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November 04, 2019

AEP-NRC-2019-56
10 CFR 50.54(f)

Docket Nos.: 50-315
50-316

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

Donald C. Cook Nuclear Plant Units 1 and 2
SEISMIC PROBABILISTIC RISK ASSESSMENT IN RESPONSE TO NEAR TERM TASK FORCE RECOMMENDATION 2.1: SEISMIC

References:

1. Letter from E. J. Leeds and M. R. Johnson, U. S. Nuclear Regulatory Commission (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, Agencywide Document Access Management Systems (ADAMS) Accession No. ML12053A340.

2. Electric Power Research Institute Report 1025287, "Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," dated November 27, 2012, ADAMS Accession No. ML12333A170.

3. Letter from Q. S. Lies, Indiana Michigan Power Company (I&M), to NRC, "Donald C. Cook Nuclear Plant Units 1 and 2, Seismic Hazard and Screening Report (CEUS Sites), Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," dated March 27, 2014, ADAMS Accession Number ML14092A329.

4. Letter from F. Vega, NRC to L. J. Weber, I&M, "Donald C. Cook Nuclear Plant, Units 1 and 2 - Staff Assessment of Information Provided Pursuant to Title 10 of the Code of Federal Regulations Part 50, Section 50.54(f), Seismic Hazard Reevaluations for Recommendation 2.1 of the Near-Term Task Force Review of Insights from The Fukushima Dal-ichi Accident (TAC Nos. MF3873 and MF3874)," dated April 21, 2015, ADAMS Accession No. ML15097A196.

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5. Letter from W. M. Dean, NRC, to Power Reactor Licensees 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 Accession No. ML 15194A015.

6. Letter from L. Lund, NRC, to J. P. Gebbie, I&M, "Donald C. Cook Nuclear Plant, Units 1 and 2 - Response to Request for Extension of Seismic Probabilistic Risk Assessment Submittal," dated February 1, 2018, ADAMS Accession No. ML18011A217.

This letter transmits a seismic probabilistic risk assessment (SPRA) for the Donald C. Cook Nuclear Plant (CNP) in response to the Nuclear Regulatory Commission's (NRC) request for information.

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

Reference 2 provided industry guidance developed by Electric Power Research Institute, regarding the screening, prioritization and implementation details, for the resolution of Fukushima NTTF Recommendation 2.1: Seismic. Reference 2 was used to compare the CNP reevaluated seismic hazard to the CNP design basis hazard. The CNP Units 1 and 2 Seismic Hazard and Screening Report (Reference 3) documented the conclusion that the ground motion response spectrum exceeded the design basis seismic response spectrum in the 1 to 10 Hertz range, and therefore a SPRA was required.

Reference 4 documented the NRC Staff Assessment of the CNP Units 1 and 2 seismic hazard submittal and the Staff's conclusion that once it is adjusted for a layer of beach sand, the CNP seismic hazard reevaluation is suitable for other activities associated with the NRC NTTF Recommendation 2.1: Seismic.

Reference 5 provided the NRC final determination of licensee SPRAs. Table 1a – "Recommendation 2.1 Seismic - Information Requests," of Reference 5, specified the submittal date for the CNP SPRA as June 30, 2018. Reference 6 documented NRC's approval of an extension of the SPRA due date to November 6, 2019.

Enclosure 1 to this letter provides an affirmation statement. Enclosure 2 provides the CNP Units 1 and 2 SPRA Summary Report, which contains the information requested in Enclosure 1, Item (8)B of Reference 1.

This letter contains no new or revised commitments. Should you have any questions please contact Michael K. Scarpello, Regulatory Affairs Director, at (269) 466-2649.

Sincerely,

Quinton A. Lies

Q. Shane Lies
Site Vice President

JRW/kmh

Enclosures:

1. Affirmation
2. Donald C. Cook Nuclear Plant Unit 1 & Unit 2, Seismic Probabilistic Risk Assessment in Response to 10 CFR 50.54(f) Letter with Regard to NTTF Recommendation 2.1, Seismic, dated November 2019.

c: R. J. Ancona – MPSC
R. F. Kuntz – NRC Washington D.C.
EGLE – RMD/RPS
NRC Resident Inspector
D. J. Roberts – NRC Region III
A. J. Williamson – AEP Ft. Wayne, w/o enclosures

Enclosure 1 to AEP-NRC-2019-56

AFFIRMATION

I, Q. Shane Lies, being duly sworn, state that I am the Site Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this request with the U. S. Nuclear Regulatory Commission on behalf of I&M, and that the statements made and the matters set forth herein pertaining to I&M are true and correct to the best of my knowledge, information, and belief.

Indiana Michigan Power Company

Quinton A. Lies

Q. Shane Lies
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 4 DAY OF November, 2019

Danielle Burgoyne
Notary Public

My Commission Expires 04-04-2024

DANIELLE BURGOYNE
Notary Public, State of Michigan
County of Berrien
My Commission Expires 04-04-2024
Acting in the County of Berrien

Enclosure 2 to AEP-NRC-2019-56

Donald C. Cook Nuclear Plant Unit 1 & Unit 2, Seismic Probabilistic Risk Assessment in Response to 10 CFR 50.54(f) Letter with Regard to NTTF Recommendation 2.1, Seismic, dated November 2019.

DONALD C. COOK NUCLEAR PLANT
UNIT 1 & UNIT 2
SEISMIC PROBABILISTIC RISK ASSESSMENT IN RESPONSE TO 50.54(f) LETTER WITH REGARD TO NTTF RECOMMENDATION 2.1, SEISMIC

November 2019

DONALD C. COOK NUCLEAR PLANT (CNP) UNITS 1 & 2
SEISMIC PROBABILISTIC RISK ASSESSMENT SUMMARY REPORT

Table of Contents

1.0 Purpose and Objective (Abbreviations and acronyms are defined in Section 8.0 of this enclosure.)

Following the accident at the Fukushima Dai-ichi nuclear power plant resulting from the March 11, 2011, Great Tohoku Earthquake and subsequent tsunami, the NRC established an 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. The NRC issued a 10 CFR 50.54(f) letter [1] on March 12, 2012, requesting information to assure that these recommendations are addressed by all U.S. nuclear power plants. The 10 CFR 50.54(f) letter [1] requests that licensees and holders of construction permits under 10 CFR Part 50 reevaluate the seismic hazards at their sites against present-day NRC requirements and guidance.

A comparison between the reevaluated seismic hazard and the design basis for the CNP Units 1 and 2 was performed, in accordance with the guidance in the SPID [2], and submitted [3] to the NRC. That comparison concluded that the GMRS, which was developed based on the reevaluated seismic hazard, exceeds the design basis seismic response spectrum in the 1 to 10 Hz range. Therefore, an SPRA has been developed in response to Item (8) in Enclosure 1 of the 10 CFR 50.54(f) letter [1].

This report describes the SPRA developed for CNP, and provides the information requested by Item (8)(B) of Enclosure 1 of the 10 CFR 50.54(f) letter [1], and by Section 6.8 of the SPID [2]. The CNP SPRA model has been peer reviewed (as described in Appendix A of this report) and found to be of appropriate scope and technical capability for use in assessing the seismic risk for CNP, identifying which SSCs are important to seismic risk, and describing plant-specific seismic issues and associated actions planned or taken in response to the 10 CFR 50.54(f) letter [1]. 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 CNP SPRA.

2.0 Information Provided in This Report

The following information was requested in the 10 CFR 50.54(f) letter [1], Enclosure 1, Paragraph (8)B, for plants performing an SPRA.

(1) A List of the significant contributors to SCDF for each seismic acceleration bin, including importance measures (e.g., Risk Achievement Worth, F-V and Birnbaum)

(2) A summary of the methodologies used to estimate the SCDF and SLERF, including the following:

i. Methodologies used to quantify the seismic fragilities of SSCs, together with key assumptions
ii. SSC fragility values with reference to the method of seismic qualification, the dominant failure mode(s), and the source of information
iii. Seismic fragility parameters
iv. Important findings from plant walkdowns and any corrective actions taken
v. Process used in the seismic plant response analysis and quantification, including the specific adaptations made in the Internal Events PRA model to produce the seismic PRA model and their motivation
vi. Assumptions about Containment performance

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

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

Note that 10 CFR 50.54(f) letter [1], Enclosure 1, Paragraphs (1) through (7), regarding the seismic hazard evaluation reporting, also apply, but have been satisfied through the previously submitted CNP Seismic Hazard and Screening Report [3]. Additionally, 10 CFR 50.54(f) letter [1], Enclosure 1, Paragraph (9) requests an SFP evaluation. The CNP SFP evaluation has been submitted [53] and accepted [54] by the NRC Staff.

Table 2-1 of this report provides a cross-reference between the 10 CFR 50.54(f) [1] reporting items noted above and the location in this report where the corresponding information is discussed.

The CNP SPRA has been developed and documented in accordance with the SPID [2], which defines the principal parts of an SPRA. The main elements of the CNP SPRA correspond to those described in Section 6.1.1 of the SPID, i.e.:

- Seismic hazard analysis
- Seismic fragility analysis
- Systems/accident sequence analysis
- Risk quantification

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

The CNP SPRA and associated documentation has been peer reviewed against the PRA Standard [4] in accordance with the process defined in NEI 12-13 [5], as documented in the CNP SPRA Peer Review Report [6]. The CNP 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, insights, and the peer review process and results. This report is intended to meet the 10 CFR 50.54(f) information request [1] in a manner that will enable the NRC staff to determine the validity of key input data and calculation models used, and to assess the sensitivity of the results of key aspects of the analysis.

The content of this report is organized as follows:

Section 3.0, CNP Seismic Hazard and Plant Response, provides information related to the CNP seismic hazard analysis.

Section 4.0, Determination of Seismic Fragilities for the SPRA, provides information related to the determination of seismic fragilities for CNP SSCs included in the seismic plant response.

Section 5.0, Plant Seismic Logic Model, provides information regarding the plant seismic response model (seismic accident sequence model) and the quantification of results.

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

Section 7.0, References, identifies the documents referenced in this report.

Section 8.0, Abbreviations and Acronyms, provides a list of acronyms used in this report.

Appendix A of this report provides an assessment of SPRA technical adequacy for Response to NTTF 2.1 Seismic 10 CFR 50.54(f) Letter [1], including a summary of the CNP SPRA peer review.

Table 2-1
Cross-Reference for 10 CFR 50.54(f) Enclosure 1 SPRA Reporting

10 CFR 50.54(f) Letter [1] Reporting Item	Description	Location in this Report
B(1)	List of the significant contributors to SCDF, including importance measures.	Section 5
B(2)	Summary of the methodologies used to estimate the SCDF and SLERF.	Sections 3, 4, 5
B(2)i	Methodologies used to quantify the seismic fragilities of SSCs, together with key assumptions.	Section 4
B(2)ii	SSC fragility values with reference to the method of seismic qualification, the dominant failure mode(s), and the source of information.	Table 5.4-2 provides 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 F-V or RRW. Seismic qualification reference is not provided as it is not relevant to development of SPRA.
B(2)iii	Seismic fragility parameters.	Table 5.4-2 provides fragilities (Am and beta) information for the top risk significant SSCs based on standard importance measures such as F-V or RRW.
B(2)iv	Important findings from plant walkdowns and any corrective actions taken.	Section 4.2
B(2)v	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
B(2)vi	Assumptions about Containment performance.	Sections 4.3, 5.1.4.3, and 5.1.7

Table 2-1
Cross-Reference for 10 CFR 50.54(f) Enclosure 1 SPRA Reporting

10 CFR 50.54(f) Letter [1] Reporting Item	Description	Location in this Report
B(3)	Description of the process used to ensure that the SPRA is technically adequate, including the dates and findings of any peer reviews.	Appendix A
B(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 [2] Section 6.8 Item Description	Location in this Report
Documentation criteria for a SPRA are identified throughout the PRA Standard [4]. Utilities are expected to retain that documentation consistent with the Standard.	This was an expectation relative to documentation of the SPRA, that the utility retains, to support application of the SPRA to risk-informed plant decision-making.
A report should be submitted to the NRC summarizing the SPRA inputs, methods, and results.	The entire report addresses this.
The level of detail needed in the submittal should be sufficient to enable the NRC to understand and determine the validity of all input data and calculation models used.	The entire report addresses this. The report identifies key methods of analysis, 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.	The entire report addresses this. Sensitivity results are discussed in the 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.	The entire report addresses this.
It is not necessary to submit all of the SPRA documentation for such an NRC review. Relevant documentation should be cited in the submittal, and be available for NRC review in easily retrievable form.	The entire report addresses this. The report summarizes important information from the SPRA, with detailed information in lower tier documentation.

*The items listed in this table do not include those designated in SPID Section 6.8 as "guidance."

3.0 CNP Seismic Hazard and Plant Response

CNP is a dual unit Westinghouse PWR housed in an Ice Condenser Containment. The units are situated on the eastern shore of Lake Michigan, in Lake Township, Berrien County, approximately 11 miles south-southwest of the city of Benton Harbor, MI.

The geological profile of the site consists of a simple stratigraphic sequence of deposits consisting of a surface deposit of dune sand which overlies older beach sand, and is underlain by glacial lake clays, glacial till, and shale (which is part of the bedrock sequence). As documented in the UFSAR [17], bedrock consists of a mixed sequence of sedimentary deposits including shale, limestone, sandstone, and dolomite ranging in age from the Cambrian to Pennsylvanian period. The bedrock sequence is underlain by Precambrian igneous and metamorphic rocks, which are collectively defined as the crystalline basement.

CNP is a soil site, the containment areas as well as the remainder of the plant areas were excavated to an elevation of 588 ft. Major Category 1 structures are founded on mat foundations installed on the overlying compacted sand, re-compacted sand or stiff clay deposits. The Class I Tanks were founded on compacted backfill. The areas were first excavated down to the dense beach sands and then brought back to foundation grade with controlled compacted backfill.

