Phase 2 activities.	Entirety of the report.
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.	Entirety of the report addresses this. This report summarizes important information from the SPRA, with detailed information in lower tier documentation
Documentation criteria for a SPRA are identified throughout the American Society of Mechanical Engineers / American Nuclear Society (ASME/ANS) Standard [4]. Utilities are expected to retain that documentation consistent with the Standard.	This is an expectation relative to documentation of the SPRA that the utility retains to support application of the SPRA to risk-informed plant decision-making.

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

3.0 CGS Seismic Hazard and Plant Response

Section 3 provides a high-level summary of the seismic hazard assessment for the CGS site. Detailed information regarding the CGS site hazard was provided to NRC in the seismic hazard information submitted to the NRC [3] in response to the NTTF 2.1 Seismic information request [1]. The response presented the probabilistic seismic hazard analysis (PSHA) for the site and the development of the horizontal GMRS for the control point elevation for safety-related structures. The hazard submittal also describes additional results from the PSHA that are used in the quantification of the CGS SPRA.

The CGS site is located on the Hanford Department of Energy (DOE) Site in south-central Washington State. The site is a soil site consisting of approximately 525 feet of primarily granular sediments overlying the Saddle Mountains Basalt sequence. The granular sediments consist of approximately 45 feet of glacialfluvial gravels overlying a thick sequence of Pliocene sediments, which are designated as the Ringold formation. The Ringold formation consists of alternating layers of gravel-dominated sediments and fine-grained sediments that are variably cemented. Figure 3-1 shows the shear wave velocity profile developed for the suprabasalt sediments at the CGS site. The underlying Saddle Mountains Basalt sequence consists of alternating layers of flood basalts and sedimentary interbeds. Figure 3-2 shows the shear wave profiles developed for the Saddle Mountains Basalt sequence at the CGS site. The Saddle Mountains Basalt sequence in turn overlies the more massive Wanapum and Grande Ronde basalts. Additional details are provided in the NTTF 2.1 hazard submittal [3]. The control point for defining ground motions at the CGS site is the finished grade ground surface at elevation 441 feet.

3.1 Seismic Hazard Analysis

The seismic hazard for the CGS was conducted in two stages. The first stage involved assessment of the hazard at a reference point at the top of the Wanapum basalt at a depth of approximately 1,300 feet below the ground surface at the CGS site. This hazard assessment was performed as part of the sitewide PSHA for the Hanford DOE site conducted by Pacific Northwest National Laboratory (PNNL) [7]. The second stage involved characterization of the dynamic response of the soils and rocks above the top of the Wanapum basalts to earthquake ground motions [8], [9] and then convolving that response with the reference rock hazard to provide an assessment of the seismic hazard at the CGS control point, defined as the plant finished grade [10].

3.1.1 Seismic Hazard Analysis Methodology

The PNNL seismic hazard assessment for reference rock at the CGS site was developed as part of the Hanford Sitewide PSHA. The PNNL study was conducted as a Senior Seismic Hazard Analysis Committee (SSHAC) Level 3 study following the guidance given in Budnitz [11] and in NRC Guidelines for SSHAC Level 3 and 4 Hazard studies [12]. The study identified and characterized the seismic sources in the region and developed ground motion models specific to characterization of motions at the top of the Wanapum basalt at the CGS site. The study produced mean and fractile hazard curves for peak ground acceleration (PGA) and 5-percent damped spectral acceleration at frequencies of 50, 33.3, 25, 20, 13.3, 10, 6.67, 5, 3.33, 2.5, 2, 1.33, 1, 0.67, 0.5, 0.33, 0.2, 0.13, and 0.1 Hz for the reference rock location. In addition, characterization of the response spectra for earthquakes contributing to the hazard at the site was provided in the form of conditional mean spectra (CMS) for each of the above spectral frequencies and for 27 annual exceedance frequencies (AEFs) ranging from 10^{-2} to 10^{-7}. Details of this assessment are provided in [7]. Figure 3-3 shows the uniform hazard response spectra (UHRS) developed by PNNL [7] for the top of Wanapum basalt horizon at the CGS site.

For the CGS SPRA, the plant-specific hazard at the control point was assessed by developing a probabilistic characterization of the response of the materials above the reference rock horizon used in [7] and convolving this response with the reference rock hazard following Approach 3 as described in McGuire [13] and Bazzurro [14]. The characterization of the dynamic properties of the CGS site is described in Calculation 25709-000-K0C-0000-00001 [8]. The properties are based on measurements made at the CGS and adjacent sites as part of development of the license application [15]. Figure 3-1 shows the shear wave velocity profile developed for the sediments overlying the Saddle Mountains Basalts by [8]. The characterization of shear modulus reduction (G/Gmax) and damping in these sediments followed the recommendations given in Appendix B of [2].

The characterization of the dynamic properties of the Saddle Mountains Basalts and sedimentary interbeds was provided by [7]. Two alternatives were developed for the shear wave velocities in the basalts (Figure 3-2). Associated damping ratios for the basalts were assessed based on estimates of shallow crustal damping (kappa) and the basalts were assumed to behave linearly under dynamic loading. Shear modulus reduction and damping relationships for the sedimentary interbeds were developed based on published relationships.

Bechtel Power [9] used the above characterization of the dynamic properties of the CGS site in combination with the characterization of response spectra for contributing earthquakes developed by [7] to develop a probabilistic representation of the amplification of the CGS site. Site amplification functions were developed for each of the 20 ground motion frequencies for which [7] provided reference rock hazard results. Epistemic uncertainty in the amplification was characterized by developing amplification functions for the four alternative combinations of the two sets of G/Gmax and damping relationships for the suprabasalt sediments and the two alternative basalt velocity profiles.

Using Approach 3 of [13] and [14], Bechtel Power [10] convolved the probabilistic site amplification functions with the reference rock hazard curves developed by [7] to produce hazard curves for horizontal motion at the ground surface control point at the CGS site. The resulting hazard curves were interpolated to obtain response spectral accelerations corresponding to AEFs of 10^{-4} and 10^{-5}. These values were used to form the 10^{-4} and 10^{-5} horizontal control point UHRS shown on Figure 3-4. The procedure given in Regulatory Guide 1.208 [16] was then used to develop the horizontal GMRS, which is also shown on Figure 3-4. Table 3-1 lists the control point 10^{-4} and 10^{-5} horizontal UHRS and horizontal GMRS. Vertical response spectra were developed as described in Subsection 3.1.4.

3.1.2 Seismic Hazard Analysis Technical Adequacy

The reference rock hazard for the CGS site was developed as part of a SSHAC Level 3 study. The study was conducted under the continuous observation of a Participatory Peer Review Panel (PPRP) that reviewed both the technical adequacy of the assessments and the adherence to the SSHAC guidelines provided in [11] and [12]. Documentation of the acceptance of the PPRP of the final report is provided in Appendix B of [7].

The CGS SPRA hazard methodology and analysis associated with the horizontal GMRS were submitted to the NRC [3], and found to be technically acceptable by NRC for application to the CGS SPRA [17].

The CGS seismic hazard analysis was also subjected to an independent peer review against all seismic hazard analysis (SHA) requirements for Capability Category II in the PRA Standard [4]. The peer review assessment and subsequent disposition of peer review findings are described in Appendix A.

3.1.3 Seismic Hazard Analysis Results and Insights

This section provides the final seismic hazard results used in the CGS SPRA. The SPRA quantification for the CGS site was performed using the seismic hazard curve for 2.5 Hz spectral acceleration at the soil surface. Table 3-2 lists the mean and fractile hazard curves for 2.5 Hz and they are shown on Figure 3-5.

The evaluation of plant fragilities was based on the 10^{-5} UHRS for the ground surface control point. These spectra are shown on Figure 3-6 and are listed in Table 3-3.

PNNL [7] developed an extensive logic tree model to quantify the epistemic uncertainty in the assessment of the seismic hazard at the CGS site. The major contributors to the epistemic uncertainty in high frequency motions (e.g. PGA and 10 Hz pseudo spectral acceleration (PSA)) are the components of the ground motion characterization model for crustal earthquakes, principally the adjustments to the Wanapum reference site conditions (Vs-kappa adjustments), the scaling of the median models, and the value of aleatory variability. The largest contributions to epistemic uncertainty from seismic source characterization are uncertainty in the b-value of the truncated exponential model fit to the seismicity and uncertainty in the style of faulting for adjacent fault-specific sources. For low frequency motions (e.g. 1 Hz PSA), the contribution from uncertainty in the Vs-kappa adjustments is replaced by uncertainty in the anelastic attenuation rate for ground motions from large, distant earthquakes occurring on the Cascadia subduction zone and uncertainty in maximum magnitude becomes a significant contributor to the overall uncertainty in the hazard.

Sensitivity analyses presented in the site response analysis [9] indicate that the epistemic uncertainty in the Saddle Mountains Basalt sequence shear wave velocities and in the G/Gmax and damping relationships for the suprabasalt sediments each produce a variation of approximately 5 percent in the spectral acceleration at 2.5 Hz. Sensitivity analyses presented in Calculation 2579-0000-K0C-0000-00004 [10] indicate that the assumption of the minimum level of site amplification has a significant impact on the GMRS and UHRS at frequencies above 10 Hz but has minimal impact at frequencies below 6 Hz.

3.1.4 Horizontal and Vertical 10^{-5} UHRS

As discussed in Section 4, fragility analyses for the CGS were developed using the horizontal 10^{-5} UHRS. A mean vertical 10^{-5} response spectrum was developed by Sage Engineers [18] using the vertical to horizontal response spectral ratio methodology described in Gulerce [19]. The V/H ratios were assessed conditionally on the ground motion levels of the horizontal 10^{-5} UHRS relative to median motions for contributing earthquake scenarios. Figure 3-6 shows the vertical and horizontal 10^{-5} spectra. The horizontal 10^{-5} UHRS, the V/H ratios and the vertical 10^{-5} spectrum are listed in Table 3-3. V/H ratios were also developed for the 10^{-4} mean hazard level by [18]. A vertical GMRS was not developed because the GMRS was not used for fragility evaluations. The control point for seismic input was taken at the soil surface for all structures.

Table 3-1 CGS Control Point Horizontal UHRS and GMRS

Period (s)	Frequency (Hz)	5%-damped Spectral Acceleration for: (g)		
		10^{-4} UHRS	10^{-5} UHRS	GMRS
0.010	100.000	0.2484	0.4288	0.2484
0.020	50.000	0.2951	0.5057	0.2951
0.030	33.333	0.3471	0.6242	0.3471
0.040	25.000	0.3916	0.7238	0.3916
0.050	20.000	0.3595	0.6537	0.3595
0.075	13.333	0.4341	0.8088	0.4341
0.100	10.000	0.4978	0.9638	0.5067
0.150	6.667	0.7427	1.4240	0.7501
0.200	5.000	1.2160	2.4340	1.2711
0.300	3.333	1.3236	2.8030	1.4474
0.400	2.500	0.7958	1.7767	0.9078
0.500	2.000	0.7360	1.7620	0.8878
0.750	1.333	0.5313	1.3565	0.6748
1.000	1.000	0.3781	0.9234	0.4634
1.500	0.667	0.3089	0.7104	0.3609
2.000	0.500	0.1851	0.4552	0.2281
3.000	0.333	0.0837	0.1917	0.0974
5.000	0.200	0.0435	0.0912	0.0472
7.500	0.133	0.0262	0.0540	0.0280
10.000	0.100	0.0196	0.0397	0.0207

Table 3-2 Horizontal CGS Control Point Hazard Curves for 2.5 Hz

5%-damped 2.5 Hz Spectral Acceleration (g)	Annual Exceedance Frequency					
	Mean	5th Fractile	16th Fractile	50th Fractile	84th Fractile	95th Fractile
0.000100	9.485E-02	6.185E-02	7.100E-02	8.736E-02	1.125E-01	1.561E-01
0.000113	9.480E-02	6.182E-02	7.097E-02	8.732E-02	1.125E-01	1.559E-01
0.000128	9.471E-02	6.176E-02	7.091E-02	8.724E-02	1.124E-01	1.555E-01
0.000145	9.457E-02	6.168E-02	7.082E-02	8.714E-02	1.123E-01	1.551E-01
0.000164	9.438E-02	6.158E-02	7.070E-02	8.698E-02	1.121E-01	1.544E-01
0.000185	9.414E-02	6.144E-02	7.054E-02	8.679E-02	1.118E-01	1.535E-01
0.000210	9.385E-02	6.128E-02	7.035E-02	8.656E-02	1.115E-01	1.526E-01
0.000237	9.352E-02	6.110E-02	7.014E-02	8.631E-02	1.112E-01	1.514E-01
0.000268	9.318E-02	6.090E-02	6.992E-02	8.603E-02	1.108E-01	1.502E-01
0.000303	9.282E-02	6.070E-02	6.968E-02	8.574E-02	1.105E-01	1.489E-01
0.000343	9.244E-02	6.048E-02	6.944E-02	8.544E-02	1.101E-01	1.477E-01
0.000388	9.206E-02	6.027E-02	6.920E-02	8.513E-02	1.097E-01	1.464E-01
0.000439	9.166E-02	6.004E-02	6.894E-02	8.482E-02	1.093E-01	1.450E-01
0.000497	9.123E-02	5.980E-02	6.866E-02	8.448E-02	1.089E-01	1.436E-01
0.000562	9.076E-02	5.953E-02	6.836E-02	8.410E-02	1.083E-01	1.421E-01
0.000636	9.019E-02	5.921E-02	6.799E-02	8.365E-02	1.077E-01	1.405E-01
0.000719	8.949E-02	5.883E-02	6.755E-02	8.310E-02	1.069E-01	1.387E-01
0.000813	8.862E-02	5.833E-02	6.698E-02	8.242E-02	1.059E-01	1.365E-01
0.000920	8.752E-02	5.771E-02	6.628E-02	8.156E-02	1.046E-01	1.340E-01
0.001041	8.618E-02	5.693E-02	6.540E-02	8.049E-02	1.030E-01	1.311E-01
0.001177	8.456E-02	5.595E-02	6.429E-02	7.917E-02	1.011E-01	1.278E-01
0.001332	8.262E-02	5.473E-02	6.294E-02	7.757E-02	9.880E-02	1.241E-01
0.001507	8.035E-02	5.324E-02	6.131E-02	7.565E-02	9.618E-02	1.200E-01
0.001704	7.774E-02	5.146E-02	5.936E-02	7.338E-02	9.320E-02	1.156E-01
0.001928	7.480E-02	4.940E-02	5.712E-02	7.079E-02	8.988E-02	1.109E-01
0.002181	7.160E-02	4.710E-02	5.463E-02	6.792E-02	8.630E-02	1.059E-01
0.002467	6.822E-02	4.464E-02	5.197E-02	6.485E-02	8.254E-02	1.008E-01
0.002791	6.475E-02	4.210E-02	4.922E-02	6.170E-02	7.869E-02	9.576E-02
0.003157	6.130E-02	3.956E-02	4.647E-02	5.852E-02	7.486E-02	9.074E-02
0.003571	5.793E-02	3.709E-02	4.379E-02	5.541E-02	7.110E-02	8.587E-02
0.004040	5.468E-02	3.472E-02	4.120E-02	5.241E-02	6.747E-02	8.119E-02
0.004570	5.156E-02	3.247E-02	3.872E-02	4.951E-02	6.395E-02	7.671E-02
0.005170	4.857E-02	3.032E-02	3.634E-02	4.672E-02	6.057E-02	7.241E-02
0.005848	4.568E-02	2.827E-02	3.405E-02	4.401E-02	5.725E-02	6.826E-02
0.006615	4.284E-02	2.629E-02	3.181E-02	4.134E-02	5.396E-02	6.421E-02
0.007483	4.002E-02	2.435E-02	2.959E-02	3.867E-02	5.065E-02	6.020E-02
0.008465	3.720E-02	2.244E-02	2.736E-02	3.597E-02	4.728E-02	5.621E-02

Table 3-2 Horizontal CGS Control Point Hazard Curves for 2.5 Hz

5%-damped 2.5 Hz Spectral Acceleration (g)	Annual Exceedance Frequency					
	Mean	5th Fractile	16th Fractile	50th Fractile	84th Fractile	95th Fractile
0.009576	3.437E-02	2.055E-02	2.513E-02	3.324E-02	4.386E-02	5.221E-02
0.010833	3.155E-02	1.869E-02	2.291E-02	3.050E-02	4.040E-02	4.824E-02
0.012254	2.877E-02	1.688E-02	2.073E-02	2.780E-02	3.696E-02	4.432E-02
0.013862	2.608E-02	1.514E-02	1.863E-02	2.516E-02	3.360E-02	4.049E-02
0.015681	2.350E-02	1.349E-02	1.664E-02	2.264E-02	3.038E-02	3.681E-02
0.017739	2.107E-02	1.195E-02	1.478E-02	2.026E-02	2.732E-02	3.330E-02
0.020066	1.881E-02	1.052E-02	1.305E-02	1.803E-02	2.447E-02	2.999E-02
0.022699	1.672E-02	9.200E-03	1.147E-02	1.598E-02	2.182E-02	2.690E-02
0.025678	1.479E-02	8.006E-03	1.004E-02	1.410E-02	1.939E-02	2.403E-02
0.029047	1.304E-02	6.930E-03	8.746E-03	1.240E-02	1.717E-02	2.139E-02
0.032859	1.146E-02	5.968E-03	7.587E-03	1.085E-02	1.516E-02	1.898E-02
0.037170	1.003E-02	5.116E-03	6.554E-03	9.470E-03	1.335E-02	1.680E-02
0.042048	8.750E-03	4.366E-03	5.638E-03	8.238E-03	1.173E-02	1.483E-02
0.047565	7.612E-03	3.711E-03	4.832E-03	7.144E-03	1.028E-02	1.306E-02
0.053806	6.604E-03	3.143E-03	4.127E-03	6.180E-03	8.987E-03	1.148E-02
0.060867	5.715E-03	2.655E-03	3.515E-03	5.332E-03	7.840E-03	1.007E-02
0.068853	4.935E-03	2.237E-03	2.986E-03	4.590E-03	6.826E-03	8.824E-03
0.077888	4.251E-03	1.880E-03	2.530E-03	3.940E-03	5.927E-03	7.709E-03
0.088108	3.653E-03	1.577E-03	2.139E-03	3.373E-03	5.132E-03	6.717E-03
0.099670	3.130E-03	1.320E-03	1.803E-03	2.878E-03	4.428E-03	5.833E-03
0.112748	2.674E-03	1.103E-03	1.516E-03	2.446E-03	3.807E-03	5.046E-03
0.127542	2.276E-03	9.180E-04	1.270E-03	2.070E-03	3.259E-03	4.348E-03
0.144278	1.930E-03	7.615E-04	1.060E-03	1.744E-03	2.778E-03	3.730E-03
0.163210	1.629E-03	6.291E-04	8.809E-04	1.463E-03	2.358E-03	3.187E-03
0.184625	1.370E-03	5.171E-04	7.289E-04	1.222E-03	1.992E-03	2.709E-03
0.208851	1.146E-03	4.226E-04	6.002E-04	1.015E-03	1.676E-03	2.293E-03
0.236256	9.536E-04	3.431E-04	4.914E-04	8.390E-04	1.403E-03	1.931E-03
0.267257	7.893E-04	2.765E-04	3.998E-04	6.895E-04	1.168E-03	1.617E-03
0.302325	6.491E-04	2.209E-04	3.229E-04	5.630E-04	9.670E-04	1.347E-03
0.341995	5.300E-04	1.749E-04	2.586E-04	4.564E-04	7.955E-04	1.114E-03
0.386871	4.293E-04	1.370E-04	2.052E-04	3.669E-04	6.492E-04	9.156E-04
0.437634	3.447E-04	1.060E-04	1.610E-04	2.922E-04	5.252E-04	7.463E-04
0.495059	2.739E-04	8.097E-05	1.247E-04	2.302E-04	4.207E-04	6.030E-04
0.560020	2.151E-04	6.091E-05	9.528E-05	1.790E-04	3.331E-04	4.821E-04
0.633503	1.668E-04	4.505E-05	7.161E-05	1.373E-04	2.603E-04	3.810E-04
0.716629	1.274E-04	3.271E-05	5.287E-05	1.036E-04	2.005E-04	2.971E-04
0.810663	9.582E-05	2.327E-05	3.828E-05	7.689E-05	1.519E-04	2.283E-04
0.917036	7.080E-05	1.620E-05	2.714E-05	5.598E-05	1.131E-04	1.726E-04

Table 3-2 Horizontal CGS Control Point Hazard Curves for 2.5 Hz

5%-damped 2.5 Hz Spectral Acceleration (g)	Annual Exceedance Frequency					
	Mean	5th Fractile	16th Fractile	50th Fractile	84th Fractile	95th Fractile
1.037366	5.135E-05	1.102E-05	1.883E-05	3.995E-05	8.269E-05	1.283E-04
1.173485	3.653E-05	7.321E-06	1.277E-05	2.791E-05	5.927E-05	9.356E-05
1.327466	2.547E-05	4.744E-06	8.455E-06	1.909E-05	4.163E-05	6.694E-05
1.501652	1.741E-05	2.998E-06	5.468E-06	1.277E-05	2.864E-05	4.697E-05
1.698694	1.166E-05	1.847E-06	3.453E-06	8.350E-06	1.931E-05	3.232E-05
1.921589	7.655E-06	1.109E-06	2.130E-06	5.344E-06	1.275E-05	2.180E-05
2.173734	4.932E-06	6.506E-07	1.284E-06	3.349E-06	8.258E-06	1.444E-05
2.458964	3.121E-06	3.725E-07	7.568E-07	2.056E-06	5.249E-06	9.386E-06
2.781622	1.942E-06	2.086E-07	4.367E-07	1.239E-06	3.277E-06	6.002E-06
3.146617	1.190E-06	1.143E-07	2.471E-07	7.328E-07	2.013E-06	3.777E-06
3.559503	7.190E-07	6.147E-08	1.373E-07	4.264E-07	1.217E-06	2.342E-06
4.026569	4.290E-07	3.245E-08	7.496E-08	2.443E-07	7.264E-07	1.434E-06
4.554923	2.531E-07	1.684E-08	4.029E-08	1.379E-07	4.279E-07	8.671E-07
5.152606	1.479E-07	8.610E-09	2.134E-08	7.689E-08	2.492E-07	5.188E-07
5.828714	8.566E-08	4.338E-09	1.115E-08	4.236E-08	1.437E-07	3.075E-07
6.593533	4.925E-08	2.157E-09	5.759E-09	2.307E-08	8.203E-08	1.807E-07
7.458715	2.814E-08	1.059E-09	2.939E-09	1.244E-08	4.646E-08	1.054E-07
8.437425	1.598E-08	5.138E-10	1.484E-09	6.639E-09	2.611E-08	6.105E-08
9.544557	9.033E-09	2.464E-10	7.408E-10	3.510E-09	1.456E-08	3.515E-08
10.796952	5.082E-09	1.168E-10	3.660E-10	1.839E-09	8.068E-09	2.012E-08
12.213694	2.846E-09	5.469E-11	1.789E-10	9.545E-10	4.439E-09	1.145E-08
13.816336	1.587E-09	2.530E-11	8.643E-11	4.911E-10	2.426E-09	6.477E-09
15.629272	8.816E-10	1.155E-11	4.128E-11	2.505E-10	1.318E-09	3.643E-09
17.680094	4.873E-10	5.206E-12	1.949E-11	1.266E-10	7.109E-10	2.036E-09
20.000000	2.678E-10	2.314E-12	9.102E-12	6.349E-11	3.807E-10	1.130E-09

Table 3-3 Horizontal and Vertical Control Point 10^{-5} UHRS for the CGS Site

Period (s)	Frequency (Hz)	Horizontal Spectral Acceleration (g)	V/H	Vertical Spectral Acceleration (g)
0.010	100.000	0.429	1.053	0.45
0.020	50.000	0.506	1.088	0.56
0.030	33.333	0.624	1.293	0.80
0.040	25.000	0.724	1.606	1.16
0.050	20.000	0.654	1.924	1.25
0.075	13.333	0.809	2.161	1.75
0.100	10.000	0.964	1.635	1.57
0.150	6.667	1.424	1.066	1.51
0.200	5.000	2.434	0.751	1.83
0.300	3.333	2.803	0.481	1.35
0.400	2.500	1.777	0.388	0.69
0.500	2.000	1.762	0.342	0.60
0.750	1.333	1.357	0.318	0.43
1.000	1.000	0.923	0.339	0.31
1.500	0.667	0.710	0.382	0.27
2.000	0.500	0.455	0.430	0.20
3.000	0.333	0.192	0.720	0.14
5.000	0.200	0.091	1.061	0.10
7.500	0.133	0.054	1.389	0.07
10.000	0.100	0.040	1.419	0.06