3.1 Seismic Hazard Analysis

The seismic hazard analysis determines the annual frequency of exceedance for selected ground motion parameters. The analysis involves use of earthquake source models, ground motion attenuation models, characterization of the site response (e.g., soil column), and accounts for the uncertainties and randomness of these parameters to arrive at the site seismic hazard.

Detailed information regarding the CNP site hazard was provided to the NRC by I&M's reevaluated seismic hazard report [3] in response to the NTTF 2.1 seismic information request in the 10 CFR 50.54(f) letter [1]. The NRC Staff assessed the information provided and determined that the compacted beach sand layer beneath the power plant area had not been included in the seismic hazard analysis. The Staff stated in [15] that once the GMRS was adjusted to account for the sand layer, the GMRS would be suitable for other seismic activities associated with NTTF Recommendation 2.1. The GMRS at the Containment Building Control Point Elevation was revised using an updated site profile that included the sand layer. The revised GMRS documented in [62] was used for the Spent Fuel Pool Integrity Analysis submitted by [53].

The legacy site investigation studies cited in UFSAR [17], along with the more recent geotechnical and geophysical investigations at the ISFSI site, were used to develop the site profile for the purpose of site-response analysis for SPRA. Moreover, available information from Palisades Nuclear Power Plant (approximately 30 miles north-northeast of CNP) was used along with the other available information to characterize the site profile at CNP.

Additional site description and composite profile development are described in Appendix G of [25].

3.1.1 Seismic Hazard Analysis Methodology

The seismic hazard analysis submitted to NRC in the CNP Seismic Hazard and Screening Report was revised as described above in Section 3.1. The updated CNP site profile is given in Table 3-1 and Table 3-2, and shown in Figure 3-1 of this report.

UHRS at MAFEs of 10^{-4}, 10^{-5}, and 10^{-6} along with FIRS were calculated at control points included in the SPRA. Developed FIRS were not directly used for the SPRA.

RLEs equivalent to 0.8 times the UHRS at MAFEs of 10^{-5} were used for the SPRA. However, in this report, calculated FIRS are shown as representative of seismic hazard calculations. GMRS was calculated at the Containment Building Control Point Elevation. Developed GMRS was not used directly in the SPRA. However, it was recalculated using the updated soil profile developed for the SPRA as discussed above. GMRS was only used for the Spent Fuel Pool Integrity Analysis. However, the revised GMRS is presented in this report for completeness.

Table 3-1:
Site Profile Developed for the SPRA

Geologic Unit	*Thickness (ft.)
Compacted granular backfill	20
Compact beach sand	34
Glacial lake clays	48
Glacial till	71
Bedrock (Paleozoic sedimentary rocks)	3935

*The ground surface is at elevation 608 ft.

Table 3-2
Shear-Wave Velocity, Unit Weight, and Poisson's Ratio for the SPRA Profile

Geologic Unit	Elevation (Top of Geologic Unit) (ft.)	Vs (ft./sec.)	Unit Weight (lb./ft.^{3})	Poisson's Ratio
Compacted granular backfill 0-10 ft.	608	634	130	0.3
Compacted granular backfill 10-20 ft.	598	847		
Compact beach sand	588	800	130	0.48
Glacial lake clays	554	1100	133	0.48
Glacial till	506	1460	147	0.48
Paleozoic sedimentary rocks	435	6000 at top with velocity gradient of 0.5 ft./sec./ft.	150-170	0.24
Crystalline basement	-3500	9285	170	-

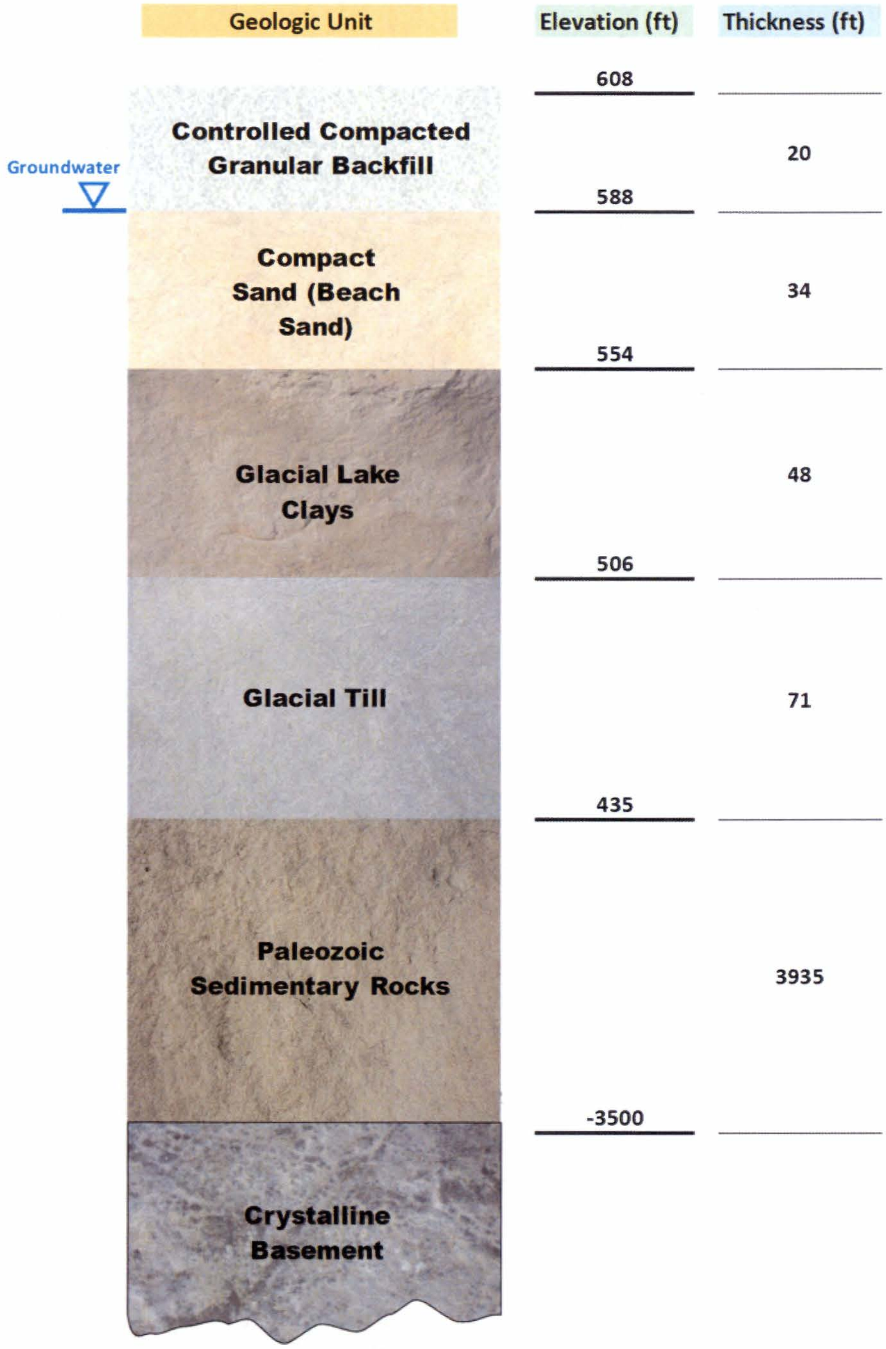


Figure 3-1

Site geologic profile developed for SPRA

The seismic hazard analysis was performed for each control point selected for the project using the 2012 CEUS-SSC [19], 2013 EPRI GMMs [20], and site-specific amplification factors using the updated site profile developed for the SPRA.

The control point elevations are shown below for Auxiliary building, Containment structures, Turbine Building/Screen House and ground surface for the development of the FIRS.

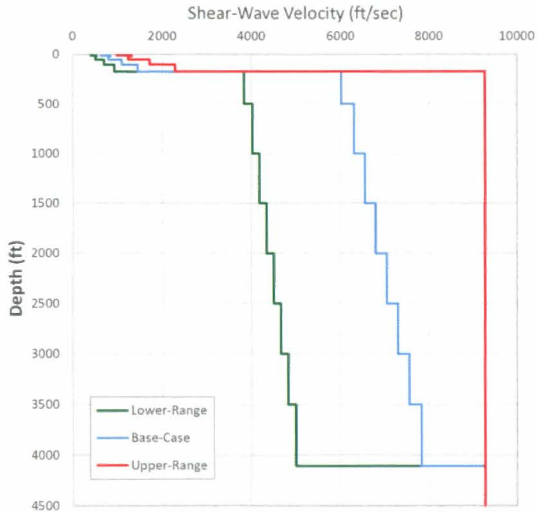
The smooth UHRS were calculated motion response spectrum using log-log interpolation from soil spectral shapes to determine the spectral acceleration at each spectral frequency for the 10^{-4} and 10^{-5} per year hazard levels. The FIRS/GMRS was calculated from the 10^{-4} and 10^{-5} UHRS at each spectral frequency. The site-specific amplification factors were developed in accordance with guidance in the SPID [2]. The seismic hazard analysis accounts for the uncertainties and randomness in the seismic source model, ground-motion model, and site response.

A seismic hazard analysis was performed to determine the CNP GMRS level response with a variety of control points. The final seismic hazard response spectra are associated with Containment buildings at elevation 584 ft. for the GMRS and an additional four FIRS as listed below:

Spectrum	Facility	Elevation	Analysis
GMRS	Containment buildings	584 ft.	Free-field outcrop
FIRS	Other (Free Field)	608 ft.	Free surface
FIRS	Auxiliary building	581 ft.	Truncated soil column
FIRS	Containment buildings	584 ft.	Full column outcrop
FIRS	Turbine bldg./Screenhouse	559 ft.	Truncated soil column

Each FIRS was calculated as a full column outcrop motion or truncated soil column response spectra, as shown above, while the GMRS was calculated as a free-field outcrop motion as defined in the applicable NRC Staff guidance [18].

The upper-range and lower-range shear-wave velocity profiles for the SPRA were calculated using the base-case profile developed for the SPRA and the epistemic uncertainty in shear-wave velocity of 0.35 in natural log units, recommended by the SPID [2] for sites where geophysical information such as very limited shear-wave velocity data exists. The epistemic uncertainty in depth to the bedrock is represented by the upper-range profile, for which hard-rock basement occurs directly beneath the soil layers. The three cases (i.e., base-case, upper-range and lower-range profiles) were used in the site response analysis to account for the epistemic uncertainty. The three profiles are shown in Figure 3-2 of this report. In the CNP Seismic Hazard and Screening Report [3], the upper-range and lower-range profiles were developed using a natural log standard deviation of 0.35 for each of the base-case profiles resulting in total six profiles.



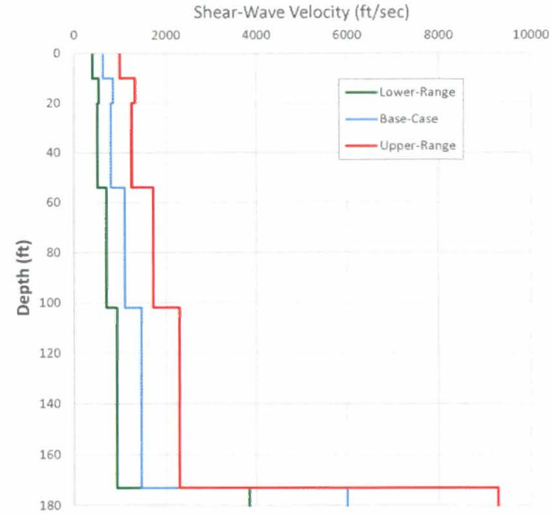


Figure 3-2

CNP base-case, upper-range and lower-range Vs profiles.
Left: Vs profile down to the hard-rock basement.
Right: Vs profile in the top 180 ft.

The aleatory uncertainty in the shear-wave velocity was accounted for by randomization according to the SPID [2].

Shear modulus reduction and damping curves were assigned to each layer following the recommendations in the SPID [2] and similar to the CNP Seismic Hazard and Screening Report [3]. The aleatory and epistemic uncertainties in shear modulus reduction and damping curves were accounted for in the site response analysis following the guidance in Appendix B, of the SPID and similar to the approach used in the CNP Seismic Hazard and Screening Report [3]. For materials at depths exceeding 500 ft., a linear response was assumed following the SPID [2].

The total effective kappa for the CNP profile was calculated in accordance with the recommendations from the SPID [2]. The RVT approach is consistent with the SPID recommendations and was also employed in the CNP Seismic Hazard and Screening Report [3].

In the updated site response analysis for the SPRA, the control motions were developed on a site-specific basis rather than on a generic basis. The input motions were derived using the low-frequency and high-frequency control motions at MAFEs of 10^{-4}, 10^{-5}, and 10^{-6} obtained from deaggregation of the CNP hard-rock hazard. Eleven input control motions with PGAs covering the range of 0.1g to 1.5g were obtained by scaling the 10^{-4}, 10^{-5} and 10^{-6} rock high-frequency and low-frequency spectra.

The mean and fractile hazard curves were calculated at seven spectral frequencies of 0.5 Hz, 1.0 Hz, 2.5 Hz, 5.0 Hz, 10.0 Hz, 25.0 Hz, and 100.0 Hz, at which EPRI GMMs are available. The UHRS were calculated at MAFEs of 10^{-4}, 10^{-5}, and 10^{-6} using the hazard curves obtained at the seven spectral frequencies. The intermediate frequencies were interpolated from soil spectral shapes calculated in the site amplification analysis. Subsequently, the horizontal GMRS and FIRS at each control point were calculated using the UHRS.

For each FIRS, a set of 100 discrete hazard curves for PGA were calculated using the logic tree end branch hazard curves. The reduction of hazard curves down to 100 hazard curves was accomplished with an algorithm that uses a range of logarithmic accelerations that replicate the mean and uncertainty in ground motion at selected AFEs of 10^{-4}, 10^{-5}, and 10^{-6}.

The CNP Seismic Hazard Analysis for future risk evaluation activities provides the horizontal FIRS at the Control Point elevations for the SPRA project, which are shown in Figure 3-3 of this report.

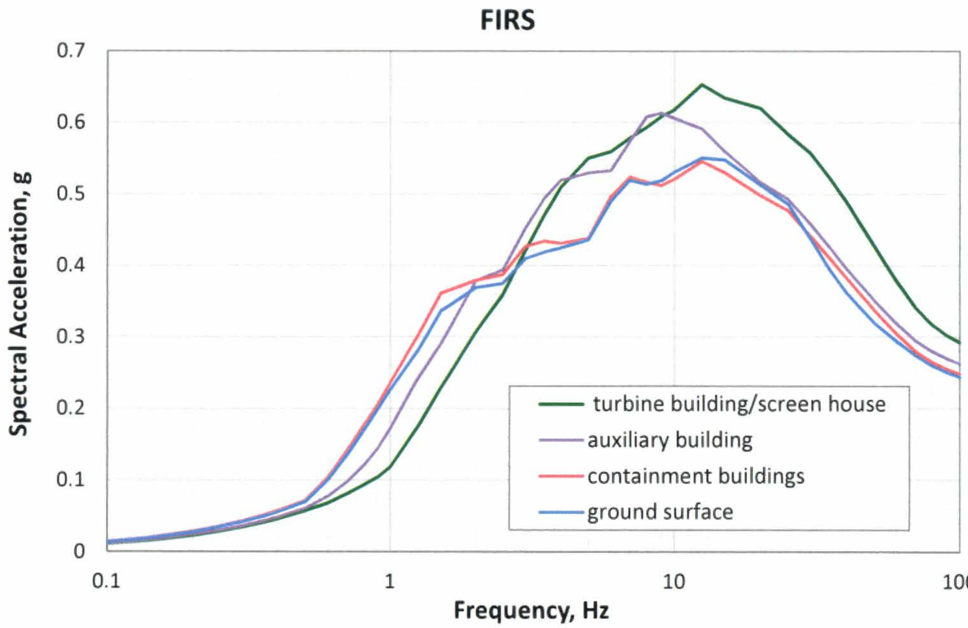


Figure 3-3

CNP horizontal FIRS for the Ground Surface, Auxiliary Building, Containment and Turbine building and Screenhouse Foundations.

The methodology for obtaining the vertical response spectra is discussed in Section 3.1.4 of this report. Additional details regarding the Seismic Hazard Analysis Methodology are included in the CNP seismic hazard analysis summary [25].

3.1.2 Seismic Hazard Analysis Technical Adequacy

As discussed in Section 3.1 of this report, CNP seismic hazard information submitted in the CNP Seismic Hazard and Screening Report was updated to include the beach sand layer in the geological profile. I&M notified [55] the NRC

Staff that correction of seismic hazard information was complete. The NRC responded [56] that the updated seismic hazard information was adequate to proceed with SPRA and the revised information would be evaluated when SPRA was submitted to the NRC.

The CNP hazard analysis has been subjected to an independent peer review against the SHA requirements of the PRA Standard [4]. The SPRA was peer reviewed relative to Capability Category II for the full set of requirements in the Standard. The results of the peer review are documented in the CNP SPRA Peer Review Report [6]. I&M considers that these results confirm that the seismic hazard information is technically adequate for use in the SPRA.

The peer review assessment, and subsequent disposition of peer review findings, are described in Appendix A of this report.

3.1.3 Seismic Hazard Analysis Results and Insights

Table 3-3 of this report provides the final seismic hazard results at the ground surface used as input to the CNP SPRA, in terms of exceedance frequencies as a function of PGA level for the mean and several fractiles. Information on the vertical hazard is discussed in Section 3.1.4 of this report.

Table 3-3
CNP Mean and Fractile Exceedance Frequencies
(Ground Surface)

	Exceedance Frequencies per Year			
PGA (g)	**0.16**	**0.5**	**MEAN**	**0.84**
0.1	5.91E-05	1.51E-04	2.72E-04	4.25E-04
0.15	2.64E-05	7.13E-05	1.33E-04	2.10E-04
0.3	4.83E-06	1.55E-05	3.35E-05	5.42E-05
0.5	1.05E-06	3.79E-06	1.01E-05	1.77E-05
0.75	2.57E-07	1.08E-06	3.29E-06	5.91E-06
1.0	8.00E-08	4.13E-07	1.31E-06	2.39E-06
1.5	1.25E-08	9.11E-08	3.02E-07	5.42E-07
3	3.57E-10	3.90E-09	1.78E-08	2.60E-08

The following main assumptions were made in the seismic hazard analysis:

- It was assumed that the CEUS-SSC earthquake catalog adequately characterized the regional seismicity. New seismicity since the compilation of the CEUS-SSC catalog was evaluated and it was concluded that an update to the activity rates was not necessary.
- Minor errors in maximum magnitude distributions for several background sources were discovered in the CEUS-SSC [21]. It was assumed that the revised maximum magnitude values provided in the EPRI guidance along with the recurrence parameters, recalculated using these magnitudes,

were correct and appropriate for use in seismic hazard analysis for the CNP site.

- It was assumed that the developed CNP site profiles (base-case, upper-range, and lower-range profiles) were adequate to characterize the CNP site dynamic properties.

The contribution of different parameters of the logic tree to the uncertainty in seismic hazard were investigated for MAFEs of 10^{-4} and 10^{-5} at 1 Hz and 10 Hz for each FIRS calculation. The contribution of different parameters to the uncertainty was similar at all control points.

At 1 Hz, the major contributor to uncertainty in ground motion at MAFEs of 10^{-4} and 10^{-5} is the epistemic uncertainty in the shear-wave velocity profile (i.e., base-case, upper-range, and lower-range profiles). The next highest contributor to the uncertainty is the epistemic uncertainty in EPRI GMM clusters. Different EPRI GMM clusters, and different RLME annual frequencies and RLME magnitudes for the NMFS are the next major contributors to the uncertainty at 1 Hz.

At 10 Hz, the major contributors to uncertainty in ground motion at MAFEs of 10^{-4} and 10^{-5} are the epistemic uncertainty in the shear-wave velocity profile and the epistemic uncertainty in EPRI GMM clusters. The next highest contributor to the uncertainty is the different EPRI GMM clusters.

An update to the CEUS SSC earthquake catalog was not required for this site-specific study, as determined by the updated seismic catalog evaluation in Appendix B to the CNP seismic hazard analysis summary [25]. The SSC was assessed for new geologic information in Appendix B to [25]. It was concluded that no updates to the CEUS SSC model were required.

In the CNP SPRA plant model described in Section 5 of this report, the hazard data in Table 3-3 of this report was discretized into 8 intervals, with parameters as listed in Table 3-4.

Table 3-4
Acceleration Intervals and Interval Frequencies as Used in SPRA Model

Interval Designator	Interval Lower Bound PGA (g)	Interval Upper Bound PGA (g)	Representative Magnitude PGA (g)	Interval Mean Frequency
%G01	0.1	0.2	0.1414	1.94E-04
%G02	0.2	0.3	0.2449	4.47E-05
%G03	0.3	0.4	0.3464	1.62E-05
%G04	0.4	0.5	0.4472	7.24E-06
%G05	0.5	0.6	0.5477	3.88E-06
%G06	0.6	0.7	0.6481	2.18E-06
%G07	0.7	0.8	0.7483	1.35E-06
%G08	0.8	Greater	0.88	2.69E-06

3.1.4 Horizontal and Vertical GMRS

The horizontal GMRS is given in Table 3-5 and shown in Figure 3-4 of this report. No vertical GMRS was developed for the SPRA.

Table 3-5
Horizontal Ground Motions Response Spectra at the Containment Foundation Control Point

Frequency (Hz)	Horizontal GMRS (g)
100	2.63E-01
90.0	2.71E-01
80.0	2.82E-01
70.0	2.99E-01
60.0	3.25E-01
50.0	3.63E-01
40.0	4.10E-01
35.0	4.40E-01
30.0	4.74E-01
25.0	5.07E-01
20.0	5.30E-01
15.0	5.55E-01
12.5	5.91E-01
10.0	5.93E-01
9.00	5.96E-01
8.00	5.94E-01
7.00	5.59E-01
6.00	5.24E-01
5.00	5.13E-01
4.00	4.99E-01
3.50	4.83E-01
3.00	4.45E-01
2.50	3.88E-01
2.00	3.56E-01
1.50	2.77E-01
1.25	2.35E-01
1.00	1.68E-01

Table 3-5
Horizontal Ground Motions Response Spectra at the Containment Foundation Control Point

Frequency (Hz)	Horizontal GMRS (g)
0.900	1.42E-01
0.800	1.19E-01
0.700	9.61E-02
0.600	7.63E-02
0.500	6.03E-02
0.400	4.83E-02
0.350	4.22E-02
0.300	3.62E-02
0.250	3.02E-02
0.200	2.41E-02
0.150	1.81E-02
0.125	1.51E-02
0.100	1.21E-02

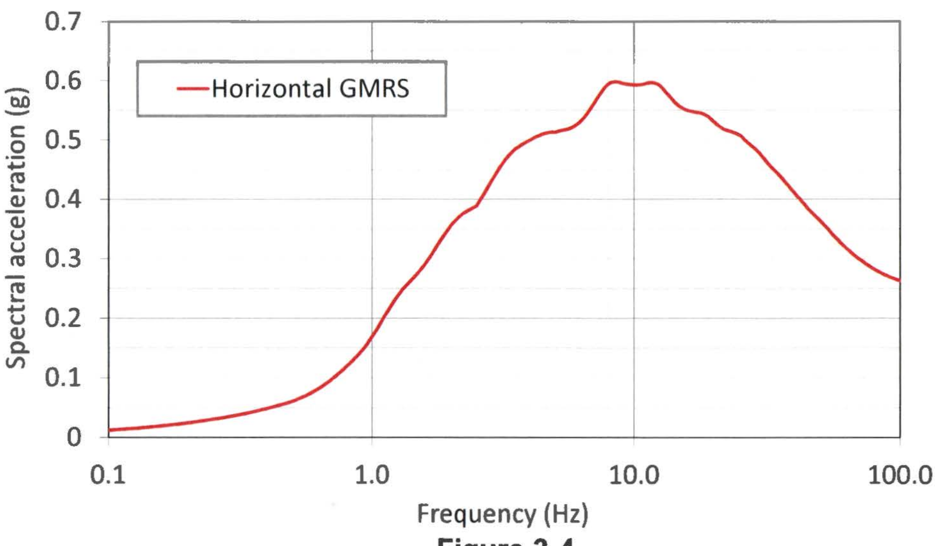


Figure 3-4
CNP horizontal GMRS at the Containment Foundation Control Point

At control points for which horizontal FIRS were obtained, the vertical FIRS were also calculated. At these control points, mean vertical UHRS at MAFEs of 10^{-4} and 10^{-5} were calculated by scaling those horizontal spectra by appropriate V/H ratios. The vertical FIRS at each control point elevation was calculated using the obtained vertical spectra at MAFEs of 10^{-4} and 10^{-5}. For soil profiles, V/H ratios developed for western North America were modified according to the frequency-axis scaling approach documented in Appendix A of EPRI 3002004396 [22]. Because V/H ratios change primarily at high-frequencies, de-aggregated magnitude and distance for high frequencies were used to develop these ratios using empirical relations from western North America scaled to CEUS conditions.

The horizontal and vertical FIRS for the ground surface along with the V/H ratios are given in Table 3-6 of this report and shown in Figure 3-5. This was also performed for the control points at the Containment, Turbine Building/Screenhouse, and the Auxiliary Building. The V/H ratios at each control point were developed using the methodology described in this section of this report. For V/H ratios, the empirical V/H ratios, developed for western North America, were evaluated for control point site conditions and high frequency earthquake scenarios obtained from seismic hazard at MAFEs of 10^{-4} and 10^{-5}, modified by frequency-axis scaling for CEUS conditions, and averaged. Lower-bound V/H ratios from NUREG/CR-6728 [24], were adopted as a lower bound for frequencies below 20 Hz for 10^{-5} and below 25 Hz for 10^{-4}. The V/H ratios for MAFEs of 10^{-4} and 10^{-5} were calculated using vertical 10^{-4} and 10^{-5} UHRS. This was a more traceable way to calculate the vertical FIRS than deriving a V/H ratio directly for the GMRS level.