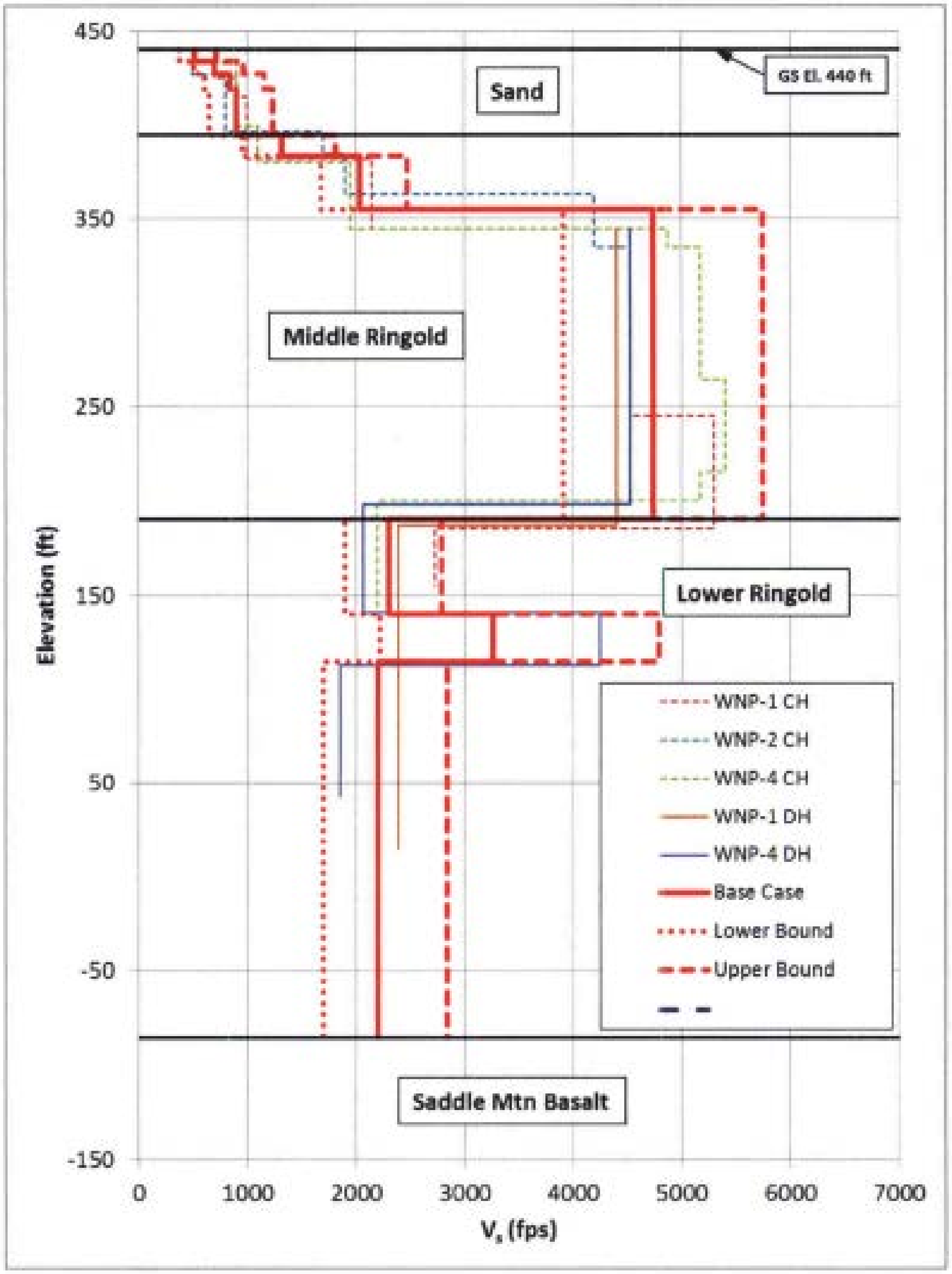

Figure 3-1 Base Case Shear Wave Velocity Profile for the Suprabasalt Sediments at the CGS Site

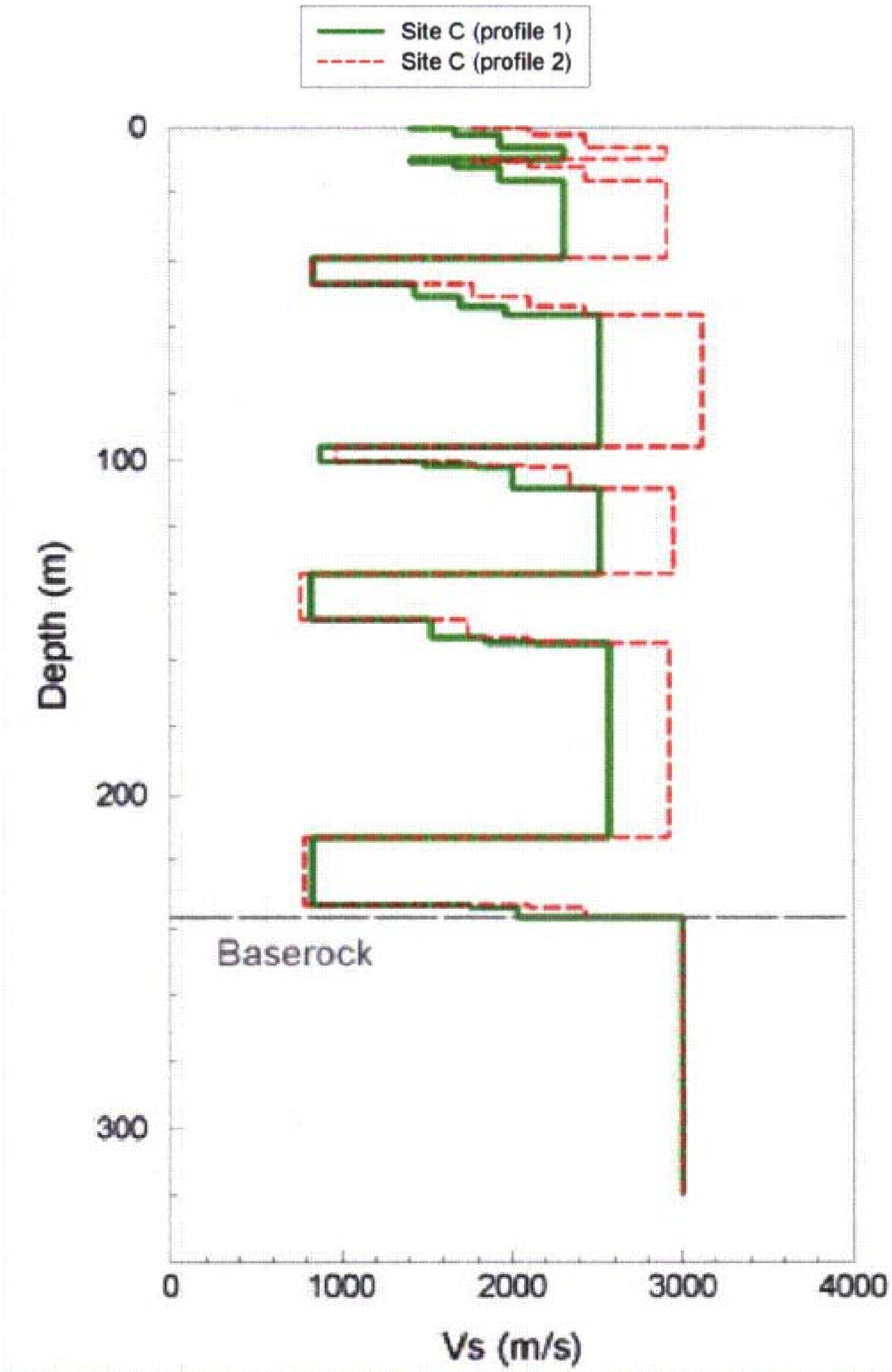

Figure 3-2 Shear Wave Velocity Profiles for the Saddle Mountains Basalt Sequence at the CGS Site

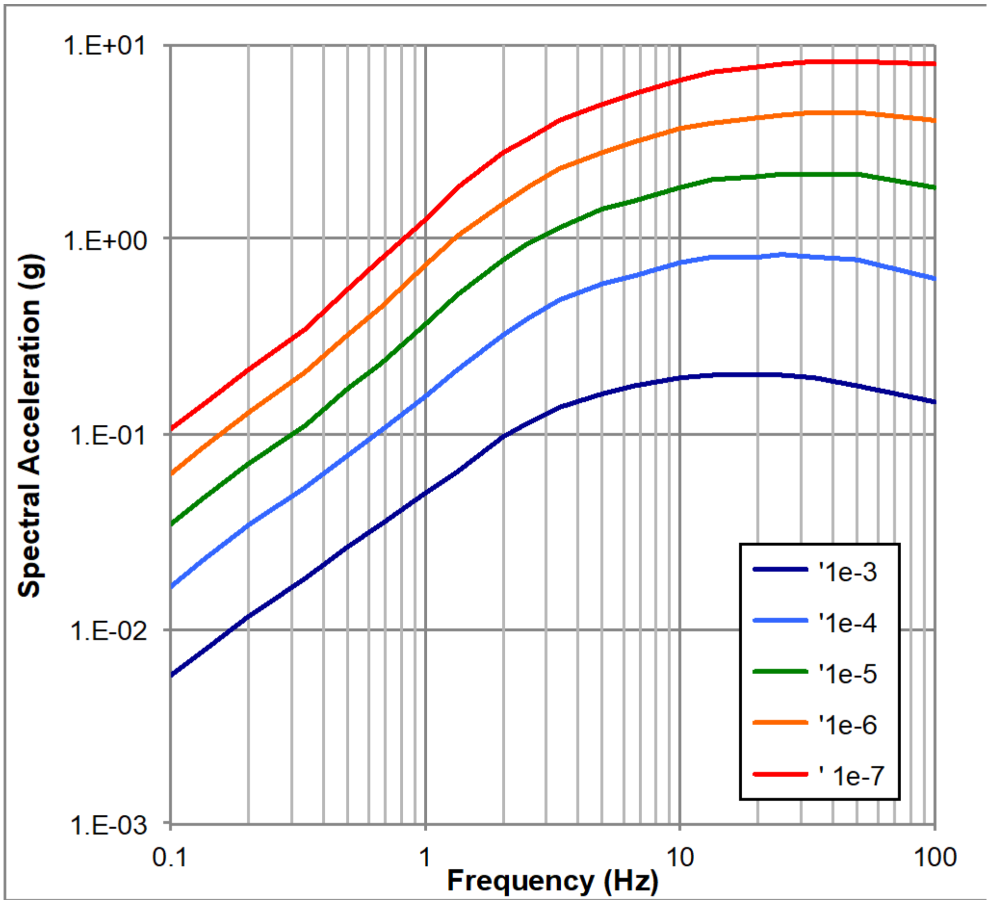

Figure 3-3 Horizontal UHRS for Top of Wanapum Basalt Reference Point for the CGS Site

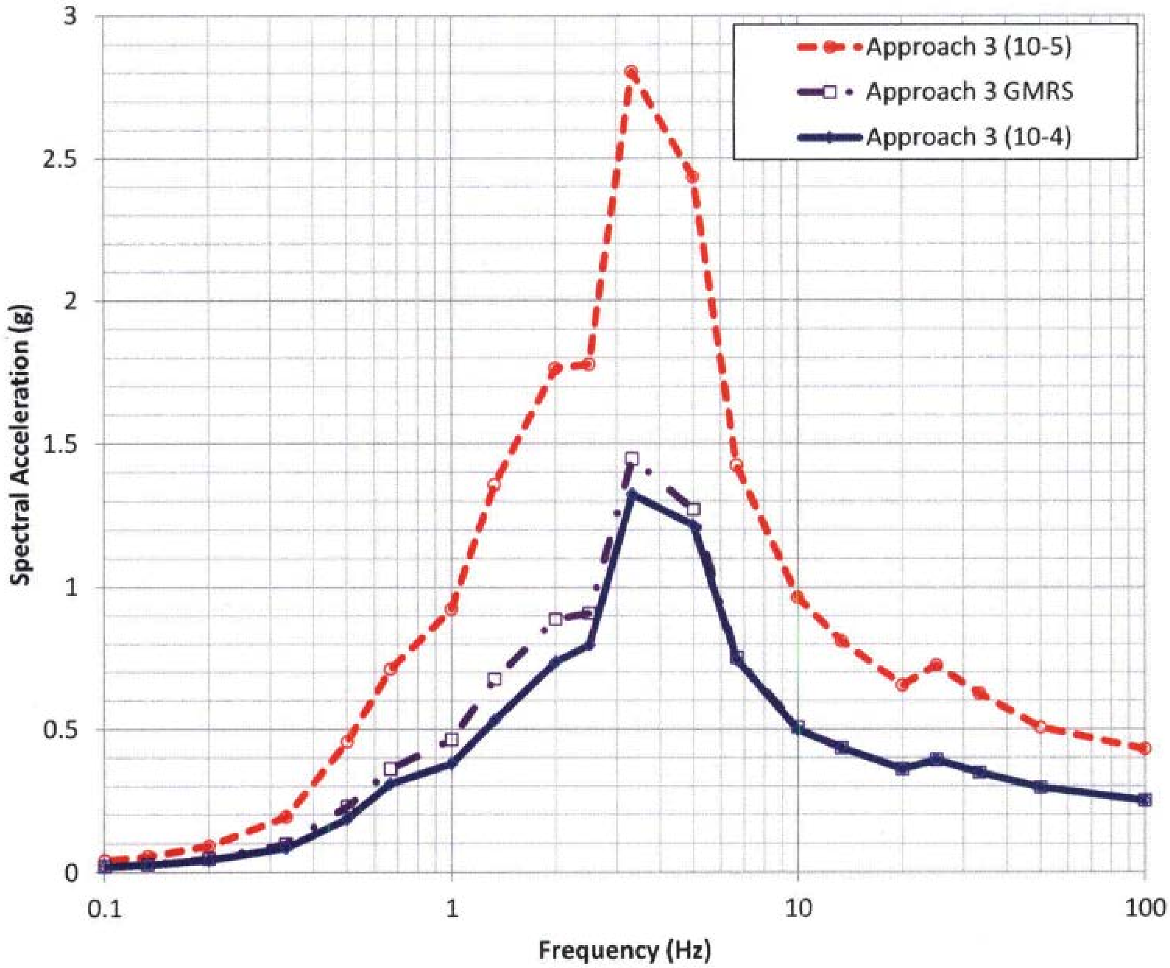

Figure 3-4 Horizontal Control Point Horizontal 10^{-4} and 10^{-5} UHRS and GMRS for the CGS Site (5% damping)

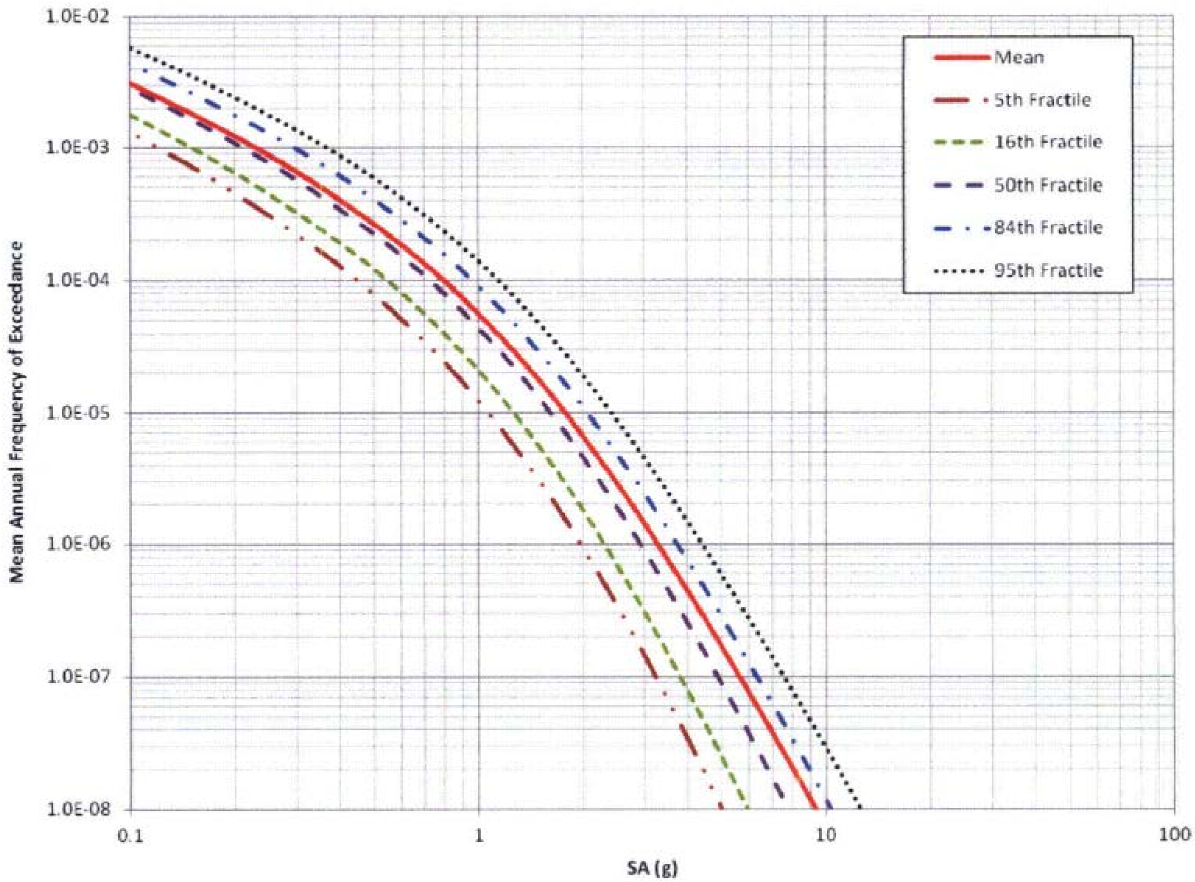

Figure 3-5 Mean and Fractile Control Point Horizontal 2.5 Hz Spectral Acceleration Hazard Curves for the CGS Site (5% damping)

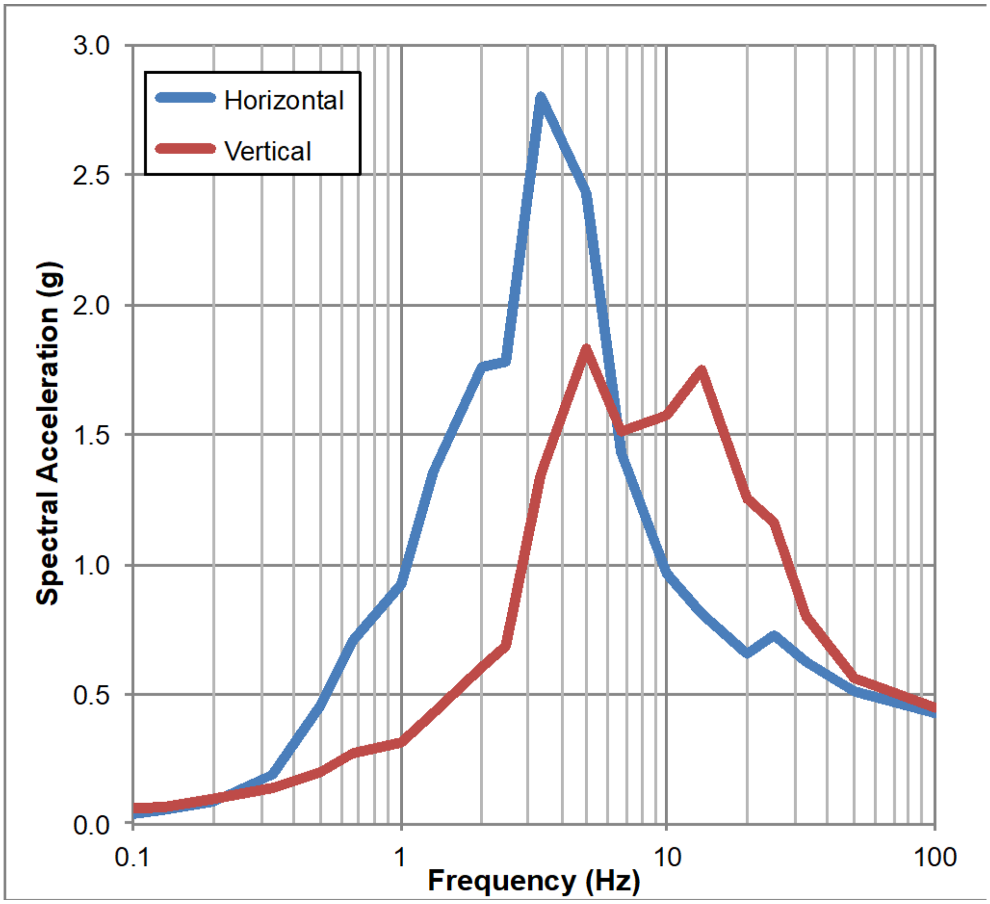

Figure 3-6 Horizontal and Vertical Control Point 10^{-5} UHRS for the CGS Site (5% damping)

4.0 Determination of Seismic Fragilities for the SPRA

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

4.1 Seismic Equipment List

For the CGS 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 CGS SPRA SEL was developed using guidance provided by the SPRA Implementation Guide [20].

4.1.1 SEL Development

The CGS FPIE PRA and individual plant examination for external events (IPEEE) evaluation, including an SPRA that was developed for the IPEEE, along with the associated documents and models, provided inputs to the SEL development. This information was used in conjunction with other plant documents such as the CGS master equipment list (MEL), plant systems descriptions, piping and instrumentation drawings (P&IDs), and electrical schematic drawings to develop the SEL [21] for use in the SPRA.

SEL SSCs were selected based on consideration of the initiating events and consequential events that could occur as a result of seismic events, the safety functions that must be fulfilled to respond to these initiating events, and the frontline and support systems that are credited in the SPRA to meet each function for core damage and large early release accident sequences.

The safety functions that would be required to respond to initiating events identified above were determined based on EPRI NP-6041-SL [22] and NUREG 1407 [23]. These safety functions are:

- Reactivity control,
- Reactor coolant system pressure control,
- Reactor coolant system inventory control,
- Decay heat removal, and
- Containment isolation and integrity.

A review of all plant systems was performed for applicability to the SEL. The frontline systems that are credited in the SPRA to meet each function for core damage and large early release accident sequences are listed in Table 4.1-1. Support systems modeled in the SPRA include the following:

- Emergency ac power (including ac distribution, fixed diesel generators, and portable diverse and flexible mitigation strategies (FLEX) diesel generators),
- 125 V and 250 V dc power,
- Standby service water (SW),
- Heating, ventilation, and air conditioning (HVAC) (reactor building recirculation air, radwaste building mixed air, diesel generator building mixed air, SW pump house recirculation air), and
- Pneumatic supplies (containment instrument air, hardened containment vent (HCV) air supply).

Table 4.1-1 CGS SPRA Frontline Systems per Safety Function

Critical Safety Function	Systems
Reactivity Control	Control rod drive (CRD) Recirculation pump trip (RPT) Standby liquid control (SLC) Reactor internals
RPV Pressure Control	Automatic depressurization system (ADS) and non-ADS main steam safety relief valves (SRVs) RPT
RPV Coolant Inventory Control (High Pressure)	High pressure core spray (HPCS) Reactor core isolation cooling (RCIC)
RPV Coolant Inventory Control (Low Pressure)	Low pressure cooling injection (LPCI) mode of residual heat removal (RHR) Low pressure core spray (LPCS) SW through RHR FLEX diesel fire pumps
RPV Depressurization	ADS and non-ADS SRVs
Containment Pressure and Temperature Control	Suppression pool cooling mode of RHR Containment spray HCV
Vapor Suppression	Wetwell-to-drywell vacuum breakers Drywell floor seal Drywell spray mode of RHR ADS and non-ADS SRVs
Containment Isolation	Nuclear steam supply shutoff system Primary containment structure

The seismic equipment list development steps in Figure 5-1 of [20] were taken into consideration while developing the SEL. For each applicable system in the SEL, the following steps were followed in the order presented to identify the appropriate SSCs to be included in the SEL from the internal events PRA, and IPEEE safe shutdown equipment list (SSEL):

1. The CGS MEL [24] was used as a starting point. The SSCs included in the Internal Events PRA were identified and marked in the MEL. This included documenting the applicable basic event(s) associated with each of the SSCs included in the internal events PRA.
2. The IPEEE SSEL was reviewed for applicability of SSEL SSCs to the SPRA SEL.

After reviewing the MEL and crosschecking against the IPEEE SSEL, the following plant information sources were used to identify any additional SSCs that should be added to the SEL:

- P&IDs,
- Electrical one-line diagrams,
- System notebooks for the internal events PRA, and
- CGS systems descriptions.

SSCs were added to include equipment applicable to the SPRA, but not modeled explicitly by the FPIE PRA. These SSCs included:

- Structures,
- Cable trays,
- Sources of seismic-induced fire,
- Sources of seismic-induced flood,
- HVAC ducts,
- Piping,
- Instrumentation utilized to perform operator actions credited by the SPRA, and,
- Seismic II/I interaction concerns, that is, non-safety related equipment that could interact with and cause damage to safety-related SSCs.

The SEL includes the following structures:

- Diesel generator building (DGB),
- Drywell / primary containment,
- SW pump houses,
- Radwaste / control building (RWCB),
- Service building,
- Turbine building (TB),
- Diesel generator (DG) fresh air intakes,
- SW spray pond,
- Reactor building (RB),
- Reactor pressure vessel (RPV) internals,
- RPV supports,
- SW cable tunnels,
- FLEX buildings, and
- Condensate storage tanks (CSTs).

SSCs were added to the SEL that can cause or contribute to initiating events. The list of plant-specific initiating events is provided in Subsection 5.1.2

The SEL includes NSSS components and other SSCs required for containment integrity.

The SEL excludes systems requiring offsite power given the assumption that offsite power is lost due to a seismic event.

The SEL excludes high seismic capacity SSCs:

- Check valves,
- Manual valves and dampers which are not required to change state,

- Flow elements,
- Flow glasses,
- Push buttons,
- Pressure elements,
- Position transmitters,
- Pressure transmitters (not for chatter, but for functionality after),
- Pressure transducers (not for chatter, but for functionality after),
- Temperature elements,
- Differential temperature transmitters (not for chatter, but for functionality after),
- Temperature switches (not for chatter, but for functionality after),
- Temperature transmitters (not for chatter, but for functionality after),
- Small relief valves,
- Small passive in-line filters that are supported only by the piping or ducting,
- Hand switches, and
- Heat tracing.

The final SEL was documented for the SPRA [21]. The resulting SEL consists of about 1200 SSC entries. The SEL was consolidated from the approximately 1300 SSCs walked down to account for components identified to be rule-of-the-box as well as items identified to be not applicable to the SPRA upon inspection.

4.1.2 Contact Chatter Evaluation

During a seismic event, vibratory ground motion can cause electrical contact devices (ECDs), such as relays, to chatter. The chattering of ECDs potentially can result in spurious signals to equipment. Most contact 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 [25] was performed for the CGS SPRA, in accordance with SPID Section 6.4.2 [2], and the ASME/ANS PRA Standard [4]. The evaluation screened most contact chatter scenarios from further evaluation based on no impact to component function. The 211 ECDs that were not screened by the chatter evaluation are listed in Table 4.1-2, along with their functions and dispositions in the SPRA with respect to modeling of seismic fragilities and operator actions.

Table 4.1-2 Disposition of Unscreened Electrical Contact Devices

ECD	Function	Disposition
E-RLY-52T/DG1/7 E-CB-DG1/7 E-RLY-52T/DG2/8 E-CB-DG2/8 DG-RLY-86/DG1 DG-RLY-86/DG2 DG-RLY-DG1/K21 RHR-RLY-K98A DG-RLY-DG2/K21 RHR-RLY-K98B	DG 1 and DG 2 output breakers trip	Modeled in fault tree for seismic failure with separation of variables (SOV) fragilities and operator recovery.
RHR-RLY-K18A RHR-RLY-K70A RHR-RLY-K18B RHR-RLY-K70B SW-RLY-62/P1A SW-RLY-62/P1B	DG 1 and DG 2 output breakers trip and lock out	Modeled in fault tree for seismic failure with SOV fragilities and operator recovery.

Table 4.1-2 Disposition of Unscreened Electrical Contact Devices

ECD	Function	Disposition
DCW-TS-11A1 DCW-TS-11A2 DG-RLY-2759/DG1 DG-RLY-40/DG1 DG-RLY-62/DG1 DG-RLY-67/DG1 DG-RLY-94/DG1 DG-RLY-DG1/K26 DG-RLY-DG1/K4 DG-RLY-DG1/K5 DG-RLY-DG1/K6 DG-RLY-DG1/K60 DG-RLY-DG1/K60A DG-RLY-DG1/K9 DCW-TS-11B1 DCW-TS-11B2 DG-RLY-2759/DG2 DG-RLY-40/DG2 DG-RLY-62/DG2 DG-RLY-67/DG2 DG-RLY-94/DG2 DG-RLY-DG2/K26 DG-RLY-DG2/K4 DG-RLY-DG2/K5 DG-RLY-DG2/K6 DG-RLY-DG2/K60 DG-RLY-DG2/K60A DG-RLY-DG2/K9 E-RLY-86/B/8	DG 1 and DG 2 failure to start or run	Modeled in fault tree for seismic failure with SOV fragilities and operator recovery.

Table 4.1-2 Disposition of Unscreened Electrical Contact Devices

ECD	Function	Disposition
HPCS-RLY-CPR/DG3 E-RLY-27/NX E-RLY-27/NY E-RLY-2762/4/1 E-RLY-2762/4/2 E-RLY-2762/4/3 E-RLY-62/4/S3 E-RLY-62/4/S4 HPCS-RLY-51V/A HPCS-RLY-51V/B HPCS-RLY-51V/C HPCS-RLY-52T/4/DG3 HPCS-RLY-E22B/K12 HPCS-RLY-R7X/DG3 HPCS-RLY-SDRX/DG3 HPCS-RLY-TD8/DG3	DG 3 output breaker trips	Modeled in fault tree for seismic failure with SOV fragilities and operator recovery.
DCW-TS-4 DLO-PS-26 HPCS-RLY-87G/A HPCS-RLY-87G/B HPCS-RLY-87G/C HPCS-RLY-OTR/DG3 HPCS-RLY-R2/DG3 HPCS-RLY-TD1/DG3	DG 3 failure to start or run	Modeled in fault tree for seismic failure with SOV fragilities and operator recovery.

Table 4.1-2 Disposition of Unscreened Electrical Contact Devices

ECD	Function	Disposition
E-CB-7/1 E-CB-7/71 E-CB-7/73 E-CB-7/75/1 E-CB-7/DG1 E-CB-B/7 LPCS-CB-P1 RHR-CB-P2A SW-CB-P1A E-CB-8/3 E-CB-8/81 E-CB-8/83 E-CB-8/85/1 E-CB-8/DG2 E-CB-B/8 RHR-CB-P2B RHR-CB-P2C SW-CB-P1B	4 kVac division 1 and 2 switchgear E-SM-7 and E-SM-8 unavailable	Modeled in fault tree for seismic failure with SOV fragilities and operator recovery.
E-RLY-86/7/DG1 E-RLY-86/B/7 E-RLY-86/8/3 E-RLY-86/8/DG2 SW-RLY-86/P1A	4 kVac Division 1 and 2 switchgear E-SM-7 / E-SM-8 locked out	Modeled in fault tree for seismic failure with SOV fragilities.
E-RLY-5051/7/71/A E-RLY-5051/7/71/C	4 kVac load center E-SL-71 de-energized	Modeled in fault tree for seismic failure with SOV fragilities and operator recovery.
E-RLY-5051/7/73/A E-RLY-5051/7/73/C E-CB-73/7A E-CB-73/7F	4 kVac load center E-SL-73 de-energized	Modeled in fault tree for seismic failure with SOV fragilities and operator recovery.
E-RLY-5051/8/81/A E-RLY-5051/8/81/C	4 kVac load center E-SL-81 de-energized	Modeled in fault tree for seismic failure with SOV fragilities and operator recovery.
E-RLY-5051/8/83/A E-RLY-5051/8/83/C	4 kVac load center E-SL-83 de-energized	Modeled in fault tree for seismic failure with SOV fragilities and operator recovery.

Table 4.1-2 Disposition of Unscreened Electrical Contact Devices

ECD	Function	Disposition
E-CB-71/7B E-CB-71/7C E-CB-81/8B E-CB-81/8C E-CB-83/8A E-CB-83/8F	4 kVac load centers E-SL-71, E-SL-81 and E-SL-83 de-energized	Modeled in fault tree for seismic failure with SOV fragilities and operator recovery.
SW-42-8BB6A SW-42-7BA7A SW-42-7BB6A SW-42-8BA10C	SW flow diversion	Modeled in fault tree for seismic failure with SOV fragilities.
FPC-42-8BA7B FPC-42-8BA8C FPC-42-7BA8A FPC-42-7BA8B	Fuel pool cooling containment isolation valves spuriously open	Modeled in fault tree for seismic failure with SOV fragilities and operator recovery.
E-CB-4/2 E-CB-4/41 E-CB-4/DG3 HPCS-CB-P1 E-RLY-81/4 HPCS-RLY-5051/A HPCS-RLY-5051/C	HPCS pump trips	Modeled in fault tree for seismic failure with SOV fragilities and operator recovery.
HPCS-42-4A2D HPCS-RLY-K22 HPCS-42-4A3C	HPCS pump suction valve, HPCS-V-1, isolates	Modeled in fault tree for seismic failure with SOV fragilities and operator recovery.
RHR-RLY-5051/P2A/A RHR-RLY-5051/P2A/C RHR-RLY-86/P2A RHR-RLY-K19A RHR-42-7BB7C RHR-42-7BB7B RHR-42-7BA5C RHR-42-7BA3B RHR-42-7BB5D RHR-42-7BB5C	LPCI train A failure to start or run	Modeled in fault tree for seismic failure with SOV fragilities and operator recovery.

Table 4.1-2 Disposition of Unscreened Electrical Contact Devices

ECD	Function	Disposition
RHR-42-8BB5D RHR-42-8BB5C E-RLY-62X1/8 RHR-RLY-5051/P2B/A RHR-RLY-5051/P2B/C RHR-RLY-86/P2B RHR-RLY-K19B RHR-42-8BA5C RHR-42-8BA5B RHR-42-8BA5D	LPCI train B failure to start or run	Modeled in fault tree for seismic failure with SOV fragilities and operator recovery.
E-RLY-62X1/8 RHR-42-8BA2D RHR-RLY-5051/P2C/A RHR-RLY-5051/P2C/C RHR-RLY-86/P2C	LPCI train C failure to start or run	Modeled in fault tree for seismic failure with SOV fragilities and operator recovery.
LPCS-RLY-5051/P1/A LPCS-RLY-5051/P1/C LPCS-RLY-50GX/P1 LPCS-RLY-86/P1 E-RLY-62X1/7 LPCS-42-7BA2A	LPCS failure to start or run	Modeled in fault tree for seismic failure with SOV fragilities and operator recovery.
MS-42-8BA6D MS-42-S11D1D	Main steam line drain isolation valves spuriously open	Modeled in fault tree for seismic failure with SOV fragilities and operator recovery.
CIA-42-7B7B CIA-42-7B7A CIA-42-8B7D	Supplemental nitrogen supply to main steam SRVs isolated	Modeled in fault tree for seismic failure with SOV fragilities and operator recovery.
RCIC-DPIS-13B RCIC-DPIS-7B RCIC-PS-12B RCIC-PS-12D RCIC-PS-22B RCIC-PS-22D RCIC-DPIS-13A RCIC-PS-12A RCIC-PS-12C RCIC-PS-22A RCIC-PS-22C	RCIC auto isolation	Modeled in fault tree for seismic failure with SOV fragilities and operator recovery.

Table 4.1-2 Disposition of Unscreened Electrical Contact Devices

ECD	Function	Disposition
RCIC-42-S21A10C	RCIC condenser vacuum valve, RCIC-V-69, isolates	Modeled in fault tree for seismic failure with SOV fragilities and operator recovery.
RCIC-42-S11D3C RCIC-42-S11D3B LD-MON-1B (K2) LD-MON-1B (K3) LD-RLY-K02B RCIC-42-8BA9D RCIC-RLY-K32 RCIC-RLY-K33 RCIC-RLY-K48 RCIC-RLY-K54 RCIC-RLY-K55 RCIC-RLY-K58 RCIC-RLY-K59 RCIC-42-S11D7A LD-MON-1A (K2) LD-MON-1A (K3) LD-RLY-K02A RCIC-42-S11D6C RCIC-RLY-K15 RCIC-RLY-K29 RCIC-RLY-K47 RCIC-RLY-K56 RCIC-RLY-K57 RCIC-RLY-K60 RCIC-RLY-K61	RCIC system valves spuriously operate; RCIC fails to run	Modeled in fault tree for seismic failure with SOV fragilities and operator recovery.
SW-PS-1A SW-PS-1B	SW train A and B pumps fail to start	Modeled in fault tree for seismic failure with SOV fragilities and operator recovery.
SW-RLY-5051/P1A/A SW-RLY-5051/P1A/C	SW train A pump fails to start or run	Modeled in fault tree for seismic failure with SOV fragilities and operator recovery.
RHR-42-8BB7D RHR-42-8BB7B SW-RLY-5051/P1B/A SW-RLY-5051/P1B/C SW-RLY-86/P1B	SW train B pump fails to start or run	Modeled in fault tree for seismic failure with SOV fragilities and operator recovery.

4.2 Walkdown Approach

This section provides a summary of the methodology and scope of the seismic walkdowns performed for the SPRA. Walkdowns were performed by personnel with appropriate qualifications as defined in the SPID. Walkdowns of those SSCs included on the seismic equipment list were performed as part of the development of the SEL, and to assess the as-installed condition of these SSCs for use in determining their seismic capacity and performing initial screening.

Several seismic walkdowns were performed for CGS in the past to support seismic evaluations, including the Individual Plant Examination of External Events (IPEEE) [26], NTTF 2.3: Seismic [27], and the Expedited Seismic Evaluation Program (ESEP) [28]. New walkdowns were performed for all SSCs on the CGS SPRA SEL. Available documentation from the prior walkdowns was consulted to supplement the data collected during the SPRA walkdowns. The prior walkdown information was used to inform the walkdown planning. The NTTF 2.3: Seismic and ESEP walkdown photographs and reports were used to the extent available to review items that were not accessible during the SPRA walkdowns (e.g., to evaluate the mounting of internals inside electrical cabinets that could not be opened during the SPRA walkdown).

The objectives of the SPRA seismic walkdown included the following:

- Collect information that may assist the systems analyst in deciding whether to add or remove SSCs from the SEL.
- Observe and document the current as-built condition of SSCs included on the SEL.
- Identify SSCs that are seismically rugged and can be removed from the SPRA model.
- Assess whether each SSC satisfies the requirements of the EPRI NP-6041-SL [22] screening tables and assign a seismic capacity ranking: Rugged, High, Medium, or Low. These rankings assist the development of representative seismic fragilities to help identify and prioritize the risk-significant SSCs for detailed fragility evaluation, as discussed in Section 2 of the seismic walkdown report [29].
- Identify realistic failure modes (e.g., functionality, structural integrity, or anchorage failure) of the SEL components, and identify further information (drawings, design analysis reports, and / or seismic qualification reports) that may be required for their fragility evaluation.
- Collect key data such as dimensions, materials, physical condition, and configuration that may be required in future fragility evaluations.
- Identify conditions or configurations that could potentially have an influence on the seismic fragility of an SSC.
- Identify SSCs within each equipment class that are the same or similar to other SSCs. Similarity information will be used as basis for bounding case analyses. It may also be used by the systems analyst in consultation with

the fragility analysts to inform decisions and sensitivity studies to address correlation in the systems modeling effort. The seismic plant response modeling and risk quantification documentation address correlation assumptions and sensitivities based on the walkdown observations and the fragility evaluations.

- Identify SSCs that are not required for safe shutdown but whose structural failure or deformations may adversely influence nearby SEL items or undermine pathways required for operator actions; this includes review for seismic spatial (i.e., II/I) interaction, seismic-induced fire, seismic-induced flood concerns, and masonry walls.
- Observe and record seismic deficiencies.

Walkdowns were performed in accordance with guidance in SPID Table 6.5 and the associated requirements in the PRA Standard. Additional walkdown criteria were developed in accordance with EPRI NP-6041-SL [22]. Seismic capacity rankings of “Rugged”, “High”, “Medium”, and “Low” were developed as described in Section 2 of [29] to inform the development of preliminary fragilities for importance ranking. EPRI NP-6041-SL states that the seismic review team (SRT) should review a sample from each group of similar SEL items in full detail (i.e., “full scope walkdown”) and review the remaining reasonably accessible SEL items using a “walk-by” to confirm similarity and record any relevant differences in configuration. The SRT members judged items to be similar based on equipment construction, dimensions, seismic qualification requirements, anchorage type, and configuration. In a full scope walkdown, the SRT collected detailed notes on the equipment configuration and performed a detailed review against the criteria and caveats from Appendix F to EPRI NP-6041-SL [22]. The SRT also recorded photographs, measurements, sketches, and any other data that can be used to inform the seismic fragility evaluations. One lead item (or a sample of similar items) was reviewed using a full scope walkdown. Other similar components were reviewed briefly to confirm similarity with the lead items and to check for anomalies in installation and / or dissimilar spatial interaction concerns.

The walkdown effort was conducted during three primary walkdown periods, plus limited-scope supplemental walkdowns, as follows:

- During three two-week periods from 31 October 2016 through 11 November 2016, 3 April 2017 through 14 April 2017, and 22 May 2017 through 2 June 2017. These walkdowns covered all items on the then-current SEL. The last walkdown coincided with a plant maintenance outage and included walkdown of SEL items inside Containment.
- On 13 September 2017, 18 January 2018, and 28 February 2019. These supplemental walkdowns were performed to examine specific details, e.g., items added to the SEL and internals of cabinets that could not be opened during previous walkdowns.

4.2.1 Significant Walkdown Results and Insights

Consistent with the guidance from EPRI NP-6041-SL [22], no significant adverse findings were noted during the CGS seismic walkdowns. The SSCs on the SEL were observed to be of generally good seismic design, construction, anchorage, and maintenance. They were also observed to be largely free of significant seismic interaction concerns and obvious signs of degradation or corrosion. A few adverse seismic interaction conditions primarily affecting chatter-sensitive components were observed, some of which were resolved by CGS and the rest were considered in the seismic fragility evaluation.

Of the nearly 1,300 individual components walked down, nearly all were ranked *Rugged*, *High*, or *Medium*, and about ten components were ranked *Low*. *Rugged*-ranked components may be removed from the SPRA model or assigned a high enough representative fragility such that they do not contribute significantly to seismic risk. *High*-ranked components can be considered to have a high-confidence-low-probability-of-failure (HCLPF) in-structure S_a capacity of at least 1.8g. *Medium*-ranked components typically require evaluation of anchorage or other conditions, e.g., interactions to ensure they do not control, before a HCLPF in-structure S_a capacity of at least 1.8g can be assigned. *Low*-ranked components were non-safety-related items, e.g., some FLEX components, and were assigned low seismic capacities.

Components on the SEL were evaluated for seismic anchorage and interaction effects, and effects of component degradation, such as corrosion and concrete cracking, for consideration in the development of SEL fragilities. In addition, walkdowns were performed on operator pathways, and the potential for seismic-induced fire and flooding scenarios was assessed. The walkdown observations were adequate for use in developing the SSC fragilities for the SPRA.

4.2.2 Seismic Equipment List and Seismic Walkdowns Technical Adequacy

The CGS SPRA SEL development was subjected to an in-process independent technical review by industry experts. Comments from the review were resolved satisfactorily.

The CGS SPRA SEL development and walkdowns were subjected to an independent peer review against the full set of applicable requirements (i.e., the relevant seismic fragility analysis (SFR) and seismic plant-response analysis (SPR) requirements) for Capability Category II in the PRA Standard [4].

The peer review assessment, and subsequent disposition of peer review findings, are described in Appendix A, and establish that the CGS SPRA SEL and seismic walkdowns are suitable for this SPRA application.

4.3 Dynamic Analysis of Structures

This section summarizes the dynamic analyses of structures that contain systems and components important to achieving a safe shutdown, using Soil Structure Interaction (SSI) Analyses.

4.3.1 Fixed-base Analyses

Since CGS site is a soil site, SSI analyses were required to determine the building responses and in-structure response spectra (ISRS) needed for the determination of SSC fragilities for the SPRA. SSI analyses were used to develop ISRS for the RB, RWCB, DGB, TB, and CSTs. Seismic input to SEL components located in other CGS structures is not significantly affected by SSI.

The Service Building is a lightweight steel-framed structure with a large basement that moves with the surrounding soil. The Service Building does not house SEL components, but its collapse can damage the CST piping. Therefore, only the forces in the main force resisting system needed to evaluate structure fragility were required. This structure was therefore analyzed as a fixed-base lumped-mass stick model (LMSM) using Response Spectrum analysis.

4.3.2 Soil Structure Interaction Analyses

Probabilistic seismic response analysis including SSI effects was performed in compliance with the guidance of ASCE/SEI 4-16 [30]. The probabilistic response analysis implementation followed the approach documented in NUREG/CR-2015 [31], which implements Latin Hypercube Sampling (LHS). Variables in the LHS included the strain-compatible soil profiles, structure frequency and damping, and the earthquake acceleration time histories. New finite element models of the structures were developed for the SSI analyses. The SSI simulations were performed using computer program CLASSI [32]. Thirty CLASSI simulations were developed by stratified sampling from each of these variables. The sample space of each variable was divided into thirty equal-probability bins. From each probability bin, one representative value was selected for the variable.

The SSI analysis was performed for a Reference Earthquake ground motion spectrum that represents the seismic hazard levels that contribute most significantly to the SPRA outcome. The Reference Earthquake was selected to be the Mean 1E-5 Uniform Hazard Response Spectrum (UHRS), i.e., ground motions that have Mean Annual Frequencies of Exceedance (MAFE) of 1E-5 /yr at the soil surface in the horizontal direction. A risk-consistent vertical ground motion spectrum was developed for the Reference Earthquake (Section 4.1 in [33]). Thirty sets of three independent components of time history records were developed using spectral matching. The time histories were conditioned such that the median spectral accelerations for the ensemble of time histories match the 5% damped spectra over the frequency range of 0.5 Hz to 50 Hz. These time histories were then modified to include earthquake component variability using randomly generated scale factors from lognormal probability distributions with median values of 1.0 and logarithmic standard deviations for the horizontal and vertical components of 0.18 and 0.25, respectively.

The SSI analysis developed a set of thirty strain-compatible soil profiles consistent with the Reference Earthquake ground motions. A soil column of 85 ft depth was considered for the SSI analysis. The soil stratum below this depth is significantly stiffer than above and was represented by an elastic half-space. A statistically representative set of thirty profiles was selected resulting from the probabilistic site response analyses for 1-5 MAFE horizontal ground motions (refer to Section 3). Selecting from these profiles preserves the underlying correlations between soil layer properties and is therefore preferred over randomly generating uncorrelated properties from each soil layer statistics.

Thirty structure frequency and damping scale factors were generated from lognormal probability distributions with median values of 1.0 and logarithmic standard deviations of 0.15 and 0.35, respectively.

A list of structures and description of relevant parameters is listed in Table 4-2. The SSI analysis methodology and results are summarized in [34].

4.3.3 Structure Response Models

Structure response was developed using new finite element models and the LHS methodology described in Section 4.3.2. Review of the existing LMSMs of the CGS structures indicated that they were generally incapable of representing realistic in-structure responses, especially torsional coupling and floor slab vertical responses, and as such they did not satisfy the modeling requirements in the SPID. The new finite element models represented median properties of each structure using computer program SAP2000 [35]. The model development generally implemented the requirements of ASCE/SEI 4-16 [30]. Each model represented the significant seismic load-resisting structural components. Beams and columns were modeled by frame elements. Walls, floors, and roofs were modeled by area elements. The structure models included the structure masses and the best estimates of non-structural masses likely to be concurrent with a significant seismic event. The best estimates of these masses were based on field walkdown observations or design documents.

Fixed-base modal analyses were performed for each structure model. These analyses confirmed that the model properly captures the dynamic characteristics of the structure and has reasonable mode shapes and frequencies. Fixed-base static 1g analyses were performed for each structural model by imposing 1g accelerations in each of the three orthogonal directions. Review of the structure displacements, forces, and foundation reactions confirmed that the load path and overall seismic mass and stiffness are appropriately represented. The fixed-base models were not used to determine structure response or structural fragilities except for the Service Building collapse fragility (refer to Section 4.3.1).

The stiffness of concrete elements expected to crack at the Reference Earthquake ground motion was adjusted following the provisions of ASCE/SEI 4-16 [30]. For each of the SAP2000 structure models, a preliminary seismic response analysis was performed using uncracked concrete properties to assess the expected extent of concrete cracking and seismic response level. Structure damping was assigned based on the expected response levels following the recommendations of [30]. As discussed in Section 4.3.3, the probabilistic structure response analysis accounted for variability in structure stiffness and damping using the LHS Method.

Two sets of probabilistic response analysis were performed for each structure. The first analysis set considered all the LHS variables and represented the composite structure response variability from sources of both randomness and uncertainty. The second analysis set considered only the sources of randomness in order to separate out randomness and uncertainty. Randomness was considered to be due to the earthquake ground motion only. The median soil profile and structure stiffness and damping were used in this analysis set and only the earthquake acceleration time histories were varied.

For each analysis set, thirty ISRS were calculated from the acceleration time histories output at each of selected locations in each building for ten damping ratios ranging from 1% to 20%. These locations cover the geographic extent of systems and components included on the CGS SEL. Median and 84% non-exceedance probability (NEP) ISRS were generated at each location and direction for each damping ratio. The displacement time histories were calculated by double-integrating and baseline-correcting the acceleration time histories output. Median and 84% NEP displacements relative to the free field and relative building-to-building were calculated at representative locations.

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

Structure	Foundation Condition	Type of Model	Analysis Method	Comments/Other Information
Reactor Building (RB)	Soil	FE	Probabilistic SSI	Partially bonded embedded foundation. Foundation impedance and scattering matrices were developed using computer program SASSI2010 [36] for input into CLASSI.
Radwaste / Control Building (RWCB)	Soil	FE	Probabilistic SSI	
Diesel Generator Building (DGB)	Soil	FE	Probabilistic SSI	The SSI analyses included structure-soil-structure

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

Structure	Foundation Condition	Type of Model	Analysis Method	Comments/Other Information
				interaction (SSSI) effects from the adjacent RB and RWCB.
Turbine Building (TB)	Soil	FE	Probabilistic SSI	Houses no SSCs on the SEL, but its potential collapse is a seismic interaction concern for the adjacent structures.
Condensate Storage Tanks (CSTs)	Soil	FE	Probabilistic SSI	
Service Building	Soil	LMSM	Response Spectrum	Lightweight steel structure with large basement that moves with surrounding soil. Only structure collapse fragility is of interest.

4.3.4 Seismic Structure Response Analysis Technical Adequacy

The CGS SPRA Seismic Structure Response and Soil Structure Interaction Analysis were subjected to an in-process independent technical review by industry experts. Comments from the review were resolved satisfactorily.

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

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

4.4 SSC Fragility Analysis

The SSC seismic fragility analysis considers the impact of seismic events on the probability of SSC failures at a given value of a seismic motion parameter. The seismic motion parameter for the CGS SPRA was selected to be the 5% damped horizontal ground spectral acceleration at the soil surface at a frequency of 2.5 Hz. This parameter is representative of the ground motion significant to the response of the major CGS structures, all of which have median fundamental SSI frequencies between 1.6 Hz and 3 Hz. The surface hazard curve slopes at these frequencies are comparable and are significantly different from the slope of the PGA hazard curve. Therefore, using the PGA hazard curve for risk quantification could lead to unacceptable bias. The selected ground motion parameter used for the CGS SPRA was reviewed by a panel of industry experts who agreed that this parameter would provide the most realistic results for use in the SPRA. The fragilities of the SSCs that participate in the SPRA accident sequences, i.e., those included on the seismic equipment list (SEL), are addressed in the model. Seismic fragilities for the significant risk contributors, i.e., those which have an important contribution to plant risk, are intended to be generally realistic and plant-specific based on actual current conditions of the SSCs in the plant, as confirmed through the detailed walkdown of the plant.

This section summarizes the fragility analysis methodology and presents a summary of the fragilities, calculation method and failure modes for those SSCs determined to be sufficiently risk important, based on the final SPRA quantification (Section 5). Important assumptions and important sources of uncertainty, and any particular fragility-related insights identified, are discussed.