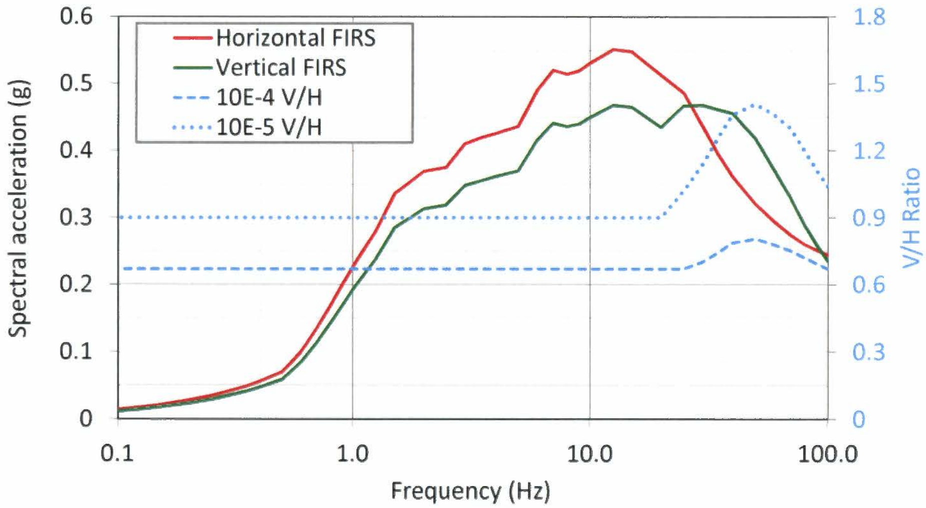


Figure 3-5

CNP ground surface horizontal and vertical FIRS and V/H ratios for MAFEs of 10^{-4} and 10^{-5}

Table 3-6
CNP Ground Surface Horizontal and Vertical FIRS and V/H Ratios for MAFEs of 10^{-4} and 10^{-5}

Frequency (Hz)	Horizontal FIRS (g)	10^{-4} V/H	10^{-5} V/H	Vertical FIRS (g)
100	2.44E-01	6.70E-01	1.04E+00	2.35E-01
90	2.51E-01	6.93E-01	1.11E+00	2.59E-01
80	2.60E-01	7.20E-01	1.20E+00	2.89E-01
70	2.74E-01	7.52E-01	1.30E+00	3.30E-01
60	2.94E-01	7.79E-01	1.36E+00	3.71E-01
50	3.20E-01	8.05E-01	1.41E+00	4.18E-01
40	3.62E-01	7.87E-01	1.36E+00	4.56E-01
35	3.94E-01	7.44E-01	1.26E+00	4.61E-01
30	4.37E-01	7.02E-01	1.14E+00	4.68E-01
25	4.86E-01	6.70E-01	1.02E+00	4.67E-01
20	5.13E-01	6.70E-01	9.00E-01	4.35E-01
15	5.48E-01	6.70E-01	9.00E-01	4.65E-01
12.5	5.51E-01	6.70E-01	9.00E-01	4.68E-01
10	5.31E-01	6.70E-01	9.00E-01	4.50E-01
9	5.19E-01	6.70E-01	9.00E-01	4.40E-01
8	5.14E-01	6.70E-01	9.00E-01	4.36E-01
7	5.20E-01	6.70E-01	9.00E-01	4.41E-01
6	4.90E-01	6.70E-01	9.00E-01	4.16E-01
5	4.36E-01	6.70E-01	9.00E-01	3.70E-01
4	4.25E-01	6.70E-01	9.00E-01	3.61E-01
3.5	4.19E-01	6.70E-01	9.00E-01	3.55E-01
3	4.10E-01	6.70E-01	9.00E-01	3.48E-01
2.5	3.75E-01	6.70E-01	9.00E-01	3.19E-01
2	3.69E-01	6.70E-01	9.00E-01	3.13E-01
1.5	3.36E-01	6.70E-01	9.00E-01	2.85E-01
1.25	2.80E-01	6.70E-01	9.00E-01	2.38E-01
1	2.26E-01	6.70E-01	9.00E-01	1.92E-01

Table 3-6
CNP Ground Surface Horizontal and Vertical FIRS and V/H Ratios for MAFEs of 10^{-4} and 10^{-5}

Frequency (Hz)	Horizontal FIRS (g)	10^{-4} V/H	10^{-5} V/H	Vertical FIRS (g)
0.9	1.99E-01	6.70E-01	9.00E-01	1.68E-01
0.8	1.68E-01	6.70E-01	9.00E-01	1.43E-01
0.7	1.35E-01	6.70E-01	9.00E-01	1.14E-01
0.6	9.97E-02	6.70E-01	9.00E-01	8.46E-02
0.5	6.98E-02	6.70E-01	9.00E-01	5.92E-02
0.4	5.58E-02	6.70E-01	9.00E-01	4.74E-02
0.35	4.88E-02	6.70E-01	9.00E-01	4.14E-02
0.3	4.19E-02	6.70E-01	9.00E-01	3.55E-02
0.25	3.49E-02	6.70E-01	9.00E-01	2.96E-02
0.2	2.79E-02	6.70E-01	9.00E-01	2.37E-02
0.15	2.09E-02	6.70E-01	9.00E-01	1.78E-02
0.125	1.74E-02	6.70E-01	9.00E-01	1.48E-02
0.1	1.40E-02	6.70E-01	9.00E-01	1.18E-02

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 CNP SPRA. The subsections provide brief summaries of these elements. The process for developing SSC seismic fragilities for the SPRA is documented in the CNP Fragility Plan [30].

4.1 Seismic Equipment List

For the CNP SPRA, an SEL Notebook [8] was developed that includes those SSCs that are important to achieving safe shutdown following a seismic event, mitigate radioactivity release if core damage occurs, and were included in the SPRA model. The methodology used to develop the SEL was consistent with the guidance provided in the EPRI SPRA Implementation Guide [10].

4.1.1 Summary of SEL Development

The first step in developing the SEL was to determine the potential initiating events that could occur as a result of a seismic event and equipment whose seismic failure could cause the initiating event. The initiating events considered could occur either directly as a result of the earthquake, or consequential events that occur subsequent to the earthquake. The process began with identification of potential seismic initiating events in the FPIE PRA quantification notebook [23].

Based on the FPIE PRA and review of other potential seismic initiators, the following initiating events that could be seismically induced were identified and dispositioned for inclusion in the SPRA:

- Loss of off-site power
- Seismic induced initiating events leading directly to core damage
- Loss of CCW
- Loss of ESW
- ATWS
- Internal flooding
- ISLOCA
- Small, Medium and Large Break LOCA
- Loss of condenser heat sink
- Loss of instrument air
- Loss of main feedwater
- MFLB
- MSLB
- SGTR
- TRA
- Loss of 250 volt DC power
- LOCA beyond ECCS capability (RCS rupture causing inventory loss greater than the ECCS can makeup).

The safety functions that are required to respond to the initiating events identified above were determined based on EPRI NP-6041-SL [7] and the CNP IPEEE Summary Report [49]. These safety functions are:

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

The SEL includes all plant components and structures whose seismic-induced failure could either give rise to an initiating event or degrade capability to mitigate an initiating event. A preliminary SEL was developed based on seismic-relevant portions of the FPIE PRA. This preliminary SEL was then supplemented by a series of reviews intended to identify potential seismically risk-significant components not modeled by the FPIE PRA. Consistent with the SPRA model, the SEL contains all seismically risk-relevant Unit 1 and 2 components.

Enhancements to the SEL were identified by using system P&IDs, area flood maps, and electrical diagrams to ensure that all necessary components were on the SEL.

The following types of equipment were evaluated for addition to the SEL using this process:

- Components required to maintain pressure boundary integrity of the modeled systems.
- Reactor coolant system components (NSSS Components) including: reactor pressure vessel (and supports); reactor internals; Steam Generators; reactor coolant pumps (for RCS integrity), main RCS piping, and pressurizer.
- Distribution systems (i.e., piping, HVAC ducting, and cable trays).
- Electrical panels, cabinets, and instrument racks needed to provide emergency and/or control power for components on the SEL, including main control room bench boards.
- Equipment or instrumentation that would be required per the plant emergency procedures after an earthquake.
- Equipment associated with the Hydrogen Ignition system.
- Equipment associated with the SDG System.
- Components credited for unit crosstie capabilities.
- Isolation valves less than 2 in. nominal size whose correlated failure could exceed the 2 in.^{2} LERF opening size.
- Non-safety related piping and fire sources due to the inclusion of additional fire and flood scenarios as a result of implementing the most recent EPRI guidance on seismically induced fire and flood [57].
- Components whose spurious operation can occur due to relay chatter.
- Structures associated with SEL equipment were added to the SEL; Auxiliary Building, Containment, and Turbine Building /Screenhouse

Seismic walkdown lists were developed and walkdowns performed in conjunction with development of the SEL. Section 4.2 below addresses the walkdown approach that resulted in identifying components added to the base SEL. Plant areas in which operators would need to perform seismic response actions were reviewed for travel path accessibility and evaluated for potential SC-II/I impact to components.

There were also several components on the SEL that were SC-I components that were determined to fall within the ROB, as defined in SQUG documentation [31]. The ROB states that child components mounted on or in larger pieces of parent equipment do not have to be considered separately. These ROB components were tracked in the fragility database by identification of the parent item.

The final SEL contains over 2,000 items (equipment, components, piping and events). Additional details on the development of the SEL are included in the SEL Notebook [8].

4.1.2 Relay Evaluation

During a seismic event, vibratory ground motion can cause relays to chatter. The chattering of relays can potentially result in spurious signals to equipment. Most relay chatter is either acceptable (does not impact the associated equipment), is self-correcting, or can be recovered by operator action. An extensive relay chatter evaluation was performed for the CNP SPRA, in accordance with SPID [2] Section 6.4.2 and the PRA Standard [4] Section 5-2.2. The evaluation resulted in most relay chatter scenarios being screened from further evaluation because there would be no impact on the associated component's function.

The detailed chatter evaluation performed for relays was documented in the CNP Relay Chatter Analysis [46]. Relays identified as impacting to system operation (due to seismic failure or chatter) were included in the SPRA model. Relay fragility parameters for the chatter-significant relays for the two Unit SPRA from report [46] are presented in Table 3-2 in Attachment A of a high frequency relay analysis report [29]. Based on the relays parameters, relays were placed in relay chatter fragility groups. Each relay fragility group was based on unique combinations of location (building, elevation, room, cabinet / panel) and relay model, yielding groups that can be exactly represented by a single set of fragility parameters.

Seismic risk-significant relays groups for CDF and LERF are identified in Tables 5.4-2 and 5.5-2 of this report.

4.2 Walkdown Approach

This section provides a summary of the methodology and scope of the seismic walkdowns performed for the SPRA. The walkdown plan is detailed in Appendix "A" of the CNP Fragility Plan [30].

The SRTs utilized for the SPRA walkdowns were comprised of at least two qualified SQUG seismic engineering experts with extensive experience in fragility assessment. Walkdowns of those SSCs included on the seismic equipment walkdown list were performed, as part of the development of the SEL. Walkdowns assessed the as-installed condition of these SSCs for use in determining their seismic capacity and performing initial screening, to identify potential SC- II/I spatial interactions and look for potential seismic-induced fire/flood interactions. The fragilities walkdowns were performed in accordance with the criteria provided in EPRI NP- 6041-SL [7] and GIP guidance [31]. The seismic fragility walkdowns were conducted on both Unit 1 and Unit 2 equipment.

The information obtained was used to provide input to the fragilities analysis and SPRA modeling (e.g., regarding correlation and rule-of-the-box considerations).

For some SEL SSCs, walkdowns documented in [61] had been performed in response to NTTF 2.3: Seismic, or in support of the ESEP [9]. Information from those walkdowns was used when the appropriate level of detail needed for the SPRA was available, and were not walked down again for the SPRA.

Non-safety equipment items and systems were initially assigned a conservatively low fragility and not walked down. After initial quantification, if non-safety equipment and piping systems had a large impact on LERF and CDF, these systems were walked down and examined more closely and a system or equipment specific fragility was performed. This included non-safety interaction hazards for fire and flood. Fire and flooding sources were selected via the guidance provided in EPRI report 3002012980 [57].

Potential spray and flooding scenarios from non-safety related piping systems and SEL components were reviewed during the walkdowns, and flood sources, including the fire-protection system were evaluated. Identified scenarios that were included in the model were the following:

- Seismic failure of the NESW piping causing flooding in the Auxiliary Building including the Diesel Generator Rooms
- Seismic failure of eyewash piping in the Battery and Switchgear Rooms
- Charged Fire Protection piping in the Auxiliary Building (Note - most of the Fire Protection piping is either dry or not charged)
- Non-safety piping systems that would be disabled if loss of offsite power occurs but due to close fragility with loss of offsite power, may continue to operate for some period of time:
    - Demineralized Water
    - Primary Water
    - Feedwater
    - Condensate
    - Circulating water

4.2.1 Significant Walkdown Results and Insights

No significant findings were noted during the CNP seismic walkdowns. Walkdown results were documented in the SPRA walkdown report [48].

Components on the SEL were evaluated for seismic anchorage and interaction and documented on SPRA walkdown data sheets [48]. The walkdowns also assessed the effects of component degradation, such as corrosion and concrete cracking for consideration in the development of SEL fragilities. Additionally, walkdowns were performed for operator pathways, and seismic-induced fire and flooding scenarios, that were incorporated into the CNP SPRA model. The walkdown observations were used in developing the SSC fragilities for the SPRA.

4.2.2 Seismic Equipment List and Seismic Walkdowns Technical Adequacy

The CNP SPRA SEL development and walkdowns were subjected to an independent peer review against the pertinent requirements (i.e., the relevant SFR and SPR requirements) in the PRA Standard [4].

The peer review assessment, and subsequent disposition of peer review findings, is described in Appendix A of this report, and establishes that the CNP SPRA SEL and seismic walkdowns were suitable for this SPRA application.

4.3 Dynamic Analysis of Structures

This section summarizes the seismic structure response and SSI analyses methodology used, discusses the significant/limiting seismic structure response, and structure fragility results for the SSCs modeled in the SPRA. This section also discusses important assumptions and sources of uncertainty, and describes identified fragility-related insights. The seismic structure response and SSI analyses methodology is described in detail in the Summary of Building Response Analysis for the CNP SPRA [63].

4.3.1 Fixed-base Analyses

CNP is a soil site, fixed-base analyses are not applicable.

4.3.2 SSI Analyses

CNP is a soil site where all safety related structures are founded on or embedded in the soil where SSI effects are expected. The effects of uncertainty in SSI are dominated by the soil response. Uncertainty in the CNP seismic analysis was accounted for by evaluating the SSI model for three soil profiles, BE, UB, and LB in accordance with ASCE Standard 4-16 [32]. These three soil profiles account for the variation in the measured site-specific soil properties and account for most of the uncertainty in seismic response of the CNP structures. Note that some calculations were performed in accordance with ASCE 4-13 (draft) prior to the issuance of ASCE 4-16 [32]. An evaluation was performed which determined that the technical content did not change between the ASCE 4-13 and ASCE 4-16 [32] versions. That evaluation is documented in Section 5.4 of Report 15C4313-RPT-007 [40].

The GRS were the site-specific 1E-5 UHRS multiplied by a factor of 0.80, which has a horizontal PGA of 0.40g and is defined at an elevation of 609' (plant grade). The RLE is the scaled 0.8* 1E-5 UHRS for each of the three building models included in the SPRA.