4.4.1 SSC Screening Approach

Screening of SSCs was performed based on the following two criteria:

- Inherently rugged SSCs: SSCs considered to be rugged were those judged by the SRT to have a very high seismic capacity, such that they should have negligible contribution to seismic risk, as discussed in Section 5.1 of Report 168059-R-04 [37]. Examples of items assessed to be rugged by the SRT include check valves, local instruments, sensors, transmitters, manual valves that are not required to change state, and terminal boxes. The SRT reviewed these components on the SEL during the seismic walkdown to confirm that they were obviously robust, lightweight, well anchored, and free of interaction concerns or signs of physical distress.
- Capacity-based screening: The capacity-based screening, also referred to as the screening-level fragility, is discussed in Section 9 of [37]. While it was first developed as the fragility for which convolution with the mean hazard curve meets the thresholds recommended for SCDF and SLERF in Section 6.4.3 of the SPID [2], the SPRA systems analysts developed an alternate screening-level fragility using the SPRA model such that the F-V of a screened-out component is less than about 0.005. The latter screening-level fragility was more stringent and was adopted for the SPRA. This screening-level fragility was used for only three components. These components were not screened from the SPRA plant-response model but instead were retained and assigned the screening-level fragility.

4.4.2 SSC Fragility Analysis Methodology

Fragility evaluations were performed for function and anchorage of each SSC, and where applicable, for spatial interaction concerns, seismic-fire and/or seismic-flood interactions, chatter, and structural integrity of the SSC or its supporting structure. Seismic demands were based on the probabilistic structure response analysis (Section 4.3). Seismic functional capacities were obtained from component-specific plant seismic qualification tests and industry-wide compilations of test-based capacities for similar components as compiled in EPRI NP-7147-SL [38], [39] and [40], EPRI NP-5223-SL [41] and EPRI 3002010668 [42] as well as earthquake experience-based capacities according to EPRI 1019200 [43] and EPRI 3002011627 [44]. Anchorage evaluations were performed separately for components. The concrete breakout strength in tension for cast-in-place headed and hooked anchor bolts with no edge distance or spacing limitations was determined following the recommendations of EPRI 3002008099 [45]. The anchorage strengths of other anchors was determined by the Concrete Capacity Design (CCD) Method [46] if International Code Council (ICC) evaluation reports signifying that the anchors satisfy current acceptance criteria were available, or using the provisions in Appendix O to EPRI NP-6041-SL [22] and EPRI NP-5228-SL [47] if not. Median and design strengths of concrete and steel structural components were determined based on the applicable criteria in material codes, e.g., ACI 349-13 [48] and ANSI/AISC N690-12 [49]. The comprehensive set of criteria for CGS fragility evaluation is presented in Section 8 of 168059-CD-01 [33].

The following graded approach was followed to focus the fragility effort on the risk-significant SSCs. The fragility evaluation method used for each SSC increased in rigor according to its risk significance, as follows:

- Representative (i.e., preliminary) fragilities were developed for all the SSCs on the final SEL using plant-specific SSC demands and capacity data. These fragilities were based on the Hybrid Method but used simplified calculations and conservatively biased judgment to preserve resources and avoid missing a potentially important SSC.
- Potentially risk-significant SSCs were identified based on preliminary risk quantifications using the representative fragilities.
- Detailed fragilities were developed for the risk-significant SSCs.
    - For the major risk-significant SSCs, the Separation of Variables (SOV) Method, which is considered to be more rigorous, was used. SPID [2] stipulates that the SOV Method be used for the risk-dominant SSCs that are major contributors to the plant seismic risk.
    - For other risk-significant SSCs, either the SOV or Hybrid Methods were used.
- Additional risk-significant SSCs were identified based on subsequent risk quantifications that used the available detailed fragilities. Additional detailed fragilities were developed for these SSCs.
- This iterative process was followed until the risk quantification results became stable.

The Hybrid Method introduced in the SPID [2] estimates the HCLPF seismic capacity and then uses generic variability parameters to estimate the median capacity. The CGS Hybrid fragilities followed a more rigorous approach, which is presented in Section 8.3 of [33]. In this approach, the HCLPF and median seismic capacities were each estimated independently for each SSC, and SSC-specific variability parameters were calculated using these two capacities. A minimum value on the variability parameter was imposed to ensure that the calculated values are reasonable.

The structure fragilities of the RWCB and TB dominated seismic risk, followed by the RB structural fragility. Influential equipment fragilities were typically comparable to or better than these structure fragilities, which has typically not been the case in past SPRA quantification. This was attributed to three main reasons: (1) CGS is a relatively young plant within the U.S. nuclear power fleet and has more robust equipment of newer vintage; (2) the variability in the RWCB structure response is particularly high, which extends the lower tail of the fragility distribution towards low ground motions that have higher risk significance; and (3) while the structures were well designed, their horizontal frequency ranges approached the peak of the horizontal ground spectrum at 3.3 Hz. Consequently, the structure seismic demands are nearly driven by the peak in the ground spectrum, while the higher frequency content in the ground motion is filtered out by the low frequencies of the structure. Accordingly, no significant ISRS amplification takes place at the typical equipment frequencies of 5 Hz to 15 Hz.

Given the importance of these structure fragilities, fragility evaluations using nonlinear pushover analyses were performed to obtain more realistic parameters. The pushover analysis accounted for the progression of structure member yielding and force redistribution to other structure members that form the seismic load path. This included yield capacity evaluations of both shear walls and floor diaphragm potential critical sections. The inelastic energy dissipation was evaluated explicitly considering the effects of SSI for each structure. Pushover analysis of the TB also established that its collapse cannot result in overall failure of the RWCB and can only result in overall failure of the RB with a best estimate probability of 0.2. Detailed fragilities for all three major CGS building structures were developed using the more rigorous SOV Method.

The CGS equipment generally had substantial seismic fragilities. The SPRA risk quantification indicated that motor control centers (MCCs) in the RWCB and DGB were significant contributors to seismic risk. Detailed fragilities were developed for these MCCs using the SOV Method. These fragilities were governed by mounting weld configurations between the base angles and the steel base frames and evaluated inelastic energy dissipation in these flexible connections.

Detailed fragilities were developed using the SOV Method for all chatter-sensitive devices for shaking and impact (for components where impact between adjacent components or building was a realistic concern for chatter). Detailed SOV fragilities were developed for several other equipment components, including the main control room (MCR) AHUs, CSTs, and SW pond spray rings.

Detailed fragilities were developed using the Hybrid Method for about ninety SSCs, including minor structures, NSSS components, MCR electrical panels and ceiling, RB MCCs, RWCB and DGB AHUs, and emergency core cooling system (ECCS) pumps. These SSCs had minor risk contributions. The remaining SSCs typically had insignificant risk contributions.

Detailed guidance was developed on modeling correlations between seismic fragilities (Section 10 of the CGS fragility report [37]). Correlation between the structure fragilities of the RB, RWCB, and TB had a significant influence on realism in the plant risk. Verification of the modeling and implementation of this guidance used Monte Carlo Simulation and other methods.

The fragility analysis evaluated building-to-building impact and relative displacement effects on SSC fragilities. It concluded that the only SSC fragilities governed by building impact involved component chatter. Other potentially affected SSCs had higher fragility margins for the effects of building impact and relative displacements than they had for shaking.

Non-vibratory seismic-induced hazards, including ground failures, upstream dam breaches, and potential release of radioactive material from the Hanford Reservation were screened and found to be insignificant contributors to plant risk (Section 11.3 of [37]).

Fragility parameters were developed for input to sensitivity and uncertainty analyses performed by the SPRA analyst. This input addressed potentially significant judgments, simplifications, conservatism, or modeling uncertainties in the fragility evaluations (Section 12 of [37]).

4.4.3 SSC Fragility Analysis Results and Insights

Refer to Section 5 for a tabulation of the fragilities for the SSCs and correlated SSC groups determined to be significant risk contributors based on the final SPRA quantification.

4.4.4 SSC Fragility Analysis Technical Adequacy

The CGS SPRA SSC Fragility Analysis was subjected to an in-process independent technical review by industry experts. Comments from the review were resolved satisfactorily.

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

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

5.0 Plant Seismic Logic Model

This section summarizes the adaptation of the CGS FPIE PRA model to create the seismic PRA plant response (logic) model.

The seismic plant response analysis models the various combinations of structural, equipment, and human failures given the occurrence of a seismic event that could initiate and propagate a seismic core damage or large early release sequence. This model is quantified to determine the overall SCDF and SLERF and to identify the important contributors, e.g., important accident sequences, SSC failures, and human actions. The quantification process also includes an evaluation of sources of uncertainty and provides a perspective on how such sources of uncertainty affect SPRA insights. No portions of the internal events PRA top-logic fault tree were removed; FPIE PRA initiating events that are not applicable to seismic events were excluded from the SPRA quantification by setting these initiating events to logical false.

5.1 Development of the SPRA Plant Seismic Logic Model

The CGS seismic response model was developed by starting with the CGS internal events at-power PRA model of record as of January 28, 2019, and adapting the model using CAFTA 6.0b [50] and FRANX 4.2 [51] in accordance with guidance in the SPID [2] and PRA Standard [4], including adding seismic fragility-related basic events to the appropriate portions of the internal events PRA, and adjusting the internal events PRA model human reliability analysis to account for response during and following a seismic event.

CGS SPRA logic model development employed the methods described in the following subsections.

5.1.1 General Approach

The CGS SPRA top-logic model is based on the CGS FPIE top-logic model. The FPIE top-logic model includes modeling of the extended loss of ac power (ELAP) FLEX strategy, which was carried over to the SPRA model development. Both seismic and random SSC failures are accounted for in the CGS SPRA model. No portions of the CGS FPIE PRA top-logic fault tree model were removed during the SPRA development. Initiating events not applicable to the SPRA modeling, such as the loss of feedwater initiator (which is subsumed by a loss of offsite power) are set to logical false (value 0) by FRANX 4.2 [51].

The FPIE top-logic model was modified using CAFTA 6.0b [50] to include fault tree logic specific to model plant response following seismic events, including [52]:

1. Seismic-specific accident sequences, such as loss of control room function,
2. Initiating events that proceed directly to core damage,
3. Initiating events that proceed directly to large early release,

4. Fault tree logic and basic events to support mapping fragilities to SSCs not explicitly modeled by the FPIE PRA,
5. System trains added to support the SPRA modeling,
6. Human failure events added to address seismic events,
7. Contact chatter scenarios, and
8. Seismic-induced fire and seismic-induced flooding scenarios.

The CGS seismic hazard is discretized into 46 intervals and these intervals are treated as seismic initiators with assignment of interval-specific seismic fragilities, and summation of the PRA solution over all intervals.

The FRANX 4.2 XINIT option [51] was used to integrate the seismic hazard and plant response and to produce a top-logic SPRA model capable of quantifying SCDF and SLERF outside of FRANX. SPRA cutsets are produced by quantifying the top-logic SPRA model in PRAQuant 5.2 [53] using FTREX 1.8 [54] and QRecover 2.08 [55]. The SPRA results are produced by processing the SPRA cutsets in ACUBE 2.0 [56].

5.1.2 Selection of Seismic Initiating Events and Consequential Events

Seismic initiating events were systematically selected by reviewing:

1. FPIE initiating events for applicability to seismic events,
2. SSCs on the CGS master equipment list for applicability to seismic events,
3. The list of CGS structures for applicability to seismic events if damage occurs,
4. Interfacing systems loss of coolant accident (ISLOCA) pathways,
5. Review of earthquake operating experience at world nuclear power plants,
6. IPEEE, other external hazards,
7. Combinations of seismic initiating events to ensure that seismic initiating events that occur in combination were addressed by the SPRA model (e.g., loss of offsite power (LOSP) and loss of cooling accident (LOCA)).

CGS seismic initiating events include the following:

1. Seismic failures that lead directly to core damage:
    a. Loss of control room function with concurrent LOCA,
    b. Piping failures in RB 471, 501 or 522 elevations, or
    c. Cable tray failures in ECCS pump rooms, RB 422 or 441 elevations.
2. Seismic failures that lead directly to large early release:
    a. Structural failure of RWCB or RB,

   b. Loss of control room function with break / leak outside containment,
   c. Structural failure of main control room ceiling light fixture anchorages,
   d. Structural failure of RPV internals, or
   e. Damage to RPV level reference leg piping,
3. Loss of control room function,
4. Loss of control room HVAC,
5. Break / leak outside containment:
   a. SCRAM discharge volume (SDV) integrity fails,
   b. Leakage through SDV vent and drain valves,
   c. Main steam leak or break outside containment,
   d. RCIC steam line break outside containment, or
   e. Reactor water cleanup (RWCU) break outside containment,
6. ISLOCA,
7. LOCA (small, medium and large), and
8. LOSP.

CGS SPRA consequential events consist of those carried over the FPIE PRA: induced LOCAs (SRVs stuck open or RPV overpressurization), anticipated transient without SCRAM (ATWS), station blackout (SBO), ELAP; as well as seismically-induced fires and seismically-induced floods.

5.1.3 Seismic PRA Accident Sequence Development

New accident sequences identified by the SPRA development were found to be represented by existing FPIE event trees. The new accident sequences were developed by making copies of applicable FPIE accident sequence logic, uniquely labelling the fault tree gates, and adding this new sequence logic to the appropriate locations in the SPRA top-logic model.

5.1.4 Modeling Assumptions

The CGS SPRA important modeling assumptions consist of the following:

1. The SPRA models sequences that involve a loss of offsite power. Offsite power is not recoverable during the SPRA mission time. Sequences which do not involve a loss of offsite power are not included in the SPRA quantification. However, LOSP was not credited for eliminating or limiting the consequences of failure of items that would lose power in the event of LOSP.

2. The effectiveness of secondary containment (reactor building) to mitigate releases from primary containment (drywell / wetwell) is not credited by the SPRA model, consistent with the FPIE PRA model.
3. Seismic-fire scenarios are modeled as a full burn-up at time zero of the physical analysis unit in which the ignition source is located. No credit is modeled for fire suppression. These conservatisms do not significantly impact the SPRA results [57].
4. Fully correlated response of same or very similar equipment in the same structure and elevation is assumed.
5. Failure of the diesel generator room mixed air HVAC ducts are assumed to fail the respective diesel generator. DG room heatup calculations have not examined the condition in which the HVAC ducts have been damaged. In the absence of dedicated calculations, this approach represents a conservative bias.

5.1.5 Large Early Release Frequency Model

SPRA large early release sequences are based on the FPIE PRA and supplemented by seismic-induced LERF contributors, such as RB or RWCB damage at seismic levels above the fragilities established for these structures [52]. Additional containment isolation pathways applicable to seismic events were identified and modeled by the SPRA.

5.1.6 Contact Chatter Scenarios

All unscreened contact chatter scenarios identified by the contact chatter assessment were incorporated into the SPRA. The potential for operator recovery from each chatter scenario was examined, and operator actions were included in the SPRA modeling.

5.1.7 SSC Response Correlation

The SPRA fully correlates the responses of same or very similar equipment in the same structure and elevation. Partial correlation is explicitly modeled for the composite fragility of three CGS structures: RB, RWCB and TB. No other partial correlation is modeled.

The resulting correlated component groupings include the following, as applicable:

- RWCB, RB and TB structures (partial correlation),
- Division 1 and 2 switchgear,
- Division 1 and 2 load centers,
- Division 1 and 2 motor control centers,
- Onsite diesels DG-1 and DG-2,
- Division 1 and 2 diesel generator auxiliary components (e.g., starting air tanks; cooling water reservoirs; control panels; fuel transfer pumps),

- Transformers,
- Inverters,
- Battery chargers,
- Batteries,
- Electrical distribution panels,
- Air-handling units,
- Fan coolers,
- Electrical contact device contact chatter groups (42 correlation groups),
- Main control room control panel / relay panel groups (five correlation groups),
- Control panels,
- Containment isolation valves (correlation of inboard with outboard valves),
- Storage tanks (e.g., underground diesel fuel),
- System valves that have similar locations and designs (e.g., SCRAM discharge volume vent and drain valves),
- Heat exchangers,
- Hydraulic control unit assemblies,
- SCRAM discharge volume tanks,
- Main steam safety relief valves,
- Air accumulators, and
- Instrument racks.

5.1.8 Seismic Human Reliability Analysis

The CGS seismic human reliability analysis (HRA) follows the guidance of EPRI 3002008093 [58]. A systematic approach was used to identify the internal events human failure events (HFEs) that should be included in the seismic PRA. Additional actions were added to the SPRA in cases where such actions would impact the overall assessment of risk.

Risk-significant HFEs were given detailed treatment in the SPRA. Adjustments were made in plant damage state-specific analyses for each of the HFEs that received detailed HRA. Additional fault tree modeling was provided for detailed HFEs to determine which damage state version of the HFE is able to be expressed in the fault tree logic. Plant damage state definitions were adopted from [58].

For non-significant HFEs, failure probabilities were adjusted to account for the effect of seismic conditions on human error probability (HEP) development using the EPRI screening approach. This was performed by applying integrated performance shaping factors using the methodology presented in [58].

A seismic HRA dependency analysis was performed and incorporated into the SPRA quantification to account for the dependencies between multiple HFEs in the same cutset [52].

5.1.9 Seismic-Induced Fire

Seismic-induced fires were postulated and modeled using the EPRI methodology for seismically induced fire [59]. Potential sources of seismic-induced fire were identified as follows [52]:

1. A seismic-induced fire global analysis boundary was defined where seismic-induced fires were postulated to occur. This boundary includes all plant locations that contain equipment with the potential to produce a fire and impact equipment on the SEL and/or electrical cables that support equipment on the SEL.
2. The plant master equipment list [24] was analyzed during the SEL development to identify SSCs if, and only if: a) the SSC poses the potential to produce a seismic-induced fire; and b) located in physical analysis units that could affect equipment credited by the SPRA. If an SSC failed to meet either of these criteria, it was screened from seismic-induced fire scenarios.

Seismic-induced fires were modeled in the SPRA for all ignition sources identified by this review, unless the fire produces an impact no different from loss of the component itself and totaled about 40 scenarios [52]. Due to the potential challenges to suppress a fire given a seismic event, no credit is taken for fire suppression, and the fire scenarios are modeled as full burn-up of the physical analysis units in which the ignition sources exist. Seismic-induced fires were modeled by combining the fragility of the ignition sources with their conditional probabilities of producing a fire, given damage.

5.1.10 Seismic-Induced Flood

Seismic-induced floods were postulated and modeled using the EPRI methodology for seismically induced flood guidance [59]. Potential sources of seismic induced flood were identified as follows [52]:

1. A seismic-induced flood global analysis boundary was defined where seismic-induced floods were to be postulated to occur. This boundary includes all plant locations that contain equipment with the potential to produce a seismic-induced flood that can impact equipment on the SEL.
2. An inventory of all equipment within the global analysis boundary with the potential to cause a seismic-induced flood was made, based on a review of the CGS master equipment list [24], and walkdown observations.

Seismic-induced floods were modeled in the SPRA for all flooding sources identified by this review by mapping flood sources to flood scenarios modeled by the FPIE PRA or, for flood scenarios specific to seismic events, to applicable targets [52]. A total of about 50 internal flooding scenarios were modeled.

CGS is a dry site [60]. There are no external flooding scenarios capable of damaging any equipment credited by the SPRA.

5.1.11 Screening of SSC Failure Modes

All SSC failure modes were modeled in the SPRA unless the SSC was deemed by the fragilities team to be rugged, the SSC damage produces no impact to the SPRA or the impact is enveloped or addressed by another SSC [52].

5.1.12 Incorporating Seismic Failures into the System Fault Trees and PRA Model Database

FRANX 4.2 was used to integrate the seismic hazard and plant response. Fragility groups were defined for all SEL SSCs that can impact CGS plant response following an earthquake. The SPRA models 425 fragility groups [52]. These groups were incorporated into FRANX 4.2 and mapped to basic events in the CGS SPRA top-logic fault tree model that correspond to the specific SSC impacts produced by each fragility group. The XINIT feature in FRANX 4.2 [51] was then used to produce a top-logic fault tree to quantify the SPRA outside of FRANX using PRAQuant 5.2 [53]. PRAQuant 5.2 was then utilized to produce the SPRA cutsets, and ACUBE 2.0 [56] was used to produce the SPRA results.

5.2 SPRA Plant Seismic Logic Model Technical Adequacy

The CGS SPRA seismic plant response methodology and analysis were subjected to an independent peer review relative to Capability Category II for the full set of SPR supporting requirements in the Standard [4].

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

5.3 Seismic Risk Quantification

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

5.3.1 SPRA Quantification Methodology

Quantification of the SPRA top-logic model is performed using PRAQUANT 5.2 [53], FTREX 1.8 [54] and QRecover 2.08 [55]. FTREX generates the SPRA cutsets. QRecover applies cutset post-processing rules. After the initial quantification, ACUBE 2.0 [56] is used to produce more precise top event frequency estimates and to generate importance measures. ACUBE employs a binary decision diagram (BDD) algorithm to minimize overcounting in the Boolean summation of SPRA cutsets. Software employed by the SPRA quantification has been vetted through the CGS software acceptance process.

The human reliability analysis dependency analysis (HRADA) is implemented in the CGS SPRA via QRecover 2.08 recovery rules [52].

5.3.2 SPRA Model and Quantification Assumptions

Plant-specific model assumptions made in the hazard analysis, structures / fragilities analysis, and plant response modeling that are key to the SCDF or SLERF results (i.e., assumptions for which the use of an alternative consensus method or hypothesis would significantly alter the results or insights) consist of the following:

1. The effectiveness of secondary containment (reactor building) to mitigate releases from primary containment (drywell / wetwell) is not credited by the SPRA model, consistent with the FPIE PRA model..
2. Building-to-building impact has an uncertain effect on chatter events. The fragilities of chatter-sensitive contact devices considered impact between two building floors to result in simultaneous chatter of all the components located on those floors.
3. The CGS SPRA quantification employs fragilities truncation. The Expert Panel on Quantification in Seismic Margins recommendation in NUREG/CR 4334 [61] described the HCLPF capacity of a component as corresponding to the earthquake level at which it is extremely unlikely that failure of the component will occur. This recommendation is interpreted by the CGS SPRA development to mean that SSC conditional failure probabilities at ground motions below a threshold that corresponds to high confidence in a low failure probability can be considered to be practically zero. Fragility truncation at the CGS HCLPF capacities (i.e., 95% confidence and 5% probability of failure) was considered by the CGS SPRA development team to be potentially unconservative. Therefore, fragilities are truncated in the SPRA quantification below ground motions that correspond to at least 95% confidence in at most 1% probability of failure.
4. The CGS offsite power loss fragility estimate is typical of that employed by industry SPRAs. The offsite power loss fragility estimate is a source of modeling uncertainty. A consensus reasonable alternative approach is not currently available.
5. Failure of the diesel generator room mixed air HVAC ducts are assumed to fail the respective diesel generator. DG room heatup calculations have not examined the condition in which the HVAC ducts have been damaged. In the absence of dedicated calculations, this approach represents a conservative bias.
6. A reference ground motion of 2.5 Hz is selected for the CGS SPRA as most characteristic for the site. This parameter is representative of the ground motion significant to the response of the major CGS structures,

all of which have median fundamental SSI frequencies between 1.6 Hz and 3 Hz. The surface hazard curve slopes at these frequencies are comparable and are significantly different from the slope of the PGA hazard curve. Therefore, using the PGA hazard curve for risk quantification could lead to unacceptable bias.

5.4 SCDF Results

The seismic PRA performed for CGS shows that the point estimate mean seismic CDF is 2.0×10^{-5}/yr. 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 to SCDF risk are discussed in the following paragraphs.

Important Seismic Initiating Event Contributors

Figure 5.4-1 summarizes the SCDF contributors by seismic initiating event in graphical pie chart form, i.e., the SCDF contributors by initiating event.

As can be seen from the graphical display, the seismic initiators %G07 through %G22 are the dominant seismic risk contributors and contribute greater than 83% of the total CDF. The initiating event contribution to CDF comes mostly from intermediate bins. Lower ground motion bins, especially Bins 1-6, have higher frequency, but low conditional core damage probabilities (CCDPs). Higher ground motion bins, especially above Bin 28, have a high CCDP, but lower frequency.

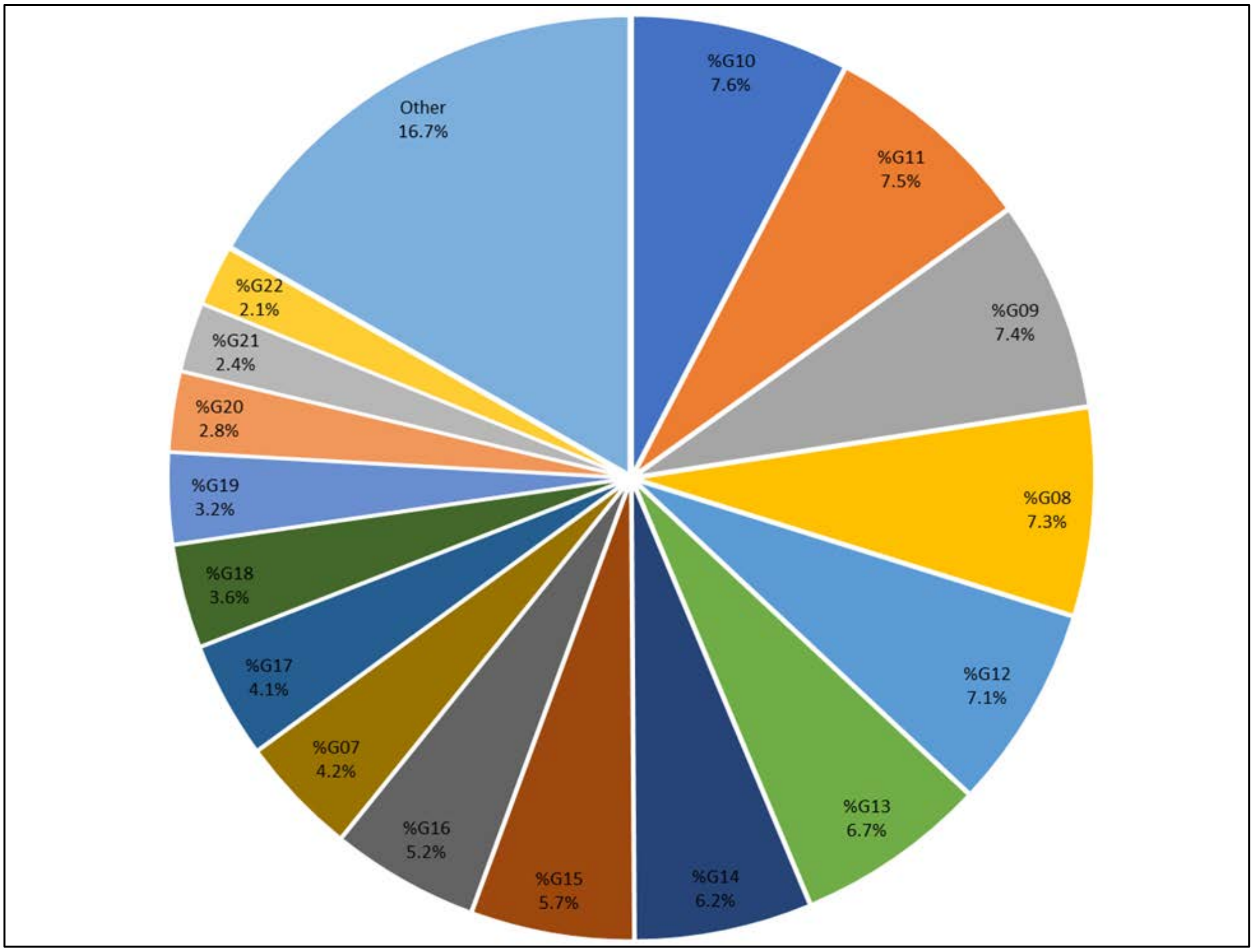

Figure 5.4-1: CGS SPRA SCDF by Initiating Event

SCDF Accident Sequence Contributors

The top SCDF accident sequences are documented in the SPRA quantification report [57]. These are briefly summarized in Table 5.4-1 and Figure 5.4-2.

Among the top SCDF contributors are: structural failure of the RWCB, RB or TB structures (sequence DAMAGE-H_E), loss of control room functionality with loss of HPCS and failure to implement remote shutdown (sequence CR-IC029), and ELAP with failure of the RCIC system (sequence ELAP118).

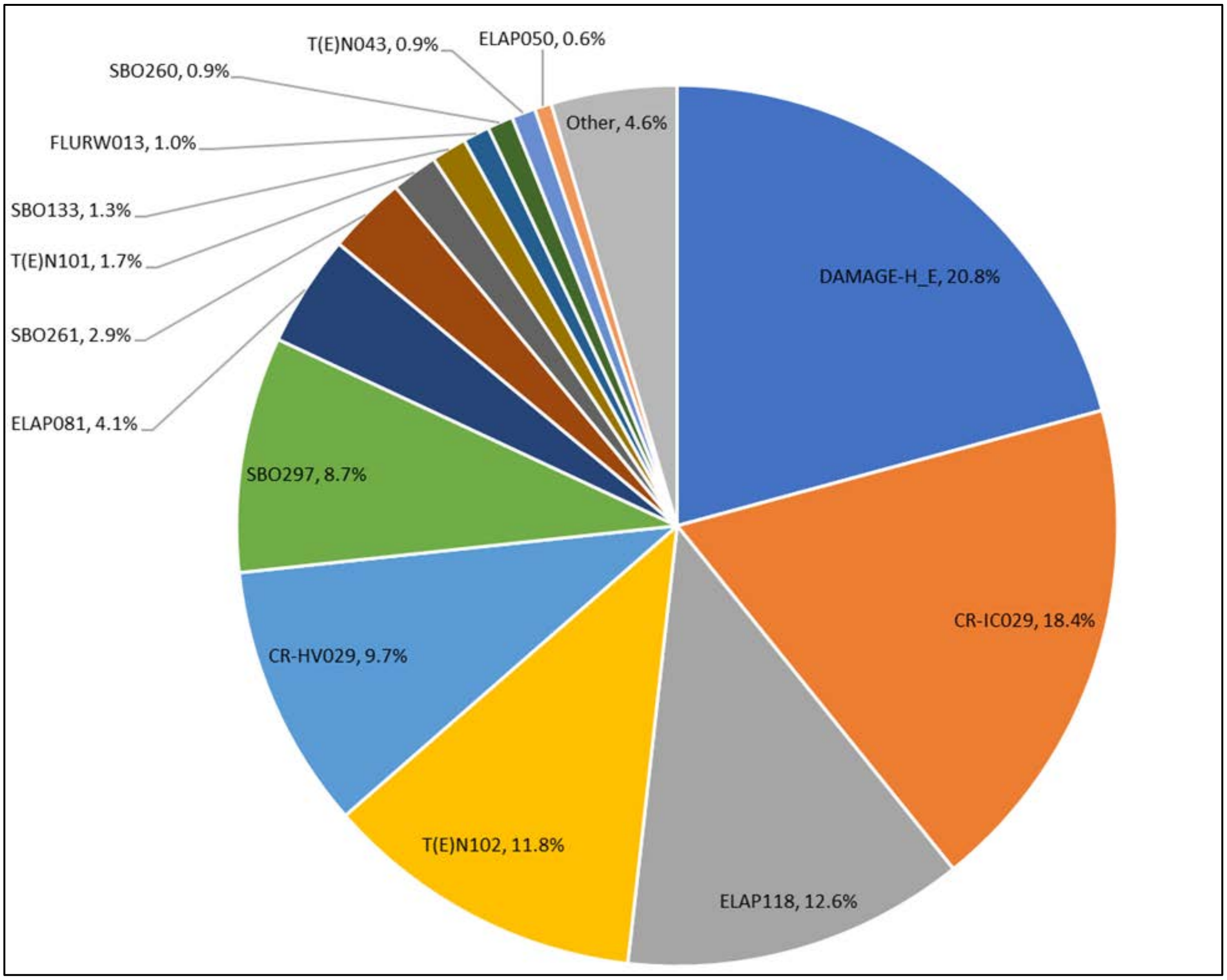

Figure 5.4-2: Accident Sequence Contribution to CDF

Table 5.4-1 Summary of Top 95% CDF Level 1 Accident Sequences

Accident Sequence	Description
DAMAGE-H_E	In this sequence, the structural failure of the RWCB, RB, or TB structures lead to both core damage and large early release. This seismic-only accident sequence (not in FPIE PRA) is described with greater detail in the SPRA plant response model notebook [52].
CR-IC029	This sequence begins with a seismic-induced loss of control room cabinets and/or essential instrumentation. HPCS fails to run and shutdown from either the remote shutdown panel (RSP) or alternate remote shutdown panel (ARS) fails. The failure of injection leads to core damage.
ELAP118	An ELAP occurs and RCIC fails to start or run in the short term, which doesn't allow time to implement FLEX strategies. Because offsite power cannot be restored, core damage occurs due to loss of injection.

Table 5.4-1 Summary of Top 95% CDF Level 1 Accident Sequences

Accident Sequence	Description
T(E)N102	This sequence is a seismic-induced LOSP. HPCS and RCIC both fail to provide high pressure makeup and RPV depressurization fails, resulting in core damage at high pressure.
CR-HV029	This sequence is like CR-IC029, except the control room fails due to HVAC loss that is not recovered.. As in CR-IC029, HPCS operation fails and shutdown from the RSP/ARS also fails.
SBO297	In this seismic-induced SBO scenario, neither HPCS nor RCIC are available to provide high pressure makeup in the short term. Without prompt offsite power recovery, core damage ensues without adequate makeup.
ELAP081	This sequence is a seismic-induced ELAP. FLEX strategies fail to be implemented, recovery of offsite power fails, and local manual operation of RCIC also fails. Without injection, core damage occurs.
SBO261	This sequence begins with a seismic-induced station blackout (both offsite power failure and failure of EDGs 1 and 2). HPCS fails in the short term, but RCIC succeeds to provide makeup in the short term. FLEX strategies fail to be implemented. Local manual operation of RCIC fails, leading to the eventual failure of RCIC, and core damage occurs.
T(E)N101	This sequence is a seismic-induced LOSP. HPCS and RCIC both fail to provide high pressure makeup. The RPV is successfully depressurized in response, however low pressure makeup sources also fail, resulting in core damage at low pressure.
SBO133	This seismic-induced SBO is characterized by early success of HPCS, but eventual failure of EDG 3 to support it in the long term. FLEX strategies fail to be implemented. Local manual operation of RCIC fails, leading to the eventual failure of RCIC and core damage occurs.
FLURW013	This sequence is a seismic-induced moderate or major break outside containment in the RWCU piping. Isolation of RWCU occurs. HPCS fails, requiring depressurization (which succeeds). Low-pressure injection sources, however, fail to provide adequate makeup.
SBO260	This sequence is like SBO261, except local operation of RCIC succeeds. However, when containment venting fails, there is no decay heat removal from containment and RCIC will eventually mechanically fail. Without any remaining makeup, core damage occurs.
T(E)N043	This sequence is a LOSP. HPCS fails to inject, but RCIC succeeds. Initiation of suppression pool cooling (SPC) fails, leading to eventual necessity to depressurize. The depressurization and low-pressure injection functions both succeed, however all decay heat removal methods fail, including venting. With no way to remove decay heat, containment fails, and injection subsequently fails.
ELAP050	This sequence is like ELAP081. In this sequence, however, a FLEX DG successfully powers the Division 1 battery chargers. RCIC fails to provide long-term injection. FLEX external injection fails to be implemented and core damage occurs due to loss of makeup.

Important Contributors to Core Damage Frequency

Importance measures for basic events are calculated with ACUBE. Because CDF importance measures are calculated on a per-seismic bin basis, the importance measures for the overall model are calculated using a weighting for the overall bin top event frequency as follows:

$$I_T = \sum_{b=2}^{46} I_{T,b} \frac{f_{T,b}}{f_T}$$

where I_T is the importance measure (RAW or F-V) for top event T (CDF or LERF), $I_{T,b}$ is the importance measure for seismic bin b, $f_{T,b}$ is the top event frequency for each seismic bin, and f_T is the overall top event frequency.

It is noted here that F-V and Criticality Importance (CI) are used interchangeably in this report and are numerically similar. ACUBE calculates CI, not F-V; F-V is used in this report because it is the more familiar metric among industry PRAs.

Table 5.4-2 provides the SCDF F-V importance measures for SSC fragilities.

Major contributors to CDF are related to fragilities for loss of offsite power, ac / dc power distribution, and the RB, RWCB and TB composite fragility, which proceeds directly to core damage [52].

The top five contributors to SCDF with F-V greater than 5.0E-3 are as follows:

Loss of offsite power (F-V = 0.50)

Offsite power is expected to have a high F-V because there is a high probability for the seismic event to fail offsite power.

Chatter Group 4 – Chatter of Electrical Contact Devices in MCC E-MC-4 (F-V = 0.24)

This contact chatter fragility event affects HPCS, which is a makeup system available at high and low-pressure and has a dedicated emergency diesel generator. HPCS is an important source of core cooling. The chatter group 4 fragility is influenced by the interactions of E-MC-4 with an adjacent instrument rack [37].

CGS Structures: Radwaste, Turbine and Reactor Buildings (F-V = 0.15)

The composite fragility for the RWCB, RB and TB structures is modeled to lead directly to core damage and large early release due to widespread damage to electrical distribution and/or piping and a loss of containment integrity [52].

RWCB Elevation 467 Motor Control Centers (F-V = 9.5E-2)

This fragility event produces a loss of safety-related ac and dc power supplies [57].

Motor Control Centers E-MC-7F and E-MC-8F (F-V = 8.4E-2)

This fragility event contributes to potential seismic-induced fire as well as the loss of power to the HVAC systems for the main control room, switchgear rooms and remote shutdown panel room [52].

Table 5.4-2 Events Risk Significant to CDF based on F-V > 0.005

Basic Event[1]	Description and Failure Mode	F-V	Median Capacity (g) [37]	βr	βu	Failure Mode[2]	Fragility Method
S_SEIS-SWY-LOSP	Loss of offsite power	5.0E-1	0.53	0.35	0.45	LOSP	Representative
S_CHTR-GR-4	Chatter Group 4 - E-MC-4 interaction	2.4E-1	2.85	0.20	0.81	CH	SOV
S_RW_TB_RB	CGS Structures: Radwaste, Turbine and Reactor Buildings	1.5E-1	4.3	0.16	0.79	N/A	SOV[3]
S_E-MC-7A_8A_S11D	RWCB 467 Elevation Motor Control Centers	9.5E-2	1.53	0.28	0.48	FIRE	SOV
S_E-MC-7F_8F	Motor Control Centers 7F and 8F	8.4E-2	1.00	0.29	0.87	FF	SOV
S_STR-BLDG-SB	Service Building Failure	4.1E-2	1.27	0.2	0.27	FF	SOV
S_HPCS-DUCT-DG3	HVAC Ducts in DG-3 Room	3.6E-2	6.38	0.24	0.92	FF	Distribution[4]
S_CHTR-GR-5A	Chatter Group 5 - E-SL-73 Interactions	2.9E-2	1.76	0.26	0.27	CH	SOV
S_CHTR-GR-1	Chatter Group 1 - Building-to-building Impact	2.3E-2	2.12	0.16	0.31	CH	SOV
S_E-MC-7AA_8AA	E-MC-7AA and E-MC-8AA	2.2E-2	1.87	0.22	0.45	FF	SOV
S_MCR-CAB-GR-B	MCR Cabinet Correlation Group B	1.9E-2	7.76	0.24	0.79	CR-IC	Hybrid
S_MCR-CAB-GR-A3	MCR Cabinet Correlation Group A3	1.9E-2	7.76	0.24	0.79	CR-IC	Hybrid
S_MCR-CAB-GR-A2	MCR Cabinet Correlation Group A2	1.9E-2	7.76	0.24	0.79	CR-IC	Hybrid
S_MCR-CAB-GR-A1	MCR Cabinet Correlation Group A1	1.9E-2	6.96	0.24	0.74	CR-IC	Hybrid
S_DLO-PS-26	HPCS ENGINE DG-ENG-1C HI CRANKCASE PRESS ALARM & SHUTDOWN (1"H2O) – Contact Chatter	1.8E-2	1.16	0.17	0.31	CH	SOV
S_CHTR-GR-29	Chatter Group 29 - Switchgear Lockout	1.7E-2	2.74	0.33	0.49	CH	SOV
S_WMA-AH-53A_B	CRITICAL SWGR ROOMS AIR HANDLING UNITS	1.6E-2	3.56	0.26	0.78	FF	SOV
S_RW-DUCT-525	HVAC ducting in the RWCB BLDG on elevation 525 & 527+	1.5E-2	3.83	0.24	0.93	FIRE	Distribution
S_CHTR-GR-11	Chatter Group - RCIC Auto Isolation	1.3E-2	2.24	0.20	0.34	CH	SOV
S_DMA-AH-1	DG ROOM STANDBY AIR HANDLING UNITS	9.7E-3	5.38	0.30	0.52	FF	Hybrid
S_E-GEN-DG4	ALTERNATE SOURCE 480 VAC DIESEL GENERATOR SET (DG 4)	9.4E-3	0.53	0.35	0.45	FF	Representative
S_RWCU-HX	Loss of RWCU pressure boundary	8.0E-3	0.62	0.24	0.66	FLOOD, HELB, BOC	QID[5]
S_RW-FIRE-437	E-MC-6C Damaged on RWCB 437 - Fire Potential	7.0E-3	1.48	0.28	0.36	FIRE	SOV
S_E-IN-3A_B	DIV 1 AND 2 CRITICAL POWER SUPPLY INVERTERS	7.0E-3	3.12	0.24	0.54	FF	QID

Table 5.4-2 Events Risk Significant to CDF based on F-V > 0.005

Basic Event[1]	Description and Failure Mode	F-V	Median Capacity (g) [37]	βr	βu	Failure Mode[2]	Fragility Method
SEIS-F-CNDP-MC-6C	CONDITIONAL PROBABILITY OF SEISMIC E-MC-6C FIRE	7.0E-3	N/A	N/A	N/A	N/A	N/A
S_RW-DUCT-501	HVAC ducting in the RWCB control room area on elevation 501	6.1E-3	4.34	0.24	0.94	FF	Distribution
S_CHTR-GR-5	Chatter Group 5 - Interactions	6.1E-3	0.53	0.35	0.45	CH	SOV

[1] The "S_" prefix denotes seismic fragility group basic events.

[2] LOSP = Loss of Offsite Power, CH = Chatter, FIRE = Seismic-Induced Fire, FF = Functional Failure, CR-IC = Loss of Control Room Instrumentation/Control, FLOOD = Seismic-Induced Flood, BOC = Seismic-Induced Break Outside Containment

[3] The building composite fragility is derived from the individual building fragilities as explained in Section 10 of R-04 [37].

[4] Distribution = Representative fragility for distribution systems

[5] QID = Fragility based on plant seismic qualification

Table 5.4-3 provides the SCDF F-V importance measures for risk-significant operator actions.

Table 5.4-3 CGS SCDF Fussell-Vesely Importance Measures for Operator Actions

Human Failure Event	Operator Action Description	F-V TOTAL
COMBINATION_662SE	Dependent HFE: SEIHUMN-HPCS-NR (failure to recover HPCS given pump suction isolation), CR-HUMN-CR-HVAC (failure to align alternate control room HVAC), ADSHUMNSTARTH3LT (failure to depressurize the RPV)	5.1E-2
COMBINATION_5SE	Dependent HFE: CR-HUMN-CR-HVAC, SEIHUMN-ALT_IC (failure to align alternate RPV level indication), OP-HUMN-RSP (failure to shut down using remote shutdown panel)	2.5E-2
SEIHUMN-HPCS-NR_BIN2	Failure to recover HPCS given pump suction isolation by realigning suction path or stopping pump	1.3E-2
RCIHUMN-LOCALNP_BIN2	Failure to locally operate RCIC without dc or ac power	7.3E-3
COMBINATION_467SE	Dependent HFE: SEIHUMN-HPCS-NR, CR-HUMN-CR-HVAC	6.6E-3
COMBINATION_19SE	Dependent HFE: SEIHUMN-HPCS-NR, RHRHUMNSP-COOLLL (failure to align suppression pool cooling)	6.0E-3
COMBINATION_307SE	Dependent HFE: SEIHUMN-HPCS-NR, RCIHUMN-CST-H3LL (failure to align RCIC suction to suppression pool)	5.8E-3
COMBINATION_188SE	Dependent HFE: SEIHUMN-HPCS-NR, CR-HUMN-CR-HVAC, OP-HUMN-RSP	5.6E-3
COMBINATION_249SE	Dependent HFE: SEIHUMN-EDG-RECOV-LOC (failure to locally recover DG – contact chatter), CR-HUMN-CR-HVAC, ADSHUMNSTARTH3LT	5.1E-3

The most significant non-seismic SSC failures (e.g., random failures of modeled components during the SPRA mission time) for SCDF are listed in Table 5.4-4.

Table 5.4-4 CGS SCDF Fussell-Vesely Importance Measures for Non-Seismic Random Failures

Component	Description and Failure Mode	F-V
EACENG-EDG3-S424	EMERGENCY DG SYSTEM DOES NOT CONTINUE TO RUN FOR 24H	2.8E-2
RCITDP-36H-1S4LL	RCIC PUMP FAILS TO RUN FOR 6 TO 36 HOURS	9.9E-3
EACENG-EDG3-S4D3-B	EMERGENCY DG-3 DOES NOT CONTINUE TO RUN FOR 4 TO 24 H	9.2E-3
EACEDG-3----T3D3	DG-3 OUT FOR MAINTENANCE	7.9E-3

Top 10 SCDF Cutset Evaluation

A review was performed of the CDF cutsets to ensure they make logical sense (e.g. safety functions and systems are properly failed by the events in a cutset, mutually exclusive events do not appear within a cutset, and other pre- and post-processing rules are implemented correctly).

Table 5.4-5 provides the Top 10 SCDF cutsets for the CGS SPRA model. The cutset result files were merged so that the cutsets from all seismic hazard intervals (i.e., %G01 through %G46) were aggregated and sorted in order of individual frequency.

Following the cutset listing is a description of the top 50 SCDF cutsets. Cutsets were grouped if they differed only in seismic bin or had other small differences.

Table 5.4-5 CGS Top Ten SCDF Cutsets

#	Cutset Frequency	Event Frequency or Prob	Event	Description
1	4.84E-07	2.46E-05	%G07	Seismic Initiating Event (0.8g to <0.9g)
		8.70E-01	INIT-RY-CONVRSN	CONVERTS CRITICAL YEARS TO RX YEARS INIT EVENTS
		2.26E-02	S_RW_TB_RB-C-G07	SEISMIC FRAGILITY FOR %G07: Group S_RW_TB_RB – RWCB, RB, TB Composite Fragility
2	4.77E-07	1.77E-05	%G08	Seismic Initiating Event (0.9g to <1g)
		8.70E-01	INIT-RY-CONVRSN	CONVERTS CRITICAL YEARS TO RX YEARS INIT EVENTS
		3.10E-02	S_RW_TB_RB-C-G08	SEISMIC FRAGILITY FOR %G08: Group S_RW_TB_RB – RWCB, RB, TB Composite Fragility
3	4.54E-07	1.28E-05	%G09	Seismic Initiating Event (1g to <1.1g)
		8.70E-01	INIT-RY-CONVRSN	CONVERTS CRITICAL YEARS TO RX YEARS INIT EVENTS
		4.08E-02	S_RW_TB_RB-C-G09	SEISMIC FRAGILITY FOR %G09: Group S_RW_TB_RB – RWCB, RB, TB Composite Fragility
4	4.26E-07	9.50E-06	%G10	Seismic Initiating Event (1.1g to <1.2g)
		8.70E-01	INIT-RY-CONVRSN	CONVERTS CRITICAL YEARS TO RX YEARS INIT EVENTS
		5.16E-02	S_RW_TB_RB-C-G10	SEISMIC FRAGILITY FOR %G10: Group S_RW_TB_RB – RWCB, RB, TB Composite Fragility
5	3.96E-07	7.16E-06	%G11	Seismic Initiating Event (1.2g to <1.3g)
		8.70E-01	INIT-RY-CONVRSN	CONVERTS CRITICAL YEARS TO RX YEARS INIT EVENTS
		6.35E-02	S_RW_TB_RB-C-G11	SEISMIC FRAGILITY FOR %G11: Group S_RW_TB_RB – RWCB, RB, TB Composite Fragility
6	3.61E-07	5.45E-06	%G12	Seismic Initiating Event (1.3g to <1.4g)
		8.70E-01	INIT-RY-CONVRSN	CONVERTS CRITICAL YEARS TO RX YEARS INIT EVENTS
		7.62E-02	S_RW_TB_RB-C-G12	SEISMIC FRAGILITY FOR %G12: Group S_RW_TB_RB – RWCB, RB, TB Composite Fragility
7	3.28E-07	4.20E-06	%G13	Seismic Initiating Event (1.4g to <1.5g)

Table 5.4-5 CGS Top Ten SCDF Cutsets

#	Cutset Frequency	Event Frequency or Prob	Event	Description
		8.70E-01	INIT-RY-CONVRSN	CONVERTS CRITICAL YEARS TO RX YEARS INIT EVENTS
		8.97E-02	S_RW_TB_RB-C-G13	SEISMIC FRAGILITY FOR %G13: Group S_RW_TB_RB – RWCB, RB, TB Composite Fragility
8	3.11E-07	1.28E-05	%G09	Seismic Initiating Event (1g to <1.1g)
		1.00E+00	ADSHUMNSTARTH3LT_BIN2	ADSHUMNSTARTH3LT HEP FOR DAMAGE BIN 2
		1.00E+00	CR-HUMN-CR-HVAC_BIN2	CR-HUMN-CR-HVAC HEP FOR BIN 2
		8.70E-01	INIT-RY-CONVRSN	CONVERTS CRITICAL YEARS TO RX YEARS INIT EVENTS
		1.15E-01	S_CHTR-GR-4-C-G09	SEISMIC FRAGILITY FOR %G09: Chatter Group 4 - E-MC-4 interaction
		5.21E-01	S_E-MC-7F_8F-C-G09	SEISMIC FRAGILITY FOR %G09: MOTOR CONTROL CENTER 7F and 8F
		8.84E-01	S_SEIS-SWY-LOSP-C-G09	SEISMIC FRAGILITY FOR %G09: Loss of offsite power
		5.25E-01	SDP9_COMBINATION_662SE	Dependent HFE: SEIHUMN-HPCS-NR (failure to recover HPCS given pump suction isolation), CR-HUMN-CR-HVAC (failure to align alternate control room HVAC), ADSHUMNSTARTH3LT (failure to depressurize the RPV)
		1.00E+00	SEIHUMN-HPCS-NR_BIN2	SEIHUMN-HPCS-NR HEP FOR DAMAGE BIN 2
9	3.06E-07	9.50E-06	%G10	Seismic Initiating Event (1.1g to <1.2g)
		1.00E+00	ADSHUMNSTARTH3LT_BIN2	ADSHUMNSTARTH3LT HEP FOR DAMAGE BIN 2
		1.00E+00	CR-HUMN-CR-HVAC_BIN2	CR-HUMN-CR-HVAC HEP FOR BIN 2
		8.70E-01	INIT-RY-CONVRSN	CONVERTS CRITICAL YEARS TO RX YEARS INIT EVENTS
		1.38E-01	S_CHTR-GR-4-C-G10	SEISMIC FRAGILITY FOR %G10: Chatter Group 4 - E-MC-4 interaction
		5.60E-01	S_E-MC-7F_8F-C-G10	SEISMIC FRAGILITY FOR %G10: MOTOR CONTROL CENTER 7F
		9.13E-01	S_SEIS-SWY-LOSP-C-G10	SEISMIC FRAGILITY FOR %G10: Loss of offsite power
		5.25E-01	SDP10_COMBINATION_662SE	Dependent HFE: SEIHUMN-HPCS-NR (failure to recover HPCS given pump suction isolation), CR-HUMN-CR-HVAC (failure to align alternate control room HVAC), ADSHUMNSTARTH3LT (failure to depressurize the RPV)
		1.00E+00	SEIHUMN-HPCS-NR_BIN2	SEIHUMN-HPCS-NR HEP FOR DAMAGE BIN 2

Table 5.4-5 CGS Top Ten SCDF Cutsets

#	Cutset Frequency	Event Frequency or Prob	Event	Description
10	3.06E-07	1.77E-05	%G08	Seismic Initiating Event (0.9g to <1g)
		1.00E+00	ADSHUMNSTARTH3LT_BIN2	ADSHUMNSTARTH3LT HEP FOR DAMAGE BIN 2
		1.00E+00	CR-HUMN-CR-HVAC_BIN2	CR-HUMN-CR-HVAC HEP FOR BIN 2
		8.70E-01	INIT-RY-CONVRSN	CONVERTS CRITICAL YEARS TO RX YEARS INIT EVENTS
		9.37E-02	S_CHTR-GR-4-C-G08	SEISMIC FRAGILITY FOR %G08: Chatter Group 4 - E-MC-4 interaction
		4.77E-01	S_E-MC-7F_8F-C-G08	SEISMIC FRAGILITY FOR %G08: MOTOR CONTROL CENTER 7F
		8.46E-01	S_SEIS-SWY-LOSP-C-G08	SEISMIC FRAGILITY FOR %G08: Loss of offsite power
		5.25E-01	SDP8_COMBINATION_662SE	Dependent HFE: SEIHUMN-HPCS-NR (failure to recover HPCS given pump suction isolation), CR-HUMN-CR-HVAC (failure to align alternate control room HVAC), ADSHUMNSTARTH3LT (failure to depressurize the RPV)
		1.00E+00	SEIHUMN-HPCS-NR_BIN2	SEIHUMN-HPCS-NR HEP FOR DAMAGE BIN 2

Group 1: Cutsets 1 through 7, 11, 14, 16, 18, 20, 22, 24, 25, 27, 28, 36, 38, and 49 are all characterized by the structural failure of the RB or RWCB structures accompanied by the failure of the RWCU heat exchanger, leading to a non-isolable break outside containment. Without the ability to mitigate this event, core damage occurs.