The ground motion input was developed using the hazard consistent site-specific BE, UB, and LB strain compatible shear wave velocity profiles, and strain-compatible damping ratio profiles. The ACS SASSI analysis code [45] was used to perform the SSI analysis and the depth of soil considered included all of the soil between bedrock for the site and the base-mats of each structure (about 120 ft. below the bottom of the base-mats for the CNP SC-I structures).

FIRS were developed for each of the structures. The elevations at which the FIRS were developed were established at the beginning of the project based on the general elevations at the bottom of the foundations of the structures. The Auxiliary Building and Turbine Building/Screenhouse are founded at multiple elevations, hence the control points for development of the FIRS were an average elevation where the majority of the structure was founded.

The Containment was modeled as embedded. The Auxiliary Building for the purposes of the SSI analysis was modeled as a surface founded structure. ASCE 4-16 [32] (Section 5.4.2.4 (a)) guidance for shallow embedded structures states that the effects of embedment may be neglected when the depth-to-equivalent-radius ratio is less than 0.3, which was the case for the Auxiliary Building. The Turbine Building/Screenhouse model was a partially embedded structural system, with Turbine Building area modeled as surface-founded and Screenhouse portion being embedded. Satisfying the guidance of ASCE 4-16 [32], cutoff frequency for the SSI analysis was chosen to be 40 Hz for horizontal and vertical response for Containment and 20 Hz for horizontal response and 40 Hz for vertical response for the Auxiliary and the Turbine Building/Screenhouse Buildings.

4.3.3 Structure Response Models

New building models were developed for the response analysis. Detailed 3-dimensional finite element models were developed for the Auxiliary Building and the Turbine Building / Screenhouse. The Containment building that included the East Main Steam Stop Enclosure was modeled with an LMSM developed from plant design documents. The Auxiliary Building, Containment and Turbine Building/Screenhouse are on independent foundations. Therefore, these buildings were analyzed as independent structures.

The ANSYS finite element program was used to model the building structures. The modeling effort was guided by two primary goals: Models are to provide the capability to estimate realistic seismic demand on in-scope SSC and the models are to satisfy review criteria of SPID [2] Section 6.3.1. As needed, each of the three models was enhanced with more detail to capture additional requirements such as torsional effects and in-plane floor flexibility. The models and other enhancements are described in more detail in Appendix 7.1 of report [63]. Also, the evaluation against the SPID review criteria is summarized in Appendix 7.1 of report [63].

For each structure, a single set of input time histories was synthesized from the FIRS. Every set consists of three records, two representing horizontal input motion and one representing vertical input motion. The response spectra associated with the time histories were checked to ensure the match to the RLE shape was acceptable. A sensitivity study was performed with a set of five real earthquake seed records to verify the acceptability of use of a single input time history.

To determine ISRS for equipment, the equipment's location coordinates and the footprint area were determined from the available information (plant's drawings, etc.). From the finite element model a set of nodes within the footprint area was identified. The ISTH corresponding to every node (nine ISTH per node) were retrieved from the database of finite element model responses. For the frequency range of interest, damping and selected direction, the ISRS were generated per every node (three per the node BE, LB and UB) and enveloped individually. The statistics of envelopes (for example for five nodes there will be five bounding envelopes) was calculated to produce the maximum, average or median spectral acceleration value, within the desired range of frequencies. The same sequence was applied to obtain the demand for items located in the Containment building, except that only a single node was used to represent the location of equipment. The bounding of soil cases (the envelope of the UB, LB and BE demand) was considered appropriate for deterministic estimate of the 84% NEP demand. Median acceleration demands were obtained by taking the average of the UB, BE, and LB responses.

ISRS with highly amplified narrow frequency content was clipped for comparison to broad-banded test response spectra and experience-based response spectra, typical of most nuclear power plant components. The guidance in EPRI TR-103959 [35] was performed for the peak clipping process.

The guidance of EPRI NP-6041-SL [7] Section 4, Table 4-1 was originally applied for structural damping of concrete and steel elements for all CNP buildings. Concrete elements were originally assigned 5% of critical damping, steel elements were assigned a conservative 1% of critical damping.

A refined 3-Dimensional FEM was developed for the Auxiliary Building in 15C4313-CAL-010 [69]. The refined model demands were used to update the fragilities of risk-significant components for the refined Auxiliary Building model and in the Containment and Turbine Building sensitivity studies (Attachments C and F of report 15C4313-RPT-003 [63]) concrete and steel damping was assigned as follows:

- For all concrete beams and concrete slabs (except the foundation mat), the Young's modulus was reduced by 50% as required per ASCE 4-16 [32] and Response 2 damping of 7% was used for these elements

- For the concrete walls, concrete columns and basemat, un-cracked properties were used and the rigidity was taken as 100% of the Young's modulus per ASCE 4-16 [32] and Response 1 damping of 4% was used for these elements.

- For the superstructure steel frame, Response 1 damping of 4% damping was used, see Table 3-1 per ASCE 4-16 [32].

Results of the response analyses were reviewed with consideration of the structural response of the concrete portions and the response dominated by the soil. It was concluded that explicitly including stiffness reductions for cracked concrete would not result in significant changes in the overall dynamic response. This was confirmed for the Auxiliary Building, Containment and Turbine Building in the sensitivity studies performed in Attachments C, E, and F, respectively in CNP document 15C4313-RPT-003 [63].

The steel superstructures were included in the Auxiliary Building and Turbine Building models for completeness. The assigned originally conservative 1% damping for steel was determined to not impact the SPRA results as no equipment item included in the model are supported by the steel portions of the structures.

Table 4-1 of this report summarizes the type of analysis and model used for each of the structures modeled in the SPRA.

Table 4-1
Description of Structures and Dynamic Analysis Methods for the CNP SPRA

Structure	Foundation Condition	Type of Model	Analysis Method	Comments/Other Information
Containment	Embedded	LMSM	Deterministic SSI	LB, BE, UB cases, one TH set for FIRS
Auxiliary Building	Embedded	FEM	Deterministic SSI	LB, BE, UB cases, one TH set for FIRS
Turbine Building / Screenhouse	Embedded	FEM	Deterministic SSI	LB, BE, UB cases, one TH set for FIRS

4.3.4 Seismic Structure Response Analysis Technical Adequacy

The CNP SPRA Seismic Structure Response and SSI were subjected to an independent peer review against the applicable requirements in the PRA Standard [4].

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

4.4 SSC Fragility Analysis

The SSC seismic fragility analysis considered the impact of seismic events on the probability of SSC failures at a given value of a seismic motion parameter, such as PGA, peak spectral acceleration, and floor spectral acceleration, etc. The SSC seismic fragility evaluations performed for CNP assessed the probability of SSC failures due to a horizontal PGA of 0.40g, which corresponds to the 1E-5 UHRS scaled by a factor of 0.80. The fragilities of the SSCs that participate in the SPRA accident sequences, i.e., those included on the SEL, were 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 plant walkdown.

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

4.4.1 SSC Screening Approach

The SPRA does not include an optional capacity-based screening approach as described in the SPID. Therefore no SSCs on the SEL were screened using a capacity based screening approach. Inherently rugged items were identified per the guidance of EPRI 1025287 and EPRI NP-6041-SL [7] however inherently rugged items were not screened out of the PRA model and were instead assigned conservative fragility parameters. Quantification results confirmed that the modeled inherently rugged items are not risk significant.

A screening process was applied by the fragility team to determine which SSCs on the SEL were to be selected for fragility analysis. The screening process included review of the plant seismic design bases, performance of seismic walkdowns, and application of industry practices related to seismic margin and SPRA studies. The seismic margins screening methodology of EPRI NP-6041-SL [7] was applied.

After completion of the screening process, an SSC was either screened-out or screened-in. The presumptive HCLPF of screened-out items exceeded the RLE level and were assigned a generic fragility. An item that was screened-in required component-specific analysis to address the failure mode cited by the SRT from the walkdown.

The project earthquake screening level for the structures is the 0.80g PGA damped at 5% peak spectral acceleration level, as identified in EPRI NP-6041-SL [7] Section 2, and was defined at the soil surface at the site. The corresponding ground motion in the peak spectral range was approximately two (2) times higher than the SSE ground motion. The project screening level for the equipment was 1.2g PGA, damped at 5% peak spectral acceleration level, as identified in EPRI NP-6041-SL [7] Section 2. The higher screening level for equipment was justified because a majority of the equipment has been qualified to SQUG-GIP caveats. These caveats justify higher (more realistic) presumptive seismic capacities.

Fragility data for screened-out SSCs were typically addressed by components with a generic fragility as discussed in EPRI 1019200 [37] Section 3.4. Generic function fragilities for screened-out SSCs in the CNP SPRA were generally taken to correspond to specific plant areas and consider the seismic demand in the area. Details are provided in Section 4.4.2. Additional fragility analyses were performed for screened-out items at the direction of the PRA team during the quantification and model refinement process. Fragility parameters were developed for the screen-in components for the failure modes identified by the SRT. The HCLPF values from the initial analysis were supplied to the plant response model team for preliminary analysis of SCDF and SLERF. Based on that initial data, dominant contributors were identified. More rigorous methods of analysis were applied to this subset of screened-in SSCs as deemed appropriate and feasible. Under this approach an existing CDFM analysis was used as a reference and a more refined analysis was performed. Refined analysis methods were per EPRI TR-103959 [35] and included using the SOV approach.

4.4.2 SSC Fragility Analysis Methodology

Seismic fragility evaluations were performed for SSCs contributing to CDF or LERF. The SSC fragility analysis was performed in accordance with the SPID [2], and the requirements defined in the PRA Standard [4]. The methods in EPRI reports NP-6041-SL [7], 1019200 [37], and TR-103959 [35] were used for calculation of seismic fragility parameters.

Seismic analyses were initially performed using the conservative, deterministic failure margin (CDFM) method of EPRI NP-6041-SL [7]. Each analysis produced a HCLPF capacity for the SSC item. Nominally the HCLPF is the capacity at which there is 95% confidence of less than 5% probability of failure. Fragility parameters are then produced by using the scaling approach of EPRI 1019200 [37]. This is equivalent to the Hybrid Method discussed in SPID [2] Section 6.4.1.

To produce the initial median capacity for SSCs, the HCLPF value is calculated and the corresponding median capacity is calculated using the SPID [2] Table 6-2 values for βr and βu. Refined analysis methods were per EPRI TR-103959 [35] and included using the SOV approach. The SOV approach is an analysis methodology where the median is established then βr and βu are determined considering all variables. In general, the refined analysis is expected to produce a more accurate median capacity estimate and more accurate log standard deviations.

4.4.2.1 Structures

Structures were either screened-in or screened-out of detailed fragility calculations based on Table 2-3 of EPRI NP-6041-SL [7]. In accordance with that table, evaluation was not required for Class I structures screened to the 0.8g level if design was by dynamic analysis for a SSE of 0.1g PGA or greater, and the structures were designed to ACI 318-63 requirements. The CNP Containment, the concrete portions of the Auxiliary Building (that are composed of SC-I shear walls), the Screenhouse Foundation including ESW Pump Rooms, and the Turbine Building Foundation including AFW Pump Rooms were designed to these requirements. These structures were originally designed by dynamic analysis to a 0.20g PGA SSE and therefore, the screening requirements were met. There were no additional screening criteria to be applied for this screening level.

Structures that screened in and were subject to more detailed analysis included masonry block walls internal to the plant that could interact with equipment on the SEL, the Turbine Building for potential collapse and the Control Room Ceiling.

Structural demands (member forces and acceleration response) required for fragility analysis were derived from seismic models based on recent NRC guidance in the SRP and industry codes and standards (ASCE 4-16, [32] and ASCE 43-05). These seismic models were detailed, three-dimensional, finite element models or LMSM (see Table 4-1) based on plant-specific information and used the 0.80*1E-5 uniform hazard as input motion. As CNP is a soil site, these models accounted for the effects of SSI. The effects of earthquake-induced settlement and liquefaction were evaluated and shown not to be significant factors in the Seismic PRA.

Details regarding the screening of CNP structures for secondary seismic hazards are included in Table 8-1 of the CNP Evaluation of Secondary Hazards [65]. The results of the Fragility calculations for the structures screened in were included in the CNP Fragility Report [40].

4.4.2.2 Components

The CNP component fragilities were derived using a multi-step approach. The EPRI CDFM method described in the Fragility Applications Guide Update [37] was used to develop and assign fragilities to the components as the first step. The EPRI CDFM Method uses the capacity based on EPRI NP-6041-SL [7] and plant-specific demands. The fragility parameters for certain risk-significant components (i.e., important contributors to CDF and/or LERF) were then refined to become more realistic by applying more detailed CDFM or SOV analysis.

Component failure modes included anchorage failures, functional failures, and failure due to seismic interactions. Anchorage capacities were typically calculated based on standard practice. Generic functional fragilities using EPRI NP-6041-SL [7] screening were generally taken to correspond to specific plant areas and consider the seismic demand in the area. Each reported value was based on the lower bound capacity of all credited equipment in the corresponding area, excluding the equipment that was assigned a specific fragility based on anchorage or interaction. Valves were taken to have different fragility groups from other component types. This approach removes excess conservatism that would be introduced with a single bounding fragility.

This approach also allows for an orderly process to refine the plant response model. The location-specific fragility, in effect, provides an initial level of refinement for improvement of the plant response model capability and identification of critical SSC items. During the refinement process updated capacity data was used to refine the generic fragilities for certain equipment using updated capacities available in EPRI 3002002997 [44], EPRI 3002011627 [67] and EPRI 3002013017 [68]. When fragilities for the different failure modes were similar, all fragilities were supplied to the PRA team for inclusion into the model. When one failure mode fragility was dominant over the others, only the dominant failure mode was modeled by the PRA team.

Seismic fragility calculations for critical relays were performed, and the high frequency capacities of risk significant relays were included as basic events in the SPRA model. The generic capacities used were lower than, or the same as, the capacities of these relays in the high frequency range as depicted in References 22 and 44. Therefore, the relay fragility evaluation addressed fragility for high frequency sensitive components as discussed in Section 6.4.2 of the SPID [2] and documented in the Relay Report [29].