Group 2: Cutsets 8, 9, 10, 12, 13, 15, 17, 19, 21, 23, 26, 29, 37, 39 through 41, 43, 46, and 47 are seismic-induced LOSP events. Seismic chatter causes HPCS to fail to start and is not recovered. Seismic fragility events fail normal MCR HVAC and operators fail to establish alternate cooling. The RSP is also non-functional due to a loss of HVAC. The control failures lead to the inability to operate RCIC. When operators fail to depressurize from the ARS, there is a loss of makeup at high pressure resulting in core damage.

Group 3: Cutsets 30 through 35 are ELAP scenarios. RCIC is also failed in these cutsets. In an ELAP without RCIC, there is no path to success to implement FLEX due to time limitations.

Group 4: Cutsets 42, 44, 45, 48, and 50 involve a loss of MCR indication/instrumentation initiating event. Operators fail to provide alternate indication in the MCR and fail to shut down from outside the MCR.

SCDF Accident Class Contributors

The dominant Level 1 accident class contributors to the CGS SCDF consist of the following:

- Class 2D (Transient with loss of decay heat removal) – 28.2%
- Class 6A1 (Station blackout sequences with early failure of HPCS and RCIC) – 23.0%
- Class 5A (Large or medium LOCA outside containment sequences with failure to isolate the break) – 20.5%
- Class 1A3 (Losses of offsite power, failures of high pressure injection and depressurization) – 11.8%
- Class 6B2 (ELAP sequences with HPCS failure, failure to implement FLEX strategies, and loss of decay heat removal) – 11.7%

The Level 1 accident class definitions for the CGS SPRA are based on those defined for the CGS FPIE PRA model. The Level 1 accident classes are described in Table 4-3 of the CGS SPRA Quantification Notebook [57].

The largest contributor is class 2D, where core damage occurs after loss of containment heat removal. The top cutsets for 2D involve HPCS failure, a loss of control room functionality and failure to safely shutdown from outside the control room.

Two of the top 5 largest class contributors are 6B2 and 6A1. Both involve SBO / ELAP sequences with failures of HPCS. This is an expected result; if offsite power and EDGs 1 and 2 are failed due to the seismic event, it's likely that the HPCS EDG is also failed.

The number three contributor is 5A, a non-isolable break outside containment. This is primarily due to cutsets involving the failure of the RWCB or RB structures, which proceed directly to core damage and large early release.

The fourth largest contributor is 1A3, which is failure of high-pressure injection with failure of depressurization.

Table 5.4-6 presents a summary of the SCDF results for each seismic hazard interval.

Table 5.4-6 Contribution to SCDF by Acceleration Interval

Hazard Interval Description	SCDF	% of Total SCDF	Cumulative CDF
Seismic Initiating Event (0.125g to <0.3g)	7.8E-9	0.0%	0.0%
Seismic Initiating Event (0.3g to <0.4g)	1.4E-8	0.1%	0.1%
Seismic Initiating Event (0.4g to <0.5g)	4.5E-8	0.2%	0.3%
Seismic Initiating Event (0.5g to <0.6g)	1.3E-7	0.7%	1.0%
Seismic Initiating Event (0.6g to <0.7g)	2.0E-7	1.0%	2.0%
Seismic Initiating Event (0.7g to <0.8g)	2.7E-7	1.3%	3.3%
Seismic Initiating Event (0.8g to <0.9g)	8.5E-7	4.2%	7.5%
Seismic Initiating Event (0.9g to <1g)	1.5E-6	7.3%	14.8%
Seismic Initiating Event (1g to <1.1g)	1.5E-6	7.4%	22.2%
Seismic Initiating Event (1.1g to <1.2g)	1.6E-6	7.6%	29.8%
Seismic Initiating Event (1.2g to <1.3g)	1.5E-6	7.5%	37.3%
Seismic Initiating Event (1.3g to <1.4g)	1.5E-6	7.1%	44.5%
Seismic Initiating Event (1.4g to <1.5g)	1.4E-6	6.7%	51.2%
Seismic Initiating Event (1.5g to <1.6g)	1.3E-6	6.2%	57.4%
Seismic Initiating Event (1.6g to <1.7g)	1.2E-6	5.7%	63.1%
Seismic Initiating Event (1.7g to <1.8g)	1.1E-6	5.2%	68.3%
Seismic Initiating Event (1.8g to <1.9g)	8.3E-7	4.1%	72.4%
Seismic Initiating Event (1.9g to <2g)	7.4E-7	3.6%	76.0%
Seismic Initiating Event (2g to <2.1g)	6.5E-7	3.2%	79.2%
Seismic Initiating Event (2.1g to <2.2g)	5.7E-7	2.8%	82.0%
Seismic Initiating Event (2.2g to <2.3g)	4.9E-7	2.4%	84.5%
Seismic Initiating Event (2.3g to <2.4g)	4.3E-7	2.1%	86.6%

Table 5.4-6 Contribution to SCDF by Acceleration Interval

Hazard Interval Description	SCDF	% of Total SCDF	Cumulative CDF
Seismic Initiating Event (2.4g to <2.5g)	3.5E-7	1.7%	88.3%
Seismic Initiating Event (2.5g to <2.6g)	3.0E-7	1.5%	89.8%
Seismic Initiating Event (2.6g to <2.7g)	2.6E-7	1.3%	91.1%
Seismic Initiating Event (2.7g to <2.8g)	2.2E-7	1.1%	92.2%
Seismic Initiating Event (2.8g to <2.9g)	1.9E-7	0.9%	93.1%
Seismic Initiating Event (2.9g to <3g)	1.7E-7	0.8%	94.0%
Seismic Initiating Event (3g to <3.1g)	1.5E-7	0.7%	94.7%
Seismic Initiating Event (3.1g to <3.2g)	1.3E-7	0.6%	95.3%
Seismic Initiating Event (3.2g to <3.3g)	1.1E-7	0.5%	95.9%
Seismic Initiating Event (3.3g to <3.4g)	9.5E-8	0.5%	96.3%
Seismic Initiating Event (3.4g to <3.5g)	8.3E-8	0.4%	96.7%
Seismic Initiating Event (3.5g to <3.6g)	7.3E-8	0.4%	97.1%
Seismic Initiating Event (3.6g to <3.7g)	6.3E-8	0.3%	97.4%
Seismic Initiating Event (3.7g to <3.8g)	5.6E-8	0.3%	97.7%
Seismic Initiating Event (3.8g to <3.9g)	4.8E-8	0.2%	97.9%
Seismic Initiating Event (3.9g to <4g)	4.3E-8	0.2%	98.1%
Seismic Initiating Event (4g to <4.1g)	3.7E-8	0.2%	98.3%
Seismic Initiating Event (4.1g to <4.2g)	3.3E-8	0.2%	98.5%
Seismic Initiating Event (4.2g to <4.3g)	2.9E-8	0.1%	98.6%
Seismic Initiating Event (4.3g to <4.4g)	2.6E-8	0.1%	98.7%
Seismic Initiating Event (4.4g to <4.5g)	2.3E-8	0.1%	98.9%
Seismic Initiating Event (4.5g to <4.6g)	2.1E-8	0.1%	99.0%
Seismic Initiating Event (4.6g to <4.7g)	1.8E-8	0.1%	99.1%
Seismic Initiating Event (>4.7g)	1.9E-7	0.9%	100.0%

5.5 SLERF Results

The seismic PRA performed for CGS shows that the point estimate mean SLERF is 8.8×10^{-6}/yr. A discussion of the mean SLERF with uncertainty distribution reflecting the uncertainties in the hazard, fragilities, and model data is presented in Section 5.6. Important contributors are discussed in the following paragraphs. Figure 5.5-1 summarizes the SLERF contributors by seismic initiating event in graphical pie chart form, i.e., the SLERF contributors by initiating event.

As can be seen from the graphical display, the seismic initiators %G07 through %G26 and %G46 are the dominant seismic risk contributors to SLERF. The contribution to SLERF per seismic bin is like that for SCDF, with most of the contribution coming from intermediate bins.

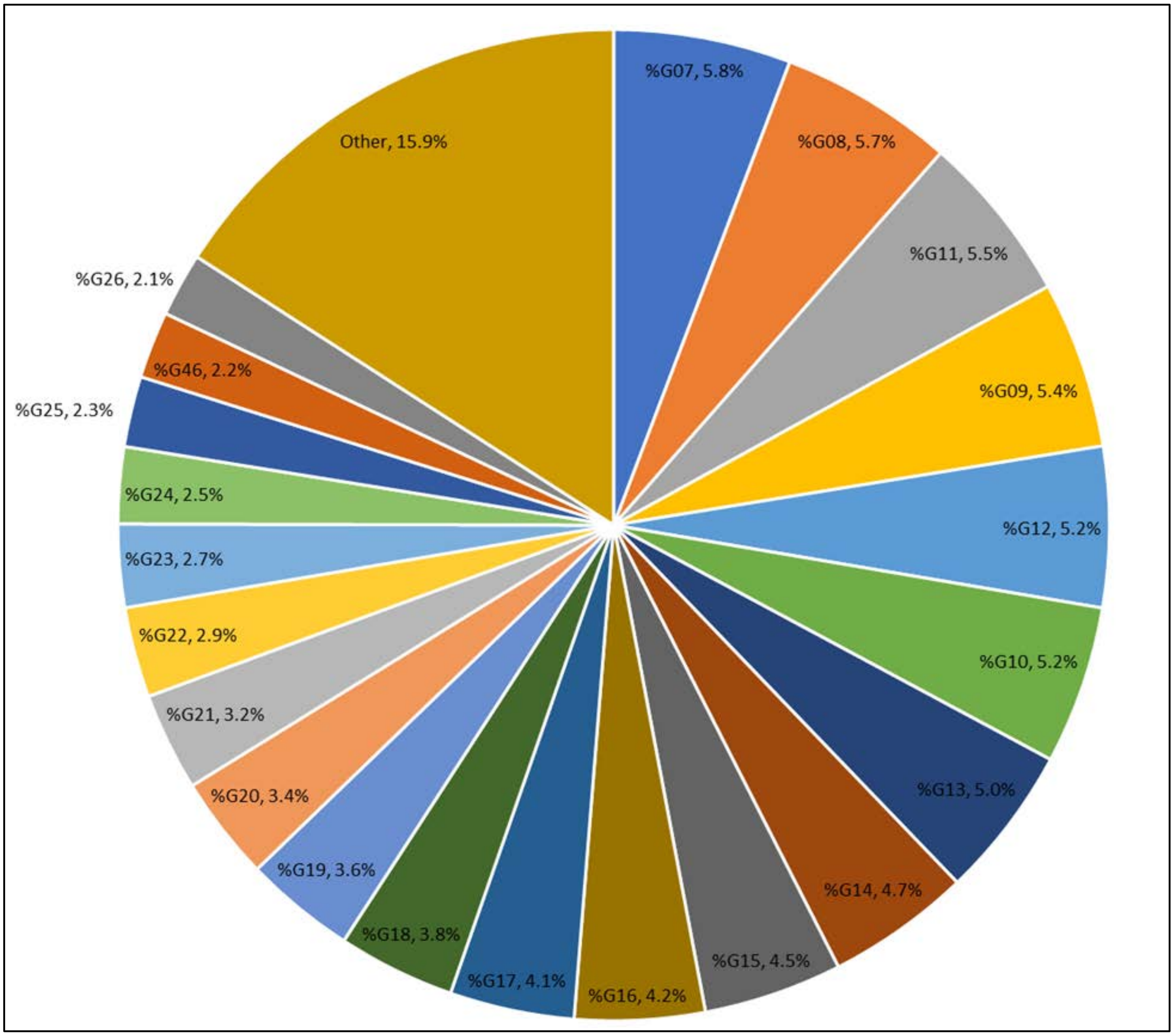

Figure 5.5-1: CGS SPRA SLERF by Initiating Event

SLERF Accident Sequence Contributors

The top SLERF accident sequences are documented in the SPRA quantification report [57]. These are briefly summarized in Table 5.5-1 and Figure 5.5-2.

The top SLERF contributors are: 5A-002 (breaks outside containment), 6A1-079 (SBOs without containment isolation), 6A1-041 (SBOs with low RCS pressure and energetic containment failure), 6A1-078 (like 6A1-041 but at high RCS pressure), and 1A-079 (loss of high pressure injection and depressurization with containment isolation failure).

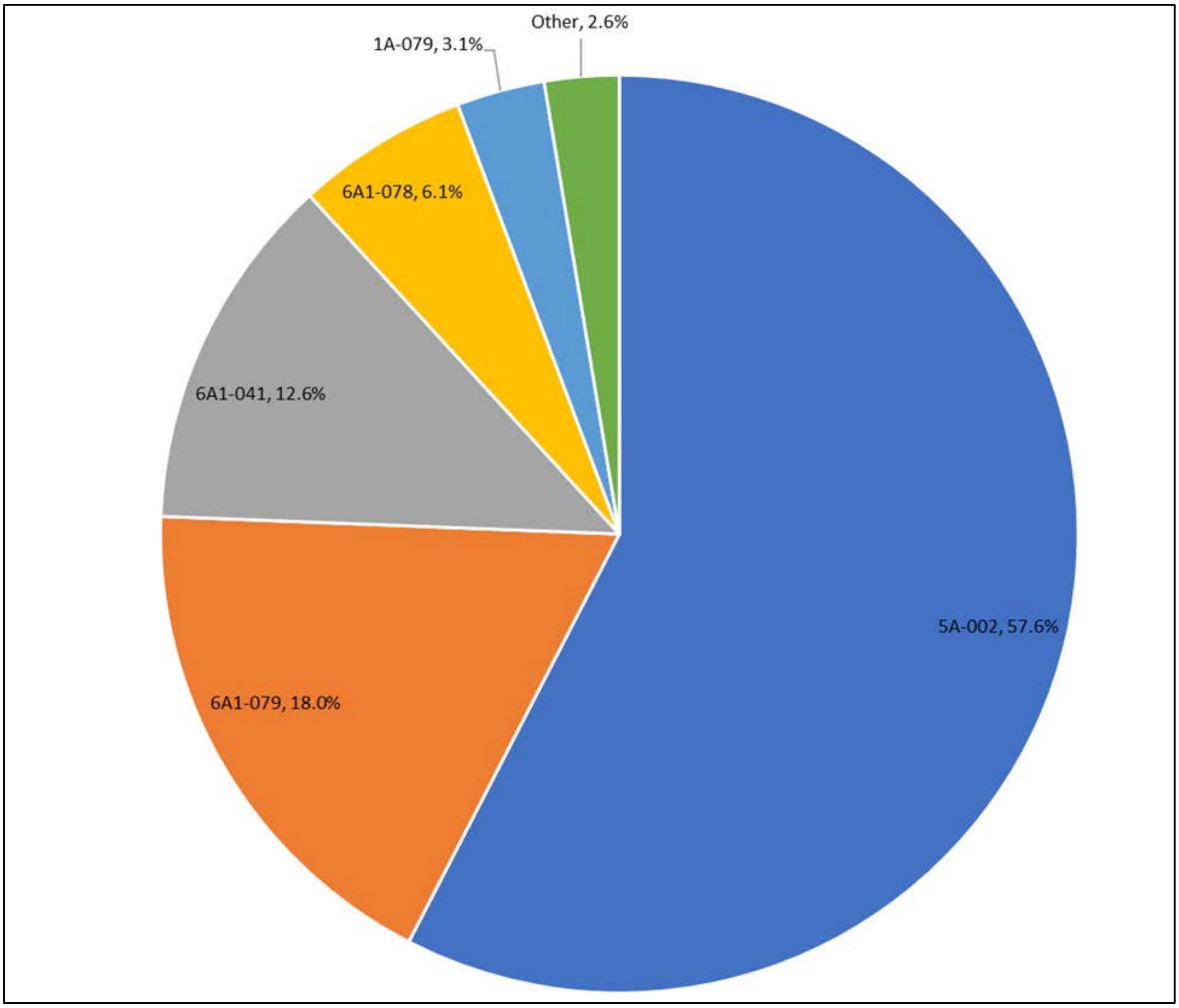

Figure 5.5-2: Accident Sequence Contribution to LERF

Table 5.5-1 Summary of Top 95% Large Early Release Sequences

Accident Sequence	Description
5A-002	Accident sequence 5A-002 reaches core damage by way of containment bypass. The break outside containment provides a pathway for unscrubbed, prompt radionuclide release.
6A1-079	These sequences begin with a 6A1 core damage state (SBO w/ failure of HPCS and RCIC and no depressurization). These sequences result in high, early release because containment penetrations are not isolated during the event.
6A1-041	The core damage sequences are like that of 6A1-079, though operators can depressurize the reactor post-core damage. The core melt progression is not terminated by late injection. Early containment failure results in large early release.

Table 5.5-1 Summary of Top 95% Large Early Release Sequences

Accident Sequence	Description
6A1-078	This sequence is like 6A1-041, but the reactor remains at high pressure during the in-vessel core melt progression.
1A-079	Core damage results from a sequence with failure of high pressure injection and failure of depressurization. Like 6A1-079, containment isolation also fails, resulting in early, unscrubbed release.

Important Contributors to Large Early Release Frequency

Table 5.5-2 provides the SLERF F-V importance measures for SSC fragilities. The SLERF F-V risk importance values are calculated in a manner similar to that discussed in Section 5.4 for SCDF F-V values.

The top 5 contributors to SLERF by F-V are as follows:

CGS Structures: Radwaste, Turbine and Reactor Buildings (F-V = 0.49)

The composite fragility for the RWCB, RB and TB structures is modeled to lead directly to core damage and large early release due to widespread damage to electrical distribution and/or piping and a loss of containment integrity [52].

Loss of offsite power (F-V = 0.26)

Offsite power is expected to have a high F-V because there is a high probability for the seismic event to fail offsite power.

Chatter Group 1 - Building-to-building Impact (F-V = 7.9E-2)

Chatter group 1 consists of ECDs whose fragilities are affected by RB-to-RWCB building-to-building impacts [37]. The impacts from this chatter group to plant operation include the automatic isolation of RCIC as well as the spurious opening of containment isolation valves in the fuel pool cooling system (FPC) [57].

Chatter Group 4 – Chatter of Electrical Contact Devices in MCC E-MC-4 (F-V = 7.1E-2)

This fragility event involves contact chatter that affects HPCS, which is a makeup system available at high and low-pressure and has a dedicated emergency diesel generator. The chatter group 4 fragility is influenced by the interactions of E-MC-4 with an adjacent instrument rack [37].

RWCB Elevation 467 Motor Control Centers (F-V = 5.1E-2)

This fragility event produces a loss of safety-related ac and dc power supplies [57].

Table 5.5-2 Events Risk Significant to LERF based on F-V > 0.005

Basic Event[1]	Description and Failure Mode	F-V	Median Capacity (g) [37]	βr	βu	Failure Mode[2]	Fragility Method
S_RW_TB_RB	CGS Structures: Radwaste, Turbine and Reactor Buildings	4.9E-1	4.3	0.16	0.79	n/a	SOV[3]
S_SEIS-SWY-LOSP	Loss of offsite power	2.6E-1	0.53	0.35	0.45	LOSP	Representative
S_CHTR-GR-1	Chatter Group 1 - Building-to-building Impact	7.9E-2	2.12	0.16	0.31	CH	SOV
S_CHTR-GR-4	Chatter Group 4 - E-MC-4 interaction	7.1E-2	2.85	0.20	0.81	CH	SOV
S_E-MC-7A_8A_S11D	RWCB 467 Elevation Motor Control Centers	5.1E-2	1.53	0.28	0.48	FIRE	SOV
S_RWCU-HX	Loss of RWCU pressure boundary	4.8E-2	0.62	0.24	0.66	FLOOD, HELB, BOC	QID5[5]
S_CHTR-GR-5B	Chatter Group 5B - E-MC-7BA Interactions	4.4E-2	0.66	0.21	0.54	CH	SOV
S_E-MC-7F_8F	Motor Control Centers 7F and 8F	2.5E-2	1.00	0.29	0.87	FF	SOV
S_CHTR-GR-11	Chatter Group - RCIC Auto Isolation	2.4E-2	2.24	0.20	0.34	CH	SOV
S_STR-BLDG-SB	Service Building Failure	2.3E-2	1.27	0.2	0.27	FF	SOV
S_HPCS-DUCT-DG3	HVAC Ducts in DG-3 Room	2.2E-2	6.38	0.24	0.92	FF	Distribution[4]
S_RW-DUCT-525	HVAC ducting in the RWCB BLDG on elevation 525 & 527+	2.1E-2	3.83	0.24	0.93	FIRE	Distribution
S_DLO-PS-26	HPCS ENGINE DG-ENG-1C HI CRANKCASE PRESS ALARM & SHUTDOWN (1"H2O)	2.1E-2	1.16	0.17	0.31	CH	SOV
S_E-MC-7AA_8AA	E-MC-7AA and E-MC-8AA	1.8E-2	1.87	0.22	0.45	FF	SOV
S_WMA-AH-53A_B	CRITICAL SWGR ROOMS AIR HANDLING UNITS	1.5E-2	3.56	0.26	0.78	FF	SOV
SEIS-F-CNDP-MC-8F	CONDITIONAL PROBABILITY OF SEISMIC FOR E-MC-7BB FIRE	1.0E-2	N/A	N/A	N/A	N/A	N/A
SEIS-F-CNDP-MC-7F	CONDITIONAL PROBABILITY OF SEISMIC FOR E-MC-7F FIRE	1.0E-2	N/A	N/A	N/A	N/A	N/A
S_CHTR-GR-13	Chatter Group 13 - HPCS Impacted due to contact chatter; recoverable	8.6E-3	2.39	0.26	0.43	CH	SOV
S_CHTR-GR-29	Chatter Group 29 - Switchgear Lockout	7.3E-3	2.74	0.33	0.49	CH	SOV
S_MCR-CAB-GR-A1	MCR Cabinet Correlation Group A1	7.1E-3	6.96	0.24	0.74	CR-IC	Hybrid
S_CHTR-GR-5A	Chatter Group 5 - E-SL-73 Interactions	7.1E-3	1.76	0.26	0.27	CH	SOV
S_DMA-AH-1	DG ROOM STANDBY AIR HANDLING UNITS	6.7E-3	5.38	0.30	0.52	FF	Hybrid
S_MCR-CAB-GR-B	MCR Cabinet Correlation Group B	6.3E-3	7.76	0.24	0.79	CR-IC	Hybrid
S_MCR-CAB-GR-A3	MCR Cabinet Correlation Group A3	6.3E-3	7.76	0.24	0.79	CR-IC	Hybrid

Table 5.5-2 Events Risk Significant to LERF based on F-V > 0.005

Basic Event[1]	Description and Failure Mode	F-V	Median Capacity (g) [37]	βr	βu	Failure Mode[2]	Fragility Method
S_MCR-CAB-GR-A2	MCR Cabinet Correlation Group A2	6.3E-3	7.76	0.24	0.79	CR-IC	Hybrid
S_MS-RPV-HV	RPV COOLDOWN VENT TO EQUIPMENT DRAIN MOTOR-OPERATED VALVES	5.5E-3	9.50	0.24	0.73	SLOCA	EQ Experience[6]

[1] The "S_" prefix denotes seismic fragility group basic events.

[2] LOSP = Loss of Offsite Power, CH = Chatter, FIRE = Seismic-Induced Fire, FF = Functional Failure, CR-IC = Loss of Control Room Instrumentation/Control, FLOOD = Seismic-Induced Flood, BOC = Seismic-Induced Break Outside Containment

[3] The building composite fragility is derived from the individual building fragilities as explained in Section 10 of R-04 [37].

[4] Distribution = Representative fragility for distribution systems

[5] QID = Fragility based on plant seismic qualification

[6] EQ Experience = Fragility based on earthquake experience

Table 5.5-3 provides the SLERF F-V importance measures for risk-significant operator actions.

Table 5.5-3 CGS SLERF Fussell-Vesely Importance Measures for Operator Actions

Human Failure Event	Description	F-V
SEIHUMN-HPCS-NR_BIN2	Failure to recover HPCS given pump suction isolation by realigning suction path or stopping pump – Plant Damage Bin 2	1.4E-2
SEIHUMN-HPCS-NR_BIN3	Failure to recover HPCS given pump suction isolation by realigning suction path or stopping pump – Plant Damage Bin 3	9.0E-3
COMBINATION_307SE	Dependent HFE: SEIHUMN-HPCS-NR, RCIHUMN-CST-H3LL (failure to align RCIC suction to suppression pool)	5.1E-3

The most significant non-seismic SSC failures (e.g., random failures of modeled components during the SPRA mission time) for SLERF are listed in Table 5.5-4.

Table 5.5-4 CGS SLERF Fussell-Vesely Importance Measures for Non-Seismic Random Failures

Component	Description and Failure Mode	F-V
EACENG-EDG3-S424	EMERGENCY DG SYSTEM DOES NOT CONTINUE TO RUN FOR 24H	7.1E-3

Top 10 SLERF Cutsets Evaluation

Table 5.5-5 provides the Top 10 SLERF cutsets for the CGS SPRA model. Following the cutset listing is a description of the top 50 SLERF cutsets. Cutsets were grouped if they differed only in seismic bin or other had other small differences.

Table 5.5-5 CGS Top 10 Seismic LERF Cutsets

#	Cutset Frequency	Event Frequency or Prob	Event	Description
1	4.84E-07	2.46E-05	%G07	Seismic Initiating Event (0.8g to <0.9g)
		8.70E-01	INIT-RY-CONVRSN	CONVERTS CRITICAL YEARS TO RX YEARS INIT EVENTS
		2.26E-02	S_RW_TB_RB-C-G07	SEISMIC FRAGILITY FOR %G07: Group S_RW_TB_RB – RWCB, RB, TB Composite Fragility
2	4.77E-07	1.77E-05	%G08	Seismic Initiating Event (0.9g to <1g)
		8.70E-01	INIT-RY-CONVRSN	CONVERTS CRITICAL YEARS TO RX YEARS INIT EVENTS
		3.10E-02	S_RW_TB_RB-C-G08	SEISMIC FRAGILITY FOR %G08: Group S_RW_TB_RB – RWCB, RB, TB Composite Fragility
3	4.54E-07	1.28E-05	%G09	Seismic Initiating Event (1g to <1.1g)
		8.70E-01	INIT-RY-CONVRSN	CONVERTS CRITICAL YEARS TO RX YEARS INIT EVENTS
		4.08E-02	S_RW_TB_RB-C-G09	SEISMIC FRAGILITY FOR %G09: Group S_RW_TB_RB – RWCB, RB, TB Composite Fragility
4	4.26E-07	9.50E-06	%G10	Seismic Initiating Event (1.1g to <1.2g)
		8.70E-01	INIT-RY-CONVRSN	CONVERTS CRITICAL YEARS TO RX YEARS INIT EVENTS
		5.16E-02	S_RW_TB_RB-C-G10	SEISMIC FRAGILITY FOR %G10: Group S_RW_TB_RB – RWCB, RB, TB Composite Fragility
5	3.96E-07	7.16E-06	%G11	Seismic Initiating Event (1.2g to <1.3g)
		8.70E-01	INIT-RY-CONVRSN	CONVERTS CRITICAL YEARS TO RX YEARS INIT EVENTS
		6.35E-02	S_RW_TB_RB-C-G11	SEISMIC FRAGILITY FOR %G11: Group S_RW_TB_RB – RWCB, RB, TB Composite Fragility
6	3.61E-07	5.45E-06	%G12	Seismic Initiating Event (1.3g to <1.4g)
		8.70E-01	INIT-RY-CONVRSN	CONVERTS CRITICAL YEARS TO RX YEARS INIT EVENTS
		7.62E-02	S_RW_TB_RB-C-G12	SEISMIC FRAGILITY FOR %G12: Group S_RW_TB_RB – RWCB, RB, TB Composite Fragility
7	3.28E-07	4.20E-06	%G13	Seismic Initiating Event (1.4g to <1.5g)
		8.70E-01	INIT-RY-CONVRSN	CONVERTS CRITICAL YEARS TO RX YEARS INIT EVENTS
		8.97E-02	S_RW_TB_RB-C-G13	SEISMIC FRAGILITY FOR %G13: Group S_RW_TB_RB – RWCB, RB, TB Composite Fragility
8	2.95E-07	3.26E-06	%G14	Seismic Initiating Event (1.5g to <1.6g)

Table 5.5-5 CGS Top 10 Seismic LERF Cutsets

#	Cutset Frequency	Event Frequency or Prob	Event	Description
		8.70E-01	INIT-RY-CONVRSN	CONVERTS CRITICAL YEARS TO RX YEARS INIT EVENTS
		1.04E-01	S_RW_TB_RB-C-G14	SEISMIC FRAGILITY FOR %G14: Group S_RW_TB_RB – RWCB, RB, TB Composite Fragility
9	2.71E-07	2.57E-06	%G15	Seismic Initiating Event (1.6g to <1.7g)
		8.70E-01	INIT-RY-CONVRSN	CONVERTS CRITICAL YEARS TO RX YEARS INIT EVENTS
		1.21E-01	S_RW_TB_RB-C-G15	SEISMIC FRAGILITY FOR %G15: Group S_RW_TB_RB – RWCB, RB, TB Composite Fragility
10	2.47E-07	2.04E-06	%G16	Seismic Initiating Event (1.7g to <1.8g)
		8.70E-01	INIT-RY-CONVRSN	CONVERTS CRITICAL YEARS TO RX YEARS INIT EVENTS
		1.39E-01	S_RW_TB_RB-C-G16	SEISMIC FRAGILITY FOR %G16: Group S_RW_TB_RB – RWCB, RB, TB Composite Fragility

Group 1: Cutsets 1 through 21, 24, 27, 29, 33, and 39 are all similar to the Group 1 SCDF cutsets. These cutsets are characterized by the structural failure of the RB or RWCB structures accompanied by the failure of the RWCU heat exchanger, leading to a non-isolable break outside containment. Without the ability to mitigate this event, core damage occurs, and the guaranteed containment bypass leads to a large early release.

Group 2: Cutsets 22, 25, 31, 35, 37, 40, 43, 47, 49 and 50 reach core damage due to seismic-induced ELAP without RCIC available. Without RCIC available, FLEX strategies are not successful and core damage ensues. FPC containment isolation pathways fail due to contact chatter, which is not recovered. The failure to isolate containment coupled with the core damage event results in a large early release.

Group 3: Cutsets 23, 26, 42, 44, and 45 are similar to Group 2 cutsets. Seismic-induced flooding in the critical switchgear rooms induce an SBO. Contact chatter fails HPCS and this condition is not recovered or the HPCS pump room cooler fails. RCIC fails and FPC containment isolation pathways fail due to contact chatter.

Group 4: Cutsets 28, 30, 32, 34, 36, 38, 41, and 48 are SBO sequences caused by seismic-induced floods. HPCS fails due to chatter, which is unrecovered by the operations staff. RCIC fails due to a separate contact chatter event, and FPC containment isolation pathways fail due to contact chatter.

SLERF Accident Class Contributors

The dominant Level 1 accident class contributors to the CGS SLERF consist of the following:

- Class 5A (Large or medium LOCA outside containment with failure to isolate the break) – 57.59%
- Class 6A1 (Station blackout sequences with early failure of HPCS and RCIC) – 36.68%
- Class 1A (Loss of high pressure injection and RPV depressurization) – 4.60%

The accident class definitions for the CGS SPRA are based on those defined for the CGS FPIE PRA model. The accident classes are described in Table 4-3 of the CGS SPRA Quantification Notebook [57].

A summary of the SLERF results for each seismic hazard interval is presented in Table 5.5-6.

Table 5.5-6 Contribution to SLERF by Acceleration Interval

Hazard Interval Description	SLERF	% of Total SLERF	Cumulative SLERF
Seismic Initiating Event (0.125g to <0.3g)	1.9E-10	0.0%	0.0%
Seismic Initiating Event (0.3g to <0.4g)	3.9E-10	0.0%	0.0%
Seismic Initiating Event (0.4g to <0.5g)	1.2E-9	0.0%	0.0%
Seismic Initiating Event (0.5g to <0.6g)	1.3E-8	0.1%	0.2%
Seismic Initiating Event (0.6g to <0.7g)	2.0E-8	0.2%	0.4%
Seismic Initiating Event (0.7g to <0.8g)	1.8E-8	0.2%	0.6%
Seismic Initiating Event (0.8g to <0.9g)	5.1E-7	5.8%	6.4%
Seismic Initiating Event (0.9g to <1g)	5.0E-7	5.7%	12.1%
Seismic Initiating Event (1g to <1.1g)	4.8E-7	5.4%	17.5%
Seismic Initiating Event (1.1g to <1.2g)	4.5E-7	5.2%	22.7%
Seismic Initiating Event (1.2g to <1.3g)	4.9E-7	5.5%	28.2%
Seismic Initiating Event (1.3g to <1.4g)	4.6E-7	5.2%	33.4%
Seismic Initiating Event (1.4g to <1.5g)	4.4E-7	5.0%	38.4%
Seismic Initiating Event (1.5g to <1.6g)	4.1E-7	4.7%	43.1%
Seismic Initiating Event (1.6g to <1.7g)	4.0E-7	4.5%	47.6%
Seismic Initiating Event (1.7g to <1.8g)	3.7E-7	4.2%	51.9%
Seismic Initiating Event (1.8g to <1.9g)	3.6E-7	4.1%	55.9%
Seismic Initiating Event (1.9g to <2g)	3.4E-7	3.8%	59.8%
Seismic Initiating Event (2g to <2.1g)	3.2E-7	3.6%	63.4%
Seismic Initiating Event (2.1g to <2.2g)	3.0E-7	3.4%	66.8%
Seismic Initiating Event (2.2g to <2.3g)	2.8E-7	3.2%	70.0%
Seismic Initiating Event (2.3g to <2.4g)	2.6E-7	2.9%	72.9%
Seismic Initiating Event (2.4g to <2.5g)	2.4E-7	2.7%	75.6%
Seismic Initiating Event (2.5g to <2.6g)	2.2E-7	2.5%	78.1%
Seismic Initiating Event (2.6g to <2.7g)	2.0E-7	2.3%	80.4%
Seismic Initiating Event (2.7g to <2.8g)	1.8E-7	2.1%	82.5%
Seismic Initiating Event (2.8g to <2.9g)	1.7E-7	1.9%	84.4%
Seismic Initiating Event (2.9g to <3g)	1.5E-7	1.7%	86.1%
Seismic Initiating Event (3g to <3.1g)	1.5E-7	1.7%	87.7%
Seismic Initiating Event (3.1g to <3.2g)	1.3E-7	1.5%	89.2%
Seismic Initiating Event (3.2g to <3.3g)	1.1E-7	1.3%	90.5%
Seismic Initiating Event (3.3g to <3.4g)	9.8E-8	1.1%	91.6%
Seismic Initiating Event (3.4g to <3.5g)	8.1E-8	0.9%	92.5%
Seismic Initiating Event (3.5g to <3.6g)	7.1E-8	0.8%	93.3%
Seismic Initiating Event (3.6g to <3.7g)	6.1E-8	0.7%	94.0%
Seismic Initiating Event (3.7g to <3.8g)	5.4E-8	0.6%	94.7%
Seismic Initiating Event (3.8g to <3.9g)	4.8E-8	0.5%	95.2%

Table 5.5-6 Contribution to SLERF by Acceleration Interval

Hazard Interval Description	SLERF	% of Total SLERF	Cumulative SLERF
Seismic Initiating Event (3.9g to <4g)	4.2E-8	0.5%	95.7%
Seismic Initiating Event (4g to <4.1g)	3.7E-8	0.4%	96.1%
Seismic Initiating Event (4.1g to <4.2g)	3.3E-8	0.4%	96.5%
Seismic Initiating Event (4.2g to <4.3g)	2.9E-8	0.3%	96.8%
Seismic Initiating Event (4.3g to <4.4g)	2.6E-8	0.3%	97.1%
Seismic Initiating Event (4.4g to <4.5g)	2.3E-8	0.3%	97.4%
Seismic Initiating Event (4.5g to <4.6g)	2.1E-8	0.2%	97.6%
Seismic Initiating Event (4.6g to <4.7g)	1.8E-8	0.2%	97.8%
Seismic Initiating Event (>4.7g)	1.9E-7	2.2%	100.0%

5.6 SPRA Quantification Uncertainty Analysis

Parametric Uncertainty

Parametric uncertainty evaluations for the CGS SPRA were developed using UNCERT 4.0 [62]. All risk-significant basic events are required to be subject to the SOKC for the parametric uncertainty evaluation. The importance tables in the preceding sections were reviewed to ensure their probabilities are sampled in accordance with the SOKC for this evaluation.

Due to software and hardware limitations, the number of cutsets passed to ACUBE for the parametric uncertainty evaluation is limited and, therefore, the mean estimates of the top event frequencies can be much higher than evaluated in the base case due to cutsets not being integrated into the BDD solution. To facilitate discussion of the percentile values in the estimate of top event frequencies, the CGS SPRA CDF and LERF are assumed to have the same mean values as those reported in Sections 5.4 and 5.5 and error factors as those estimated by UNCERT 4.0. The error factor is estimated using the following definition:

$$EF = \frac{x_{95}}{x_{50}}$$

where EF is the error factor, E_{95} is the 95th percentile of the top event frequency, and E_{50} is the median of the top event frequency.

To obtain the 95th percentile of the top event frequency using the calculated mean (from Section 5.4 or 5.5) and error factor, the following equation is used:

$$x_{95} = \bar{x} EF^{\frac{-\ln EF}{5.41}+1}$$

where $\bar{x}$ is the mean top event frequency.

SCDF Uncertainty

Parametric sampling of the CGS SPRA SCDF was performed on the base SCDF cutset file using the UNCERT 4.0 Monte Carlo sampling option, ACUBE BDD value of 2,700 cutsets, and 20,000 samples. For the CGS SPRA SCDF, the sampled 95th percentile was 1.2E-4/yr and sampled median was 3.0E-5/yr, yielding an error factor of approximately 3.8. This error factor is then used to develop a more realistic 95th percentile value based on the base case mean (2.0E-5/yr). This calculated 95th percentile value is 5.6E-5/yr.

This uncertainty error factor on SCDF is reasonable.

SLERF Uncertainty

Parametric sampling of the CGS SPRA SLERF was performed on the base SLERF cutset file using the UNCERT 4.0 Monte Carlo sampling option, ACUBE BDD value of 5000 cutsets, and 20,000 samples. The sampled 95th percentile value of SLERF was 4.4E-5/yr and sampled median was 1.1E-5/yr, yielding an error factor of approximately 3.9. Like SCDF, this error factor is used with the reported mean SLERF of 8.8E-6 to estimate a 95th percentile value of SLERF of 2.4E-5/yr.

This uncertainty error factor on SLERF is reasonable.

Modeling and Completeness Uncertainty

Generic and plant-specific sources of modeling uncertainty (SOU) were rigorously identified as well as sources of completeness uncertainty [63].

The potential sources of generic uncertainty were drawn from EPRI TR-1026511 [63]. For each of the identified generic sources of modeling uncertainty, either no reasonable alternative approach was identified, the uncertainty was addressed specifically by the model itself (e.g. treatment of high failure probabilities was addressed by using ACUBE), or the SOU was determined to be non-risk significant. No key SOUs were identified from the generic modeling uncertainty list.

The potential sources of plant-specific modeling uncertainty were generated for areas where a reasonable alternative exists to an approach taken by the PRA development. The following sources of plant-specific modeling uncertainty were found to be key sources for most applications per NUREG 1855 [64]:

1. The effectiveness of secondary containment (reactor building) to mitigate releases from primary containment (drywell / wetwell) is not credited by the SPRA model, consistent with the FPIE PRA model.
2. Building-to-building impact has an uncertain effect on chatter events. The fragilities of chatter-sensitive contact devices considered impact between two building floors to result in simultaneous chatter of all the components located on those floors.
3. The CGS SPRA quantification employs fragilities truncation. Fragilities are truncated below ground motions that correspond to at least 95% confidence in at most 1% probability of failure.

4. The CGS offsite power loss fragility estimate is typical of that employed by industry SPRAs. The offsite power loss fragility estimate is a source of modeling uncertainty. A consensus reasonable alternative approach is not currently available.
5. Failure of the diesel generator room mixed air HVAC ducts are assumed to fail the respective diesel generator. DG room heatup calculations have not examined the condition in which the HVAC ducts have been damaged. In the absence of dedicated calculations, this approach represents a conservative bias in most applications [57].
6. A reference ground motion of 2.5 Hz is selected for the CGS SPRA as most characteristic for the site [37]. This parameter is representative of the ground motion significant to the response of the major CGS structures, all of which have median fundamental SSI frequencies between 1.6 Hz and 3 Hz.

There are no significant sources of completeness uncertainty relative to this application.

5.7 SPRA Quantification Sensitivity Analysis

Quantification truncation sensitivity analyses were performed by projecting the SCDF and SLERF for the next factor of 10 lower truncation limit. This was performed to demonstrate adequate convergence of the results. The result of the analyses demonstrated that the model results exhibit adequate convergence.

Sensitivity analyses were performed for the assumptions or model uncertainties identified as important to the SPRA results in Section 5.6. Table 5.7-1 provides a summary of six sensitivity cases performed and describes the impacts on the results (sensitivity cases 1 through 6).

The fragility report [37], Section 12.9, identifies candidates for potential refinements of the calculated fragilities. Sensitivity case 7 in Table 5.7-1 examines this potential SPRA model enhancement.

Sensitivity of the SPRA results to SPRA modeling decisions are examined in Table 5.7-1 by sensitivity cases 8 and 9.

Aspects of the SPRA model that are important to the results were considered. A reasonable alternative approach exists for one aspect of the SPRA model, and sensitivity analysis case 10 was documented in Table 5.7-1.

Table 5.7-1 Summary of CGS SPRA Sensitivity Cases

ID	Topic	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made	Impact on Model	Characterization Assessment
1	Secondary containment effectiveness	The containment event trees (CETs) include a node, RB, for the effectiveness of secondary containment (reactor building) to mitigate releases from primary containment (drywell / wetwell). The FPIE PRA sets the failure probability for the RB node to 1.0 in the CETs and therefore assigns no credit to secondary containment effectiveness. Structural failure of the RWCB building proceeds to	SLERF	The FPIE modeling of the RB node is carried over to the SPRA, and RWCB structural leads directly to large early release. The SPRA therefore assigns no credit to secondary containment effectiveness.	In the absence of analysis that would be needed to model secondary containment effectiveness, the SPRA utilizes the modeling established by the FPIE PRA.	A sensitivity evaluation was performed in which secondary containment is credited to mitigate releases from primary containment as well as for structure failure of the RWCB building.	CGS SPRA LERF for this sensitivity is 7.7E-6/yr, a reduction of 13%. This represents a potential risk-significant enhancement of the SPRA pending any future requisite technical analyses for secondary containment effectiveness.

Table 5.7-1 Summary of CGS SPRA Sensitivity Cases

ID	Topic	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made	Impact on Model	Characterization Assessment
		large early release.					
2	Contact Chatter Due to Building-to-Building Impact	Building-to-building contact has an uncertain effect on chatter events	Chatter events	The fragilities of chatter-sensitive contact devices considered impact between two building floors to result in simultaneous chatter of all the components located on those floors.	The method chosen yields realistic results.	To measure the sensitivity of this approach, the fragilities for Contact Chatter Group 1 were modeled for shaking only. This produced four contact chatter groups 1A, 1C, 1D, and 1E. The LERF SPRA model was requantified.	The LERF for this case is 8.5E-6/rx-yr, a relatively small reduction of 3% from the base case. This result indicates a small sensitivity to building impact governing or not governing the failure mode for chatter.
3	Fragility Truncation	This sensitivity examines fragility truncations using different criteria.	Seismic Failures	Fragility truncation was performed at ground motions representing the 95% confidence level of 1% or less probability of failure.	The method chosen yields realistic results.	The SPRA LERF model was quantified using the following fragility truncations: 1) 99% confidence level of 1% or less probability of failure, A99|1, and 2) 95% confidence level of 5% or less probability of failure (i.e., the HCLPF).	1) The LERF for A99|1 truncation increased 0.7%. 2) The sensitivity for HCLPF truncation produced a 3% decrease in LERF. This analysis demonstrates a small sensitivity to the fragility truncation criteria. The choice to use the 95% confidence level of 1% or less probability of failure for fragility truncation is reasonable.

Table 5.7-1 Summary of CGS SPRA Sensitivity Cases

ID	Topic	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made	Impact on Model	Characterization Assessment
4	LOSP Fragility	A fragility sensitivity for LOSP should be performed.	LOSP	A_m = 0.53 β_R = 0.35 β_C = 0.57	The method chosen yields realistic results.	The SPRA CDF and LERF models were quantified using the following fragility for offsite power loss. As noted in R-04, this sensitivity fragility is based on transforming the EPRI 3002000709 PGA parameters to the Reference Earthquake reference ground motion parameter using the Reference Earthquake horizontal ground motion spectrum shape. This transformation is expected to be nonconservatively biased. Am = 1.24g βR = 0.30 βU = 0.45 HCLPF = 0.36g	The CDF for this sensitivity is 1.81E-5/rx-yr, a reduction of 22% from the base case. The LERF for this case is 8.37E-6/rx-yr, a relatively small reduction of 5% from the base case.
5	DG Room Mixed Air HVAC	DG room heatup calculations have not examined the condition in which the HVAC ducts have been damaged.	Loss of offsite power	DG room HVAC duct failures is mapped to fail the affected DG. The fragility groups are S_DG-DUCT-DG1, S_DG-DUCT-DG2, and S_HPCS-DUCT-DG3.	It is assumed that HVAC duct failure in the diesel-generator rooms causes a loss of HVAC.	Seismic-induced failure of EDG HVAC ducting leads to failure of the respective EDG. In the absence of dedicated calculations, this approach represents a conservative bias in most applications. The FV importance measures of these events are examined to assess significance.	The FVs for EDGs 1 and 2 HVAC for CDF and LERF are as follows: • EDG1 CDF: 1.6E-3 • EDG2 CDF: 3.6E-4 • EDG1 LERF: 2.0E-4 • EDG2 LERF: 4.7E-4 This source of uncertainty relevant to EDGs 1 and 2 should not be considered key for most applications.

Table 5.7-1 Summary of CGS SPRA Sensitivity Cases

ID	Topic	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made	Impact on Model	Characterization Assessment
							For HPCS EDG HVAC, the CDF and LERF FVs are 3.6E-2 and 2.1E-2.
6	Reference Ground Motion Parameter	A fragility sensitivity for reference ground motion should be explored.	Seismic failures	The reference ground motion parameter for seismic fragilities is Sa(5%,2.5Hz).	The Sa(5%,2.5Hz) reference ground motion is characteristic for the CGS site.	This sensitivity examines transformation of seismic fragilities to other ground motion parameters. The SPRA LERF model was quantified for the following ground motion parameters: 1) PGA, and 2) Sa (5%, 1Hz). The quantifications were performed by transforming the FRANX modeling input from the 2.5Hz case to the PGA and 1Hz cases.	1) LERF for the PGA case is 1.3E-5/rx-yr, an increase of 47% from the base case. 2) LERF for the 1Hz case is 8.1E-6/rx-yr, a relatively small decrease of 8% from the base case. The reference ground motion parameter chosen yields realistic results.
7	Fragility Sensitivity	The fragility report [37], Section 12.9, identifies candidates for potential fragilities calculation refinements.	Potential fragilities calculation refinements	Potential fragilities calculation refinements were modeled for this sensitivity for selected risk-significant fragility groups [37]: DG-ENG-4 DLO-PS-26 E-IN-3A and E-IN-3B HPCS-DUCT-DG3 MS-V-1 and MS-V-2 RW-DUCT-501 RW-DUCT-525	None	This sensitivity found decreases of 6% and 8% in CDF and LERF, respectively.	Selected refinements to SSC fragility calculations may impact risk-informed decisions.

Table 5.7-1 Summary of CGS SPRA Sensitivity Cases

ID	Topic	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made	Impact on Model	Characterization Assessment
8	Partial Correlation between Building Structure Fragilities	The evidence of strong partial correlation between the building structure fragilities and the quantitative approach taken to model it in the SPRA is presented in the fragility report [37], Section 10.1.	Seismic Failures	Correlated building structure fragilities and best estimate $D_{RB\|TB}$ of 0.2 are used.	The method chosen yields realistic results.	In order to understand the influence of modeling this correlation on the plant risk, the SPRA LERF model was quantified for the following: 1) Using the individual (uncorrelated) building structure fragilities and the best estimate DRB|TB of 0.2, and 2) Using the individual building structure fragilities and the upper bound DRB|TB of 0.5.	1) The LERF result using the individual (uncorrelated) building structure fragilities and the best estimate DRB|TB of 0.2 is 1.01E-5/rx-yr, a 15% increase. 2) The LERF result using the individual (uncorrelated) building structure fragilities and the best estimate DRB|TB of 0.5 is 1.11E-5/rx-yr, a 26% increase.
9	Area-based fragilities for internal fire sources and HVAC duct	Fragilities for internal fire sources and HVAC duct are developed on an area basis.	Seismic-induced fire scenarios; HVAC duct fragility groups	Area-based SSCs for plant piping, internal fire sources and internal flood sources receive fragilities for the least robust applicable component in the area [37]. For example, the internal fire fragility for an elevation of the RB may be based on the least robust transformer for that elevation.	This approach facilitated the walkdown effort and represents a conservative bias.	A sensitivity evaluation was performed in which median capacities for the following risk-significant area-based fragility groups were improved by 20%: S_RW-FIRE-437, damage to the seismic-induced fire ignition source on RW elevation 437 S_HPCS-DUCT-DG3, damage to DG-3 room mixed air HVAC duct S_RW-DUCT-501, damage to HVAC duct in RWCB elevation 501 S_RW-DUCT-525, damage to HVAC duct in RWCB elevation 525	CDF reduces 3% to 1.9E-5/yr and LERF reduces 1% to 8.7E-6/yr.