The NSSS was evaluated for fragility variables. The NSSS components evaluated includes the reactor vessel, the Steam Generators, the reactor coolant pumps, the pressurizer, reactor internals, control rod drive mechanisms and RCP shutdown seal. The fragility evaluation of these components was based on scaling of the existing design basis analysis or when sufficient data was not available, then a screening capacity was applied.

4.4.2.3 Correlation

Correlation of components (or common cause failure) was considered in accordance with the PRA Standard [4]. Fragility parameters were associated to components through the use of separate fragility groupings. Each fragility group has a set of fragility parameters that are generally associated with a single failure mode with a unique combination of capacity and demand. The failure modes can include generic functional failure, anchorage failure, seismic interaction, structural collapse, or relay chatter. The seismic capacity was determined accordingly. The demand that was associated with the fragility group was determined for the relevant locations of components in the fragility group.

The failure of all components associated with a fragility group were taken to be correlated, which was generally conservative. Fragility groups that cover a large number of components were used whenever possible in order to simplify the SPRA model.

If after initial quantification, large fragility groupings with generic fragilities were found to impact the results of the SPRA, components were reviewed and assigned to new fragility groups if possible, and justified based on the nature of the failure mode and having different capacity and demand. A new fragility group was only created if the failure of the components was reasonably justified to be uncorrelated with the failure of the components in the existing group. Further detailed discussion of correlation and uncorrelation of specific fragility groups is provided in Section 5.1.6

4.4.3 SSC Fragility Analysis Results and Insights

Detailed fragility information for those SSCs found to be risk-significant to CNP are discussed in Section 5 of this report. Table 5.4-2 details SCDF and Table 5.5-2 details SLERF significant-risk contributors. These tables are for Unit 1. Unit 2 results are similar. While there may be small numerical differences between the units, risk insights from Unit 1 are applicable to Unit 2.

4.4.4 SSC Fragility Analysis Technical Adequacy

The CNP SPRA SSC Fragility Analysis was subjected to an independent peer review against the pertinent requirements in the PRA Standard [4]. The peer review assessment, and subsequent disposition of peer review findings, is described in Appendix A of this report, and establishes that the CNP SPRA SSC Fragility Analysis is suitable for this SPRA application.

5.0 Plant Seismic Logic Model

This section summarizes the adaptation of the CNP FPIE PRA model [23] 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 was quantified to determine the overall SCDF and SLERF and to identify the important contributors (e.g., important accident sequences, SSC failures, and human actions). The quantification process also includes an evaluation of sources of uncertainty and provides a perspective on how such sources of uncertainty affect SPRA insights.

5.1 Development of the SPRA Plant Seismic Logic Model

The CNP seismic response model was developed by starting with the CNP FPIE PRA model which has been determined to be technically adequate through the peer review process. The FPIE model of record date is April 6, 2018, as noted in the CNP FPIE Quantification Notebook [23]. FPIE model peer review F&Os were reviewed and dispositioned for applicability to the seismic response model and are documented in Appendix "B" of the CNP SPRA model notebook [50]. FPIE F&Os disposition for the SPRA were determined not to uniquely impact the results of the SPRA.

Adaptation of the FPIE model to SPRA was performed in accordance with guidance in the SPRAIG [10] and the PRA Standard [4]. Adaptation included adding seismic fragility-related basic events to the appropriate portions of the internal events PRA, eliminating some parts of the internal events model that did not apply or that were screened-out, and adjusting the internal events PRA model human reliability analysis to account for response during and following a seismic event. Both random and seismic induced SSC failures were included in the model.

The seismic hazard was modeled using eight discrete hazard intervals (bins) based on increasing peak ground acceleration. The seismic hazard bins are as listed in Table 3-4 as Interval Designators. Each bin was treated as a seismic initiator and the SCDF and SLERF results were summed over all the bins to obtain the total SCDF and SLERF.

The SPRA model does not credit any FLEX installed equipment or mitigating strategies, but does model the low leak-off RCP seals. The following methods were used to develop the seismic plant response model:

5.1.1 – Initiating Events

Initiating events from the FPIE model were reviewed for relevancy to a seismic initiating event. Initiating events relevant to a seismic event were included in the seismic model. Table 5-1 summarizes this review. Complete details of this review are documented in the CNP SPRA Model Notebook [50].

Table 5-1
Internal Initiating Events Review for SPRA

Internal Events Initiating Event	Seismic PRA Considerations	Modeled as an Initiating Event in SPRA?
IE-CCW(-2) Loss of CCW	While seismic-induced CCW failure could occur, the CNP SPRA models this as a mitigating failure (i.e., in response to a general transient or LOCA) rather than as the initiating event. This approach simplifies the SPRA model structure without losing significant accident sequences. Note that the potential for RCP seal LOCA following loss of CCW is retained even when the general transient event tree is used.	No (Modeled as Mitigating Failure)
IE-ESW4 Loss of all ESW	While seismic-induced ESW failure could occur, the CNP SPRA models this as a mitigating failure (i.e., in response to a general transient or LOCA) rather than as the initiating event. This approach simplifies the SPRA model structure without losing significant accident sequences. Note that the potential for RCP seal LOCA following loss of ESW is retained even when the general transient event tree is used.	No (Modeled as Mitigating Failure)
IE-FLOOD Internal Flooding	Seismic-induced flood is included for its impacts on mitigating equipment, but not as a unique initiating event.	No
IE-ISL1/2/3/4 Interfacing System LOCA 1/2/3/4	IE-ISL1: ISLOCA on RHR cooldown suction line (isolable) IE-ISL2: ISLOCA on RHR injection line (isolable) IE-ISL3: ISLOCA on safety injection lines (non-isolable) IE-ISL4: ISLOCA on RHR cooldown return lines (non-isolable) A severe earthquake could induce an ISLOCA by rupturing the SC-I piping and/or valves in the ISLOCA pathways; however, since such an earthquake would broadly fail SC-I equipment resulting in core damage and large early release (via rupture of SC-I penetrations), and this broad impact leading to core damage and release would already be captured by the SPRA model, it is not necessary to model explicitly seismic-induced ISLOCA.	No
IE-LLO Large LOCA	While seismic-induced failure of the reactor vessel, Steam Generators, and reactor coolant pumps are assumed to lead directly to core damage, structural failure of the pressurizer rupturing its surge line is modeled as a large LOCA initiating event.	Yes
IE-MLO Medium LOCA	Seismic-induced medium LOCA may occur by a single rupture (2 in. – 6 in.), or by rupture of multiple smaller lines summing to within the medium LOCA break range.	Yes
IE-SLO Small LOCA	Seismic-induced small LOCA may occur by a single rupture (3/8 in. – 2 in.) or by rupture of multiple smaller lines summing to within the small LOCA break range.	Yes

Table 5-1
Internal Initiating Events Review for SPRA

Internal Events Initiating Event	Seismic PRA Considerations	Modeled as an Initiating Event in SPRA?
IE-LOCHS Loss of Condenser Heat Sink	The SIET uses a general transient as the minimum initiating event for ground motion exceeding an Operating Basis Earthquake. Loss of condenser heat sink is a possible initiating event for this event tree. CNP success criteria typically uses loss of main feedwater as the reference transient event since it minimizes initial Steam Generator inventory. For this reason, this type of transient is already subsumed within the general transient modeling.	No (Modeled as Mitigating Failure)
IE-LOIA Loss of Instrument Air	While seismic-induced instrument air system failure could occur, the CNP SPRA models this as a mitigating failure (for example in response to a general transient or LOCA) rather than as the initiating event. This approach simplifies the SPRA model structure without losing significant accident sequences.	No (Modeled as Mitigating Failure)
IE-LOMF Loss of Main Feedwater	The availability of main feedwater is modeled in the general transient event tree, along with other mitigating systems. It is still acceptable to retain this system since the model includes all support system dependencies, although minimal benefit is expected due to the low fragility of the system.	No (Modeled as Mitigating Failure)
IE-LSP-GR Loss of Offsite Power (Grid) IE-LSP-PC Loss of Offsite Power (Plant-Centered) IE-LSP-SC Loss of Offsite Power (Switchyard) IE-LSP-WR Loss of Offsite Power (Weather)	Offsite power initiating events include LSP caused by grid-related failures beyond the licensee controlled area, severe weather, switchyard failures, and failures of power distribution from the switchyard to plant switchgear. The internal events PRA differentiates between LSP to a single unit and to both units, with the distinction related to crosstie capability. The SPRA LOOP initiator is mapped to the plant-centered LOOP event (IELSP- PC). Since the only modeling difference between these events is the offsite power recovery time, there is no functional difference in choosing this event over the others. Since the offsite power fragility is represented by a generic fragility, representing the system at large, it is only necessary to represent this by a single event in the SPRA.	Yes
IE-MFLB Main Feed Line Break IE-MSLBI Main Steam Line Break Inside Containment IE-MSLBO Main Steam Line Break Outside Containment	Main steam line break causes rapid RCS cooldown, reactor power increase due to reactivity feedback, RCS inventory shrinkage and ECCS actuation. The primary distinction between steam line breaks inside and outside Containment is the potential for rapid isolation by the Main Steam Isolation Valves. A Feed line break causes a less severe transient since the initial RCS cooldown rate is not as great, but unisolated feed line breaks eventually cause loss of secondary heat sink on the affected Steam Generator(s). The CNP PRA uses the same event tree to model feed line and steam line breaks (both inside and outside Containment). This event tree has been demonstrated to bound all three initiating events.	Yes

Table 5-1
Internal Initiating Events Review for SPRA

Internal Events Initiating Event	Seismic PRA Considerations	Modeled as an Initiating Event in SPRA?
IE-SGTR Steam Generator Tube Rupture	Seismic-induced failure of the Steam Generators (structural failure of its supports) is modeled as a direct core damage event due to the potential correlated failure on multiple generators. Given the integral nature of the Steam Generator tubes within the generator itself, it is unlikely that shaking of the generator could result in tube rupture without also affecting generator structural integrity. The SPRA models the more severe consequence and therefore SGTR is screened from further consideration.	No (Subsumed by Direct Core Damage Modeling)
IE-TRA Transient with Power Conversion	The transient event is considered the baseline initiating event if a loss of offsite power does not occur. The internal events transient event tree is reviewed in this calculation to ensure it captures all potential seismically induced consequential events not already captured by the SIET.	Yes
IE-VDC-A(-2) Loss of 250 VDC Train A IE-VDC-B(-2) Loss of 250 VDC Train B	The 250 VDC system as a mitigating system is on the SEL, and its fragility is characterized. The SPRA models seismic-induced loss of 250 VDC as a mitigating failure, which results in core damage due to correlated failure of both trains. Additionally, an earthquake severe enough to fail this SC-I system would broadly fail SC-I equipment resulting in core damage.	No (Modeled as Mitigating failure)
IE-VEF LOCA beyond ECCS Capability	A very severe earthquake could conceivably fail major NSSS components, resulting in RCS rupture within the extra-large break range. The SPRA uses the IE-VEF initiating event for failure of the reactor vessel.	Yes
IE-T11A/T11D /T21A/T21D Loss of 4-kV Safety Bus	This event involves loss of a single 4-kV safety bus, which would broadly fail safety related equipment on the affected train. This event does not cause a substantially different transient to occur, unless the other train fails, and is adequately addressed through system dependencies in the fault tree logic.	No (Modeled as mitigating failure)
IE-XFW	This event involves excessive feedwater which may over fill the Steam Generators and damage the Turbine Driven Auxiliary Feedwater Pump due to water intrusion in the steam lines. Excessive feedwater is not a credible seismic induced initiating event because it involves overfill of the Steam Generators prior to the start of the event.	No

5.1.2 - Review of Internal Events Event Trees

Based on review of the identified seismic initiating and consequential events, the following FPIE event trees were determined to not be required as part of the two-top development as shown in Table 5-2. Required modification to event trees for use in the seismic model are shown in Table 5-3. A new event tree was created to address VSLOCA. Documentation of the event tree review is located in section 5.1.2.2 and 5.2.2 of the SPRA Model Notebook [50].

Table 5-2
FPIE Event Trees Excluded from SPRA Model

Event Tree	Reason for Exclusion
ATWS	Failure to trip is assumed to lead directly to core damage.
CCW	Loss of CCW is modeled as a consequential failure in the SPRA.
DC	Loss of DC is modeled as consequential/mitigating system failure in the SPRA.
LOOP	Since all LOOPs are modeled to be dual unit, this single unit LOOP event tree is not needed for the SPRA.
DESL, SBOSL, SDGSL	These event trees are SBO event trees that occur after a single unit LOOP. Since all LOOPs are modeled to be dual unit, these single unit LOOP based event trees are not needed for the SPRA.
ESW	Loss of ESW is modeled as a consequential failure in the SPRA.
FLDC, FLDT	Flooding is modeled as a consequential failure in the SPRA.
SGTR	SGTR is currently modeled as direct core damage for the SPRA, and thus the event tree development is not needed.

Table 5-3
FPIE Event Trees Modified for SPRA

Event Tree	Reason for Modification
TRAN	Sequences with AFW and SDS success need to consider VSLOCA.
CCWT	Sequences with AFW and SDS success need to consider VSLOCA.
DLOOP	Sequences with AFW and SDS success need to consider VSLOCA.
MSLB	Sequences with AFW and SDS success need to consider VSLOCA; HPI mission time is only 30 minutes - long term inventory is required if a VSLOCA occurs.
SDGSD	Sequences with AFW and SDS success need to consider VSLOCA. Offsite power recovery is not credited in the SPRA.
SDGDD	Sequences with AFW and SDS success need to consider VSLOCA. Offsite power recovery is not credited in the SPRA.
SBOSD	Sequences with AFW and SDS success need to consider VSLOCA. Offsite power recovery is not credited in the SPRA.
SBODD	Sequences with AFW and SDS success need to consider VSLOCA. Offsite power recovery is not credited in the SPRA.