Table 5.7-1 Summary of CGS SPRA Sensitivity Cases

ID	Topic	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made	Impact on Model	Characterization Assessment
10	Rugged SSCs	SSCs deemed to be rugged by the fragilities team [52] are not included in the SPRA modeling.	Rugged SSCs	SSCs deemed to be rugged by the fragilities team [52] are not included in the SPRA modeling.	It is assumed that rugged SSCs do not have a risk-significant impact on the plant SPRA results.	The impact to the SPRA model is examined by including in the SPRA model fragilities for all SSCs deemed to be rugged. These rugged SSCs were mapped in FRANX to their applicable impacts to the SPRA model and were assigned the screening level fragility (HCLPF = 2.4g (Am = 6g, Br =0.24, Bu = 0.32g)). When rugged SSCs are included in the SPRA modeling, there is no change to CDF and there is a 0.9% increase in LERF.	Inclusion of rugged SSCs in the SPRA modeling produces a 0.0% and 0.9% change to CDF and LERF, respectively.

5.8 SPRA Logic Model and Quantification Technical Adequacy

The CGS SPRA risk quantification and results interpretation methodologies were subjected to an independent peer review against the full set of requirements in the ASME/ANS PRA Standard [4] related to risk quantification and results interpretation.

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

6.0 Conclusions

A seismic PRA has been performed for CGS in accordance with the guidance in the SPID. The CGS SPRA shows that the seismic CDF is 2.0E-05/yr and the seismic LERF is 8.8E-06/yr. This submittal reflects the current as-built / as-operated plant as of the date of this submittal. Although the seismic CDF and LERF results are acceptable, Columbia continues to pursue closure of the remaining F&O, which deals with potential improvement in characterization of the site hazard.

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8.0 Acronyms

ADS	Automatic Depressurization System
AEF	Annual Exceedance Frequency
ANS	American Nuclear Society
ARS	Alternate Remote Shutdown Panel
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without SCRAM

BDD	Binary Decision Diagram
BOP	Balance of Plant
CCD	Concrete Capacity Design
CCDP	Conditional Core Damage Probability
CDF	Core Damage Frequency
CDFM	Conservative Deterministic Failure Model
CGS	Columbia Generating Station
CI	Criticality Importance
CMS	Conditional Mean Spectra
CRD	Control Rod Drive System
CST	Condensate Storage Tank
DGB	Diesel Generator Building
DG / EDG	Diesel Generator
DOE	Department of Energy
ECCS	Emergency Core Cooling System
ECD	Electrical Contact Device
ELAP	Extended Loss of ac Power
EN	Energy Northwest
EPN	Equipment Part Number
EPRI	Electric Power Research Institute
ESEP	Expedited Seismic Evaluation Program
F&O	Peer Review Fact and Observation
FEM	Finite Element Model
FIRS	Foundation Input Response Spectra
FPC	Fuel Pool Cooling
FPIE	Full Power Internal Events
F-V	Fussell-Vesely
GMRS	Ground Motion Response Spectrum
HCLPF	High Confidence Low Probability of Failure
HCV	Hardened Containment Vent
HEP	Human Error Probability
HF	High Frequency
HFE	Human Failure Event
HPCS	High Pressure Core Spray System
HRA	Human Reliability Analysis
ICC	International Code Council
IPEEE	Individual Plant Examination for External Events
ISRS	In-Structure Response Spectra

LERF	Large Early Release Frequency
LHS	Latin Hypercube Sampling
LMSM	Lumped Mass Stick Model
LOCA	Loss of Coolant Accident
LOSP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
MAFE	Mean Annual Frequencies of Exceedance
MCC	Motor Control Center
MCR	Main Control Room
MEL	Master Equipment List
MOV	Motor-Operated Valve
NEI	Nuclear Energy Institute
NEP	Non-exceedance Probability
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NTTF	Near Term Task Force
P&ID	Piping and Instrumentation Drawing
PGA	Peak Ground Acceleration
PNNL	Pacific Northwest National Laboratory
PPRP	Participatory Peer Review Panel
PRA	Probabilistic Risk Assessment
PRM	Plant Response Model
PSA	Pseudo Spectral Acceleration
PSHA	Probabilistic Seismic Hazard Analysis
RB	Reactor building
RCIC	Reactor Core Isolation Cooling System
RHR	Residual Heat Removal
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RSP	Remote Shutdown Panel
RW / RWCB	Radwaste Building
RWCU	Reactor Water Cleanup
SBO	Station Blackout
SCDF	Seismic Core Damage Frequency
SDV	SCRAM Discharge Volume
SEL	Seismic Equipment List

SFR	Seismic Fragility Analysis
SHA	Seismic Hazard Analysis
SIFF	Seismic-Induced Fire and Flooding
SLC	Standby Liquid Control
SLERF	Seismic Large Early Release Frequency
SMA	Seismic Margin Assessment
SOU	Sources of Modeling Uncertainty
SOV	Separation of Variables
SPID	Screening, Prioritization and Implementation Details
SPC	Suppression Pool Cooling
SPR	Seismic Plant-Response Analysis
SPRA	Seismic Probabilistic Risk Assessment
SQUG	Seismic Qualification Utility Group
SRT	Seismic Review Team
SRV	Main Steam Safety Relief Valve
SSC	Structure, System or Component
SSEL	Safe Shutdown Equipment List
SSHAC	Senior Seismic Hazard Analysis Committee
SSI	Soil Structure Interaction
SSSI	Structure-Soil-Structure Interaction
SW	Standby Service Water
TB	Turbine Building
UHRS	Uniform Hazard Response Spectra
UHS	Ultimate Heat Sink
USI	Unresolved Safety Issue

Appendix A

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

A.1. Overview of Peer Review

The CGS PRA was subjected to an independent peer review against all requirements of the ASME/ANS PRA Standard Code Case 1 [4]. The peer review assessment [6], [65], and subsequent disposition of peer review findings, are summarized here. The scope of the review encompassed all technical elements and supporting requirements (SR) for the SHA (seismic hazard), SFR (seismic fragilities), and SPR (seismic PRA modeling) 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 RG1.200 [66] and the requirements in ASME/ANS PRA Standard Code Case 1 [4], and presents the significant results of the peer review.

The CGS SPRA peer review was conducted during the week of December 10, 2018 at the Energy Northwest offices in Richland, Washington. As part of the peer review, a walk-down of portions of CGS was performed on December 11, 2018 by two members of the SFR peer review team who have the appropriate Seismic Qualification Utility Group (SQUG) training as well as one member from the SPR peer review team.

A.2. Summary of the Peer Review Process

The peer review was performed against the requirements in the ASME/ANS PRA Standard Code Case 1 [4], using the peer review process defined in NEI 12-13 [5]. The review was conducted over a four-day period, with a summary and exit meeting on the morning of the fifth day.

The SPRA peer review process defined in NEI 12-13 [5] involves an examination by each reviewer of their assigned PRA technical elements against the requirements in the Standard to ensure the robustness of the model relative to all of the requirements.

Implementing the review involves a combination of a broad scope examination of the PRA elements within the scope of the review and a deeper examination of portions of the PRA elements based on what is found during the initial review. The supporting requirements

(SRs) provide a structure which, in combination with the peer reviewers' PRA experience, provides the basis for examining the various PRA technical elements. If a reviewer identifies a question or discrepancy, that leads to additional investigation until the issue is resolved or a Fact and Observation (F&O) is written describing the issue and its potential impacts as well as suggesting possible resolution.

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

As part of the review team's assessment of capability categories, F&Os are prepared. There are three types of F&Os defined in NEI 12-13 [5]: Findings, which identify issues that must be addressed in order for an SR (or multiple SRs) to meet Capability Category II; Suggestions, which identify issues that the reviewers have noted as potentially important but not requiring resolution to meet the SRs; and Best Practices, which reflect the reviewers' opinion that a particular aspect of the review exceeds normal industry practice. The focus in this appendix is on Findings and their disposition relative to this submittal.

A.3. Peer Review Team Qualifications

The members of the peer review team consisted of the following:

- Paul Amico, Overall Review Lead – Paul Amico is a Nuclear Engineer with over forty years of experience in the performance and management of domestic and international programs involving risk and reliability technology and its application to the design and operation of nuclear reactor plants, non-reactor facilities, process plants and other technologies. Mr. Amico has been the lead reviewer for several of the industry's recent SPRA peer reviews.
- Annie Kammerer, SHA Review Lead – Dr. Kammerer is an expert in seismic hazard and risk analyses and integrated performance-based, risk-informed engineering, particularly as applied to nuclear and liquefied natural gas (LNG) facilities, with 20 years of experience.
- Jeff Kimball, SHA Review Support – Jeffrey Kimball is a Chief Seismologist with RIZZO International (RIZZO). Mr. Kimball has 38 years of experience with the evaluation and characterization of natural phenomena hazards and the design of critical facilities to resist these hazards.

- Ram Srinivasan, SFR Review Lead – Dr. Srinivasan has over forty-six 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 ISFSI. 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.
- Alejandro Asfura, SFR Review Support – Dr. Asfura is a structural engineer professional with 43 years of domestic and international experience in dynamic analysis. Throughout his professional career Dr. Asfura has been involved in major projects in structural dynamic analysis, soil dynamic analysis, deterministic and probabilistic soil-structure interaction analysis, and seismic risk analysis for the nuclear, oil and gas, insurance, transportation, mining, and processing industries in the USA, Europe, Asia, and Latin America.
- John Richards, SFR Review Support – John Richards is a Technical Executive in the Risk and Safety Management program area of the Nuclear Power Sector. In this position, he is primarily responsible for research activities relating to the development and application of methods and tools for seismic evaluation of nuclear power plant structures, systems and components. Mr. Richards has over 35 years of experience in the nuclear industry.
- Jim Chapman, SPR Review Lead – Mr. Chapman has 45 years of experience in PRA and Safety Analyses, including Emergency Operating Procedures, Severe Accident Management Guidelines, and Simulator and Operator Training.
- Phil Tarpinian, SPR Review Support – Mr. Tarpinian is employed as a corporate risk management (PRA) Engineer at Exelon. He has 36 years of experience in the nuclear field in the areas of construction, design, engineering, modifications, maintenance, operation and Probabilistic Risk Assessment (PRA). Mr. Tarpinian has 18 years of experience in the specific field of PRA.
- Daniel Kearnaghan, SPR Review Support – Mr. Kearnaghan Over 32 years of experience in the nuclear industry with 18 years in commercial nuclear power and 14 years in various Department of Energy projects. Commercial nuclear power experience includes 15 years of experience in all aspects of nuclear plant Probabilistic Risk Assessment (PRA), a PWR SRO Certification Course, and 2 years of nuclear safety analysis.

The peer review team members met the peer reviewer independence criteria in NEI 12-13 [5].

A.4. Summary of the Peer Review Conclusions

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

SHA

- As required by the PRA Standard [4], the seismic source characterization (SSC) model and ground motion characterization (GMC) model elements of the Columbia Generating Station (CGS) PSHA are based on Senior Seismic Hazard Analysis Committee (SSHAC) Study Level (SL) 3 guidance.
- A detailed site response analysis (SRA) was performed to define the PSHA ground motions at the control point elevation [7], and [9]. The level of analysis is consistent with providing input to the seismic PRA for the CGS.
- The Technical Integration teams who developed the CGS PSHA input models considered the full range of earthquake data (geological, seismological, and geophysical) to develop the SSC and GMC models.
- The CGS PSHA for the reference rock horizon was developed using a SSHAC Level 3 process [12]. All credible tectonic sources of potentially damaging earthquakes were considered, evaluated, and integrated into the assessment.
- Uncertainties in the seismic source characterization are addressed in the CGS PSHA and are explicitly included in the CGS PSHA computations.
- The effects of local site response are described in Chapter 9 of the CGS PSHA Report [7] for the SMB Stack, Bechtel Power Calculations [8], [9], and [10], and the Energy Northwest letter G02-15-045 to NRC dated March 12, 2015 [3].
- The CGS PSHA includes the propagation of epistemic uncertainty and aleatory variability in each step of the hazard analysis. Epistemic uncertainties and aleatory variability in the SSC and GMC models have been included in the quantification of the ground motion hazard at the site.
- The CGS PSHA horizontal spectral shape for reference rock site conditions is based on the GMC model which includes an assessment of ground motions at twenty spectral frequencies [7]. The horizontal spectral shape at the control point elevation is derived for the same twenty spectra frequencies [10].
- The vertical spectral shape at the control point elevation is derived using V/H ratios that consider the results of the hazard evaluation [18].
- The SPRA fragilities notebook [37] discusses a number of other hazards identified and evaluated.

SFR

- The probabilistic seismic response analysis for the structures at CGS is based on the mean UHRS for AFE of 10-5 anchored to 0.4288g PGA.
- The CGS SPRA selected the spectral acceleration at 2.5 Hz as the reference ground motion parameter for use in the SPRA quantification.
- Fixed-base 3D finite element models were developed for all the buildings and structures included in the PRA effort using computer code SAP2000.
- SSI analyses were performed for five CGS structures: RB, RWCB, DGB, and the two CSTs.
- No SSCs listed in the final SEL were screened based on capacity.

- The seismic-fragility evaluation incorporates the findings of walkdowns of the plant focusing on the anchorage, structural support, and potential systems interactions.
- The calculation of seismic-fragility parameters, including median capacity and variabilities, was performed for failure modes affecting the functions modeled in the system analysis.
- The capacities for the non-risk significant SSCs were conservatively biased, based on design basis, seismic qualification data, previous seismic evaluations, and earthquake experience data. For most of the non-risk significant SSCs, representative fragilities were used. In a few cases, fragilities for non-risk significant SSCs were calculated based on the SOV or enhanced Hybrid methods.
- SOV calculations were performed to determine fragilities for a large portion of the significant risk contributors with essentially all the remainder having fragilities developed using enhanced Hybrid Method calculations.

SPR

- As required by the PRA Standard [4], the CGS SPRA uses a Seismic Initiating Event Tree (SIET) that follows a reasonable hierarchical process to model the seismic-induced plant damage states and transfer these sequences to the appropriate core damage frequency event trees.
- The identification of initiating events is thorough and used several sources. The SPRA model was determined to reflect the as-built, as-operated plant.
- The SPRA accident sequence and system models integrate the seismic hazard with hazard interval initiators and applies seismic induced fragilities to the logic model using EPRI FRANX software, resulting in the quantification of CDF and LERF cutsets that contain the hazard initiators, seismic-induced failures, and non-seismic failures (random failures, CCF events, maintenance unavailabilities, etc.).
- Seismic-induced flooding and fire were appropriately addressed.
- The SIET sequences incorporate the seismic aspects by either leading directly to core damage or transferring to the appropriate event trees.
- The seismic HRA addressed the impact of seismic events on the HRA. The overall approach to the SHRA follows the guidance in EPRI 3002008093 [58].
- The SPRA uses a sound set of criteria for determining correlation groups.
- The relay chatter analysis is well-performed.
- The Level 2 sequences are treated appropriately and lead directly to LERF given core damage when appropriate.
- The SPRA used the internal events PRA model as the basis for developing the SEL. Also, the existing SEL from the IPEEE and plant master equipment list were used in the development of the SEL. Various SSCs and structures not explicitly modeled in the internal events model were identified based on the review of appropriate plant sources.
- The process of identifying the HFEs to be carried over to the SPRA from the FPIE, FPRA, and IFPRA was sound and well done, as was the identification of new HFEs specific to seismic that were added to the model. The specification of HRA damage bins was

reasonable. Adjustments were made to time parameters for the HFEs that received detailed HRA (the risk-significant HFEs). The time margin estimates were confirmed for the actions that were not significant and used the EPRI screening approach.

- The SPRA accident sequence and system models integrate and quantify the seismic hazard with hazard interval initiators and then apply seismic induced fragilities to the logic structure using EPRI FRANX and ACUBE software.
- A quantitative assessment of the uncertainties in CDF and LERF was performed using the EPRI UNCERT 4.0 [62] software.

The peer review team concluded that the CGS seismic PRA model is of good quality and integrates the seismic hazard, the seismic fragilities, and the systems-analysis aspects appropriately to quantify core damage frequency and large early release frequency.

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

As summarized in Table A-1, all SRs are graded as met at Capability Category II [6], [65]. Table A-2 presents summary of the Finding F&Os that have not been closed through an NRC accepted process, and the disposition for each.

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

SR	Assessed Capability Category	Associated Finding F&Os	Disposition to Achieve Met or Capability Category II
SHA			
[None]	N/A	N/A	N/A
SFR			
[None]	N/A	N/A	N/A
SPR			
[None]	N/A	N/A	N/A

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

The set of supporting requirements from the ASME/ANS PRA Standard [4] that are identified in Tables 6-4 through 6-6 of the SPID [2] define the technical attributes of an SPRA model used to respond to the 50.54(f) letter. The conclusions of the peer review discussed above and summarized in this report demonstrates that the CGS SPRA model meets the expectations for PRA scope and technical adequacy as presented in RG 1.200, Revision 2 [66] as clarified in the SPID [2].

The main body of this SPRA summary report describes the SPRA methodology, including:

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

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

The CGS SPRA reflects the as-built and as-operated plant as of the cutoff date for the SPRA, April 18, 2019 [57]. There are no permanent plant changes that have not been reflected in the SPRA model.

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

The Owners Group performed a full scope peer review of the CGS internal events PRA and internal flooding PRA that forms the basis for the SPRA to determine compliance with ASME PRA Standard, RA-S-2008, including the 2009 Addenda A [67] and RG 1.200 [66] in December 2009. This review documented findings for all supporting requirements (SRs) which failed to meet at least Capability Category II. All of the internal events and internal flooding PRA peer review findings have been closed [68] and all SRs are considered to be met.

The Owners Group performed a peer review of the CGS SPRA in December 2018 [6]. The results of this peer review are discussed above. The peer review team expressed the opinion that the CGS SPRA model is of good quality and integrates the seismic hazard, the seismic fragilities, and the systems-analysis aspects appropriately to quantify core damage frequency and large early release frequency. The general conclusion of the peer review was that the CGS SPRA is judged to be suitable for use for risk-informed applications.

- Table A-1 provides a summary of the disposition of SRs judged by the peer review to be not met, or not meeting Capability Category II.

- Table A-2 (located at the end of this appendix) provides a summary of the disposition of the open SPRA peer review findings.
- Table A-3 provides an assessment of the expected impacts to the results of the CGS SPRA of those SRs and peer review Findings that have not been fully addressed.

Table A-3 Summary of Impact of Not Met SRs and Open Peer Review Findings

SR # or F&O #	Summary of Issue Not Fully Resolved	Impact on SPRA Results
Finding 20-10	As assessed by the F&O closure independent assessment team, this finding was partially resolved by: 1) satisfactorily revising the CGS hazard, and 2) reassessing seismic fragilities based on scaling. The revised hazard and fragilities were convolved as a sensitivity evaluation. This is a source of completeness uncertainty.	The convolution of the revised hazard and fragilities sensitivity evaluation produced 4% and 34% increases in SCDF and SLERF, respectively, however, the risk-informed decisions produced as part of this report are not altered by this outcome. The differences in risk estimates for this sensitivity case are attributable to the revised hazard rather than the updated fragilities [57]. Specifically, the revised fragilities produce net decreases in CDF and LERF when convolved with the original hazard. The overall seismic capacity of the plant is generally unchanged. The SSCs whose risk-significance change significantly from the fragility revisions are as follows: • The SW spray pond structure F-V increased from 1E-3 to 5E-2 for CDF and 1E-3 to 3E-2 for LERF. • RB recirculation air fan coolers 10 and 11 F-V values increased from 4E-3 to about 1.5E-2 for CDF and 4E-3 to 1E-2 for LERF. • The F-V for the CRD hydraulic control units increased from 4.5E-3 to 9E-3 for LERF. This partially resolved F&O is not considered a source of completeness uncertainty for this submittal because the CGS SPRA uses the hazard that was approved by the NRC for this application, as discussed in [17]. The revised hazard was only used in a sensitivity study to determine the impact on risk-informed decisions and identification of dominant risk contributors. As stated, the difference in SCDF and SLERF is directly attributable to the change in the hazard and not to the seismic capacity of the SSCs at the plant.

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

The PRA Standard [4] includes a number of requirements related to identification and evaluation of the impact of sources of uncertainty and related assumptions on the PRA results. NUREG-1855 [64], EPRI 1016737 [69] and EPRI 1026511 [63] provide guidance on assessment of uncertainty for applications of a PRA. As described in NUREG-1855 [64], 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 CGS SPRA model quantification (see Section 5 of this report).
- 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 CGS 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.

A summary of potentially important sources of uncertainty in the CGS SPRA is listed in Table A-4.

Table A-4 Summary of Potentially Important Sources of Uncertainty

PRA Element	Summary of Treatment of Sources of Uncertainty per Peer Review	Potential Impact on SPRA Results
Seismic Hazard	The CGS SPRA peer review team noted that both the aleatory and epistemic uncertainties have been addressed in characterizing the seismic sources. In addition, uncertainties in each step of the hazard analysis were propagated and displayed in the final quantification of hazard estimates for the CGS site.	The seismic hazard reasonably reflects sources of uncertainty.
Seismic Fragilities	Section 12 of the Seismic Fragility Evaluation Report [37] addresses the sources of uncertainty in the SPRA	Refer to Section 5.7, sensitivity cases 2, 3, 4, and 6.

Table A-4 Summary of Potentially Important Sources of Uncertainty

PRA Element	Summary of Treatment of Sources of Uncertainty per Peer Review	Potential Impact on SPRA Results
	model including: Model uncertainty (e.g., the reference ground motion parameter for the quantification), judgment uncertainty (e.g., LOSP fragility), uncertainty due to assumptions (e.g., Switch DLO-PS-26 seismic qualification), and potential fragility improvements (e.g., more detailed analysis or design change). In addition, the various SFR reports and calculations also list the key assumptions made in the fragility analyses.	
Seismic PRA Model	The CGS SPRA peer review team noted that sources of uncertainty and related assumptions are identified and characterized well in the SPRA notebooks.	Refer to Section 5.7, sensitivity cases 1 and 5.

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

The CGS SPRA reflects the plant as of the cutoff date for the SPRA, which was April 18, 2019 [57]. There are no plant changes that would affect the SPRA that have not been reflected in the SPRA model.

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
SHA-G1	20-10	SHA-G1 requires that the horizontal response spectral shape determined in the PSHA is based on site-specific evaluations. The horizontal response spectral shape documented in Energy Northwest letter G02-15-045 to NRC [3], as supported by Bechtel Calculation 25709-000-K0C-0000-00004 [10] for the control point elevation, is not adequately justified as appropriately reflecting characteristic spectral shapes associated with the mean magnitude and distance pairs as determined in the PSHA.	Energy Northwest letter G02-15-045 to NRC dated March 12, 2015, as supported by Bechtel Calculation 25709-000-K0C-0000-00004 [10], documents the horizontal response spectral shape at MAFEs of 1x10-4 and 1x10-5. The horizontal response spectral shape is impacted by several SRA implementation issues including: (1) the defined site profile for both the upper soils and the SMB Stack, including uncertainties, (2) the defined shear modulus reduction and damping versus shear strain curves assigned to each geologic layer, including uncertainties, and (3) the approach used to limit AFs and the technical basis for this limit. The ratio between the peak spectral acceleration and the peak ground acceleration associated with the horizontal response spectral shapes at MAFEs of 1x10-4 and 1x10-5 are significantly larger than commonly associated with either UHRS or individual recordings. The technical issues identified with the SRA and the number of spectral frequencies	Consistent with any reassessment of the CGS SRA, revise and update the horizontal response spectral shape documented in Energy Northwest letter G02-15-045 to NRC as supported by Bechtel Calculation 25709-000-K0C-0000-00004 [10] for the control point elevation. If the revised UHRS at a MAFE of 1x10-5 results in significant change in horizontal spectral shape, ensure that the downstream impact on seismic fragilities is evaluated.	As assessed by the SPRA F&O closure independent assessment team [65], this finding was partially resolved by: 1) satisfactorily revising the CGS hazard [70], and 2) satisfactorily reassessing seismic fragilities based on scaling [71]. Because the spectral shape has changed, it was necessary to assess the downstream impacts on both seismic fragilities and risk quantification. Seismic fragilities were assessed based on scaling, and this analysis was found to be technically sound by the independent F&O closure assessment [65]. The revised hazard and fragilities were convolved as a sensitivity evaluation, and thus the F&O closure independent assessment team assessed this F&O to be partially resolved. See Table A-3 for a discussion of the impact on the SPRA results.

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
			assessed as part of the SRA result in diminished confidence in the technical basis and reliability of the horizontal response spectral shape. Consistent with any reassessment of the CGS SRA revise and update the horizontal response spectral shape documented in Energy Northwest letter G02-15-045 to NRC [3] as supported by Bechtel Calculation 25709-000-K0C-0000-00004 [10] for the control point elevation. If the revised UHRS at a MAFE of 1x10-5 results in significant change in horizontal spectral shape, ensure that the downstream impact on seismic fragilities is evaluated.		