5.1.3 Review of Secondary Hazards

Seismically induced internal fire and flooding, and other secondary hazards were reviewed although not required by the SPID [2]. CNP has an NFPA 805 Fire PRA that analyzes CDF and LERF from internal fires, and an Internal Flooding PRA (as part of the internal events PRA). A systematic screening process was applied to these PRAs to determine internal fire and flooding seismic scenarios. This screening approach followed EPRI 300201980 [57]. It should be noted that CNP was a pilot plant in development of this EPRI guidance. Appendix "A" of the SPRA Model Notebook [50] documents the screening process.

Other secondary external hazards were considered via review of the CNP IPEEE [49]. The CNP IPEEE examined the following external events:

- Seismic Events
- External Flooding
- Aircraft Accidents
- Severe Winds (strong winds and tornadoes)
- Ship Impact Accidents
- Off-Site Hazardous Material Accidents
- On-Site Hazardous Material Accidents
- Turbine Missiles
- External Fires

Of the listed external events, only external flooding and fires can realistically be caused by seismic events. Review of secondary seismic hazards was performed against the list of external hazards in non-mandatory Appendix 6-A of the PRA Standard. The remaining secondary hazard events were determined to be non-threating to CNP from seismic events, as documented in Section 5.1.3 of the SPRA Model Notebook [50].

5.1.4 Seismic Initiating Event Tree

The SIET models the potential seismic initiators that may result from an earthquake. Seismic events may give rise to one or more potential accident sequence. The purpose of the SIET is to transfer seismic accident sequences to the appropriate event tree to model the plant response. The nature of event tree analysis and the use of the minimum cutset approach in PRA models make it difficult to capture the impact of multiple concurrent initiating events. The initiators for the SIET are just the seismic event, with other events being treated as consequential failures. The seismic event was broken into eight hazard intervals (bins) and each was quantified independently. Summaries of the SIET, direct core damage, and direct large early release mapping are provided below. Detailed descriptions of these and other mapping can be found in Section 5.2.3 of the SPRA Model Notebook [50].

5.1.4.1 Seismic Initiating Event (S-INIT)

This top event models the seismic initiating events as defined in the SPRA Model database. This event encompasses the range of potential seismic events based on the hazard curve for CNP.

5.1.4.2 Direct Core Damage (D-CD)

This top event models seismic-induced failures that are mapped directly to core damage, and include the following failures:

- Seismic-Induced Structural Failure of the Screenhouse
- Seismic induced failure of the Main Control Room Control Boards
- Seismic-Induced LOCA Beyond ECCS Capability
- Seismic-Induced Structural Failure of the Reactor Coolant Pumps
- Seismic-Induced Structural Failure of the Reactor Coolant System Piping
- Seismic-Induced Structural Failure of the Reactor Coolant System Accumulators
- Seismic-Induced Failure of the Polar Crane

5.1.4.3 Direct Large Early Release (D-LER)

This top event models seismic-induced failures that are mapped to direct large early release, and include the following failures:

- Seismic-Induced Structural Failure of the Steam Generators
- Seismic-Induced Structural Failure of Containment
- Seismic-Induced Structural Failure of the Auxiliary Building

5.1.5 High Frequency Evaluation

Systematic screening of components susceptible to chatter was performed, which identified risk susceptible components as relays and contactor type devices. Review focused on chatter susceptible components associated with each SEL component. The assessment analyzed the potential for seismic-induced chatter to affect a component's modeled function to either mitigate or contribute to an initiating event. A number of screenings were performed to manage the large number of relays and contacts present in the plant. Quantification of the SPRA then identified relays and relay groups that are risk significant. A detailed chatter evaluation was then performed on the significant devices. Relays whose chatter was not detrimental were screened and eliminated from the model. Relays whose chatter may be detrimental remained in the SPRA logic model. The CNP Relay Chatter Analysis documents the relay chatter evaluation, and the incorporation of this evaluation into the SPRA is documented in the SEL Notebook [8].

5.1.6 Correlation

Seismic failure of multiple components sharing the same fragility was assumed to be fully correlated. Correlation was also assumed between analogous components between units, i.e., the Unit 1 and Unit 2 CCW pumps were assumed correlated. The CNP SPRA seeks to reduce conservatism associated with assuming full correlation by minimizing the number of fragility groups occurring in significant accident sequences. Fragility groups were generally subdivided and refined until the point they no longer are significant or cannot be further subdivided. Part of this process was breaking correlation for components of differing design, location, and seismic demand. Assumption 4.1 of the SPRA Model Notebook [50] addresses correlation. Fragility correlation is discussed in Section 4.4.2.3 of this report.

5.1.7 Modeling Containment Performance

The Seismic PRA considered two seismically induced failure mechanisms that might result in a loss of Containment of integrity. Phenomenological failures of containment integrity that are common to the internal events model (e.g., hydrogen combustion) were also retained in the model.

The first seismically induced mechanism involves a gross failure of the Containment pressure boundary due to seismic events. Potential failure modes include failure of the containment structure itself or failure of containment penetrations. These failure modes were assessed during the fragility evaluation of containment and were assumed to progress directly to core damage and large early release if they occur.

The second mechanism involves the failure of the Containment to become isolated and failure to maintain isolation following a seismic event. Potential failure modes include mechanical failure of isolation valves and control circuity failures affecting isolation valves. These failure types have been addressed by the assignment of fragility identifiers to all components/basic events that can be impacted by a seismic event.

Additionally a MAAP analysis for hydrogen detonation inside Containment during seismically induced events was performed which determined containment failure probabilities for different accident progression sequences. This analysis is documented in the CNP Containment Failure Assessment [51]. Containment over pressure failure was modeled on this basis.

5.1.8 Human Reliability Analysis

The seismic HRA uses a conservative screening process in accordance with EPRI HRA guidance [52], in combination with detailed assessment for risk significant HFEs. Development of the seismic HRA is documented in the CNP Human Reliability notebook [47].

5.2 SPRA Plant Seismic Logic Model Technical Adequacy

The CNP SPRA seismic plant response methodology and analysis were subjected to an independent peer review against the pertinent requirements of the PRA Standard [4]. The peer review assessment, and subsequent disposition of peer review findings, is described in Appendix A of this report, and establishes that the CNP SPRA seismic plant response analysis is suitable for this SPRA application.

5.3 Seismic Risk Quantification

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

5.3.1 SPRA Quantification Methodology

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

The EPRI FRANX software code was used to discretize the seismic hazard into the 8 seismic initiators, and create a two-top integrated fault tree. The EPRI CAFTA and PRAQuant software, along with the solver software FTREX, solve the fault tree to produce cutsets for CDF and LERF. The EPRI ACUBE code was then utilized to estimate the CDF/LERF more accurately by calculating the total frequency on the entire set of SCDF/SLERF cutsets. The ACUBE code does not use the rare events approximation as is utilized in the CAFTA software's minimum cut upper bound estimation calculation. Therefore, the ACUBE software provides a more accurate solution. The following steps were used to develop and perform the Seismic PRA model quantification for both CDF and LERF:

1) Perform initial quantification with initial fragility and HEP values and generally assuming complete seismic correlation within fragility subgroups
2) Identify fragilities and HEPs to be refined
3) Refine fragility groups to improve seismic correlation modeling
4) Identify final set of fragilities to be inserted into the model (because of model size limitations and software constraints)
5) Perform HEP dependency analysis
6) Perform truncation sensitivity to determine final truncation level
7) Solve and assemble bin cutsets into combined cutset files (one for CDF and one for LERF)
8) Finalize quantification of CDF and LERF (ACUBE analysis)
9) Evaluate basic event importance (ACUBE analysis supplemented by selected sensitivity analyses)
10) Perform uncertainty analysis (UNCERT)
11) Evaluate sensitivity cases
12) Repeat selected steps above as needed for optimization of results

5.3.2 SPRA Model and Quantification Assumptions and Approaches

The major assumptions and approaches used in the CNP SPRA modeling and quantification are as follows:

5.3.2.1 Seismic Initiating Event Tree

Development of the SIET involved a ranking of seismic initiating events from greatest to least in terms of potential risk significance, with the purpose of ensuring each interval was assigned to the most challenging initiating event that could be credibly induced by that ground motion level. This ranking involved judgment and was a source of epistemic uncertainty

5.3.2.2 Quantification Process for LERF

For the baseline LERF calculations, a conservative assumption was made that fragility events with a probability greater than 0.9 were assumed as failure (i.e., set to TRUE). This allowed a deeper level of quantification than can be achieved normally.

5.3.2.3 Quantification by ACUBE

In an exact solution of CDF or LERF, the total risk for a group of cutsets would be calculated as an "exclusive OR" summation where the union (overlap) between cutsets is excluded from the summation. This however is computationally infeasible for large cutset files. To avoid this computational burden, CAFTA approximates the total as one minus the summation of the compliments (successes) of each cutset value. This approximation is reasonable when cutset conditional probabilities are small, however it causes significant over-estimation when event probabilities are relatively large, as they can be in seismic cutsets. Use of the ACUBE software afforded a more precise solution, but potentially only to a top batch of cutsets, with the balance treated with the more conservative approach.

5.3.2.4 Limitations of ACUBE Importance Measures

ACUBE is capable of producing traditional PRA importance measures such as F-V, RAW, and Birnbaum during the quantification process. These importance measures are reported throughout this document for risk ranking and results analysis purposes. As ground motion levels increase, these importance measures become less reliable due to the "flat" nature of SPRA cutsets. This refers to the large numbers of cutsets created at higher ground motion levels with fragility events very near a probability of 1.0, which results in cutsets with very similar values. The use of complement events and success branch delete term fault tree structures results in the risk importance measures for higher bins being weighted in favor of the more severe initiating events, such as direct core damage or large early release events. For this reason, the F-V metric over-represents the real risk decrease that would be seen by improving the fragility event for the given components for higher ground motions.

Another limitation of the ACUBE importance measures occurs when some, but not all of the cutsets are able to be calculated more precisely by ACUBE. This results in over estimation of the risk importance of some components with significant cutsets below the limit that can be calculated by ACUBE. This limitation was minimized by applying ACUBE to each ground motion interval individually, which allows a larger number of cutsets to be calculated precisely.

5.3.2.5 Correlation

Seismic failure of multiple components sharing the same fragility was assumed to be fully correlated. See Section 5.1.6 for detailed discussion of the approach to correlation.

5.3.2.6 Building Failure

The fragility for SC-I buildings was mapped directly to core damage and large early release (if appropriate for the building). This was conservative because the fragilities more represent the onset of structural failure, as opposed to catastrophic collapse. This conservatism is visible, but not overly dominant, in the significant accident sequences, especially for LERF.

5.3.2.7 NSSS Component Support Failures

Seismic failure of structural supports for the Steam Generators and reactor coolant pumps were assumed to cause unrestrained motion and subsequent RCS rupture. This was conservative because the motion would likely be impeded by other structural elements, and pipe ductility may accommodate motion before rupturing.

5.3.2.8 Large LOCA

Seismic failure of the pressurizer supports was conservatively assumed to cause unrestrained motion of the pressurizer and subsequent failure of the surge line. This was a conservative approach because the pressurizer motion would be impeded by other structures in the Containment, and pipe ductility may accommodate pressurizer motion before generating a full line break. This modeling conservatism may have artificially increased the importance of seismic-induced large LOCA, which was assumed for a surge line rupture.

5.3.2.9 VSLOCA – CVCS Cross-tie Mitigation

VSLOCAs were assumed to be within the CVCS crosstie's makeup capability. VSLOCAs for CNP were assumed to be just under the threshold of small LOCA. Small LOCAs for CNP are in the range of 3/8 in. to 2 in. break sizes and VSLOCA is assumed to be a break size less than 3/8 in. The CVCS crosstie is assumed to be capable of at least 109 gpm of makeup (based on system notebook information). This makeup has been shown to be sufficient to prevent core damage event for an initial 182 gpm/RCP seal LOCA (i.e., a total of 728 gpm for all four RCPs).

5.3.2.10 VSLOCA SI Pump Mitigation

It was assumed that one SI pump was sufficient to provide inventory makeup for a VSLOCA if AFW is available, and the RCS pressure is below 1550 psi. This assumption was based upon FPIE Success Criteria Notebook information [27].

5.3.2.11 Rule-of-the-Box

In cases where components are housed within or otherwise integral to a larger component, a limiting fragility was sometimes developed to represent all of the elements associated with the larger component. For example, the Steam Generator tube bundles were considered rule-of-the-box with the overall Steam Generator itself.

Rule-of-the-box details and SEL items enveloped by rule-of-the-box criteria can be found in Section 5.1 and the Attachments of CNP Fragility Analysis Plan [30].

5.4 SCDF Results

The seismic PRA performed for CNP shows that the point estimate seismic CDF is 2.44E-05 for Unit 1 and 2.38E-05 for Unit 2. 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 of this report. Important contributors are discussed in the following paragraphs.

The top 10 SCDF accident cutsets are documented in the SPRA quantification notebook [12]. These are summarized in Table 5.4-1 of this report.

Unit 1 SSCs with the most significant seismic failure contributions to SCDF are listed in Table 5.4-2 of this report. Risk significant contributors are sorted by F-V importance. The seismic fragilities for each of the significant contributors is also provided in Table 5.4-2, along with the corresponding limiting seismic failure mode and method of fragility calculation.

Note for the tables that follow, Unit 2 results are similar. While there may be small numerical difference between the units, risk insights from Unit 1 are applicable to Unit 2.

Significant CDF and LERF seismic risk contributors for the CNP SPRA were determined by F-V. I&M developed the threshold for risk-significance based upon Regulatory Guide 1.174 [59] risk thresholds rather than a direct F-V percentage of 0.005 risk contribution of the SPRA results. Risk-significance is defined in the Quantification Notebook [12] as follows:

> For the D.C. Cook Seismic PRA, "risk-significant basic events" are defined in a manner similar to that provided in the PRA Standard (i.e., F-V importance greater than 0.005). A RAW importance value is not appropriate for events such as

seismic fragilities, so a RAW definition is not used. However, because the Seismic PRA may contain many conservative values for parameters such as seismic fragilities that are refined as necessary, the definition of "risk-significant" is tied to a static value rather than a ratio value such as F-V. Therefore, "risk-significant basic events" are defined as those having a F-V of 0.005 when calculated as a fraction of the overall maximum expected plant risk defined as CDF of 1E-4 and LERF of 1E-5, consistent with guideline values in Regulatory Guide 1.174. That is, "risk-significant basic events" are those that contribute more than 5E-7 to CDF or 5E-8 to LERF.

Table 5.4-1
Summary of Top SCDF Accident Cutsets

#	Cutset Probability	BE Probability	Input	Description
1	7.9116E-07	2.6936E-06	%G08	Seismic Initiating Event (>0.8g)
		9.1000E-01	MODE1	PLANT OPERATING IN MODE 1
		3.1844E-01	SF-SCIB-AB-C-%G08	SEISMIC FRAGILITY FOR %G08: SC-I Building - Auxiliary Building
		5.0000E-01	SF-SCIB-C-%G08	SEISMIC FRAGILITY FOR %G08: SC-I Building
		5.2843E-02	SF-SCIB-CONTC-%G08	SEISMIC FRAGILITY FOR %G08: SC-I Building - Containment Building
		1.1984E-05	SF-SG-C-%G08	SEISMIC FRAGILITY FOR %G08: Steam Generator (Room 60)
2	7.8053E-07	2.6936E-06	%G08	Seismic Initiating Event (>0.8g)
		9.1000E-0	MODE 1	PLANT OPERATING IN MODE 1
		3.1844E-01	SF-SCIB-AB-C-%G08	SEISMIC FRAGILITY FOR %G08: SC-I Building - Auxiliary Building
Cutsets 1 and 2 are direct-to-CDF events due to failure of Seismic Category I buildings at the highest seismic bin G08 (the Screenhouse for cutset 1 and the Auxiliary Building for cutset 2).				
3	7.1261E-07	3.8842E-06	%G05	Seismic Initiating Event (0.5g to <0.6g)
		1.0000E+00	0-SEQ-MSLB-005	SEQUENCE MSLB-005
		9.1000E-01	MODE1	PLANT OPERATING IN MODE 1
		2.2377E-	SF-ACCUM-C-%G05	SEISMIC FRAGILITY FOR %G05: Accumulator (Room 68)
		1.5463E-03	SF-CP-C-%G05	SEISMIC FRAGILITY FOR %G05: Main Control Room Boards (Correlated Failure)
		1.1555E-02	SF-MLOCA-C-%G05	SEISMIC FRAGILITY FOR %G05: Medium LOCA
		2.7049E-01	SF-NSCIB-TB1-C-G05	SEISMIC FRAGILITY FOR %G05: Turbine Building Collapse
		2.6309E-03	SF-POLARCRNC-%G05	SEISMIC FRAGILITY FOR %G05: Polar Crane Fragility

		7.7917E-04	SF-PZR-C-%G05	SEISMIC FRAGILITY FOR %G05: Pressurizer Support
		5.7262E-07	SF-RCP-C-%G05	SEISMIC FRAGILITY FOR %G05: Reactor Coolant Pump (Room 60)
		2.7247E-05	SF-RV-C-%G05	SEISMIC FRAGILITY FOR %G05: Reactor Vessel (Room 105)
		3.4993E-02	SF-SCIB-AB-C-%G05	SEISMIC FRAGILITY FOR %G05: SC-I Building - Auxiliary Building
		9.0135E-02	SF-SCIB-C-%G05	SEISMIC FRAGILITY FOR %G05: SC-I Building
		1.5492E-03	SF-SCIB-CONTC-% G05	SEISMIC FRAGILITY FOR %G05: SC-I Building - Containment Building
		1.6607E-03	-SF-SCIC-CONT1-C-%G05	SEISMIC FRAGILITY FOR %G05: SC-I Components in Containment up to EL. 625'.
		1.2767E-07	F-SG-C-%G05	SEISMIC FRAGILITY FOR %G05: Steam Generator (Room 60)
		1.1432E-01	SF-SLOCA-C-%G05	SEISMIC FRAGILITY FOR %G05: Small LOCA
4	7.0084E-07	7.2400E-06	%G04	Seismic Initiating Event (0.4g to 0.5g)
		1.0000E+00	0-SEQ-MSLB-005	SEQUENCE MSLB-005
		9.1000E-01	MODE1	PLANT OPERATING IN MODE 1
		6.9858E-03	SF-ACCUM-C-%G04	SEISMIC FRAGILITY FOR %G04: Accumulator (Room 68)
		2.6529E-04	SF-CP-C-%G04	SEISMIC FRAGILITY FOR %G04: Main Control Room Boards (Correlated Failure)
		4.3045E-03	SF-MLOCA-C-%G04	SEISMIC FRAGILITY FOR %G04: Medium LOCA
		1.1816E-01	SF-NSCIB-TB1-C-G04	SEISMIC FRAGILITY FOR %G04: Turbine Building Collapse
		5.9436E-04	SF-POLARCRNC-%G04	SEISMIC FRAGILITY FOR %G04: Polar Crane Fragility
		1.8469E-04	SF-PZR-C-%G04	SEISMIC FRAGILITY FOR %G04: Pressurizer Support
		7.1143E-08	SF-RCP-C-%G04	SEISMIC FRAGILITY FOR %G04: Reactor Coolant Pump (Room 60)
		3.6214E-06	SF-RV-C-%G04	SEISMIC FRAGILITY FOR %G04: Reactor Vessel (Room 105)
		8.5414E-03	SF-SCIB-AB-C-%G04	SEISMIC FRAGILITY FOR %G04: SC-I Building - Auxiliary Building
		2.7884E-02	SF-SCIB-C-%G04	SEISMIC FRAGILITY FOR %G04: SC-I Building
		2.0725E-04	SF-SCIB-CONTC-% G04	SEISMIC FRAGILITY FOR %G04: SC-I Building - Containment Building

		3.5290E-04	SF-SCIC-CONT1-C-%G04	SEISMIC FRAGILITY FOR %G04: SC-I Components in Containment up to EL. 625'.
		1.4192E-08	SF-SG-C-%G04	SEISMIC FRAGILITY FOR %G04: Steam Generator (Room 60)
		5.3775E-02	SF-SLOCA-C-%G04	SEISMIC FRAGILITY FOR %G04: Small LOCA
Cutsets 3 and 4 are seismically-induced Main Steam Line Break due to Turbine Building collapse, at G05 and G04. Auxiliary Feedwater and Feed-and-Bleed fail due to associated failures caused by the Turbine Building collapse.				
5	6.6354E-07	2.1754E-06	%G06	Seismic Initiating Event (0.6g to <0.7g)
		1.0000E+00	0-SEQ-DLOOP-001	SEQUENCE DLOOP-001
		1.0000E+00	0-SEQ-VSLOC-004	SEQUENCE VSLOC-004
		9.1000E-01	MODE1	PLANT OPERATING IN MODE 1
		5.1269E-02	SF-ACCUM-C-%G06	SEISMIC FRAGILITY FOR %G06: Accumulator (Room 68)
		5.5726E-03	SF-CP-C-%G06	SEISMIC FRAGILITY FOR %G06: Main Control Room Boards (Correlated Failure)
		9.2095E-01	SF-LSP-C-%G06	SEISMIC FRAGILITY FOR %G06: Offsite Power
		2.4047E-02	SF-MLOCA-C-%G06	SEISMIC FRAGILITY FOR %G06:Medium LOCA
		7.8400E-03	SF-POLARCRNC-%G06	SEISMIC FRAGILITY FOR %G06: Polar Crane Fragility
		2.3001E-03	SF-PZR-C-%G06	SEISMIC FRAGILITY FOR %G06: Pressurizer Support
		2.8786E-06	SF-RCP-C-%G06	SEISMIC FRAGILITY FOR %G06: Reactor Coolant Pump (Room 60)
		1.2550E-04	SF-RV-C-%G06	SEISMIC FRAGILITY FOR %G06: Reactor Vessel (Room 105)
		9.0678E-02	SF-SCIB-AB-C-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Building - Auxiliary Building
		1.9365E-01	SF-SCIB-C-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Building
		6.5251E-03	SF-SCIB-CONTC-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Building - Containment Building
		5.2031E-03	SF-SCIC-CONT1-C-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Components in Containment up to EL. 625'.
		5.8703E-02	SF-SCIV-AB1-C-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Valves in Auxiliary Building - EL. 573-591
		7.0361E-07	SF-SG-C-%G06	SEISMIC FRAGILITY FOR %G06: Steam Generator (Room 60)
		1.9285E-01	SF-SLOCA-C-%G06	SEISMIC FRAGILITY FOR %G06: Small LOCA

		7.2539E-01	SF-VSLOCA-C-%G06	SEISMIC FRAGILITY FOR %G06: Seismic-Induced Very Small LOCA
Cutset 5 is a seismically-induced loss of offsite power with a seismically-induced VSLOCA. The CVCS crosstie is failed by its operator action set to 1.0 due to the high seismic bin. Operator actions for high-pressure recirculation are also set to 1.0 due to the high seismic bin.				
6	4.9520E-07	2.1754E-06	%G06	Seismic Initiating Event (0.6g to 0.7g)
		1.0000E+00	0-SEQ-DLOOP-005	SEQUENCE DLOOP-005
		9.1000E-01	MODE1	PLANT OPERATING IN MODE 1
		5.5357E-01	RELAY_B_2_U1-C-%G06	SEISMIC FRAGILITY FOR %G06: Fragility Group Relay_B_2_U1
		5.1269E-02	SF-ACCUM-C-%G06	SEISMIC FRAGILITY FOR %G06: Accumulator (Room 68)
		5.5726E-03	SF-CP-C-%G06	SEISMIC FRAGILITY FOR %G06: Main Control Room Boards (Correlated Failure)
		9.9137E-01	SF-FLD-1N3M-C-%G06	SEISMIC FRAGILITY FOR %G06: Seismic-Induced Flood from Unit 1 Group N3
		9.2095E-01	SF-LSP-C-%G06	SEISMIC FRAGILITY FOR %G06: Offsite Power
		2.4047E-02	SF-MLOCA-C-%G06	SEISMIC FRAGILITY FOR %G06: Medium LOCA
		7.8400E-03	SF-POLARCRNC-%G06	SEISMIC FRAGILITY FOR %G06: Polar Crane Fragility
		2.3001E-03	SF-PZR-C-%G06	SEISMIC FRAGILITY FOR %G06: Pressurizer Support
		2.8786E-06	SF-RCP-C-%G06	SEISMIC FRAGILITY FOR %G06: Reactor Coolant Pump (Room 60)
		1.2550E-04	SF-RV-C-%G06	SEISMIC FRAGILITY FOR %G06: Reactor Vessel (Room 105)
		9.0678E-02	SF-SCIB-AB-C-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Building - Auxiliary Building
		1.9365E-01	SF-SCIB-C-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Building
		6.5251E-03	SF-SCIB-CONTC-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Building - Containment Building
		5.2031E-03	SF-SCIC-CONT1-C-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Components in Containment up to EL. 625'.
		5.8703E-02	SF-SCIV-AB1-C-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Valves in Auxiliary Building - EL. 573-591
		7.0361E-07	SF-SG-C-%G06	SEISMIC FRAGILITY FOR %G06: Steam Generator (Room 60)

		1.9285E-01	SF-SLOCA-C-%G06	SEISMIC FRAGILITY FOR %G06: Small LOCA
		9.8645E-01	SF-TFP-PNLS-C-%G06	SEISMIC FRAGILITY FOR %G06: 250 VDC Distribution Panel (Unit 1)
7	4.9520E-07	2.1754E-06	%G06	Seismic Initiating Event (0.6g to <0.7g)
		1.0000E+00	0-SEQ-DLOOP-005	SEQUENCE DLOOP-005
		9.1000E-01	MODE1	PLANT OPERATING IN MODE 1
		5.5357E-01	RELAY_B_2_U1-C-%G06	SEISMIC FRAGILITY FOR %G06: Fragility Group Relay_B_2_U1
		5.1269E-02	SF-ACCUM-C-%G06	SEISMIC FRAGILITY FOR %G06: Accumulator (Room 68)
		5.5726E-03	SF-CP-C-%G06	SEISMIC FRAGILITY FOR %G06: Main Control Room Boards (Correlated Failure)
		9.9137E-01	SF-FLD-1N4M-C-%G06	SEISMIC FRAGILITY FOR %G06: Seismic-Induced Flood from Unit 1 Group N4
		9.2095E-01	SF-LSP-C-%G06	SEISMIC FRAGILITY FOR %G06: Offsite Power
		2.4047E-02	SF-MLOCA-C-%G06	SEISMIC FRAGILITY FOR %G06: Medium LOCA
		7.8400E-03	SF-POLARCRNC-%G06	SEISMIC FRAGILITY FOR %G06: Polar Crane Fragility
		2.3001E-03	SF-PZR-C-%G06	SEISMIC FRAGILITY FOR %G06 Pressurizer Support
		2.8786E-06	SF-RCP-C-%G06	SEISMIC FRAGILITY FOR %G06: Reactor Coolant Pump (Room 60)
		1.2550E-04	SF-RV-C-%G06	SEISMIC FRAGILITY FOR %G06: Reactor Vessel (Room 105)
		9.0678E-02	SF-SCIB-AB-C-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Building - Auxiliary Building
		1.9365E-01	SF-SCIB-C-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Building
		6.5251E-03	-SF-SCIB-CONTC-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Building - Containment Building
		5.2031E-03	SF-SCIC-CONT1-C-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Components in Containment up to EL. 625'
		5.8703E-02	SF-SCIV-AB1-C-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Valves in Auxiliary Building - EL. 573-591
		7.0361E-07	SF-SG-C-%G06	SEISMIC FRAGILITY FOR %G06: Steam Generator (Room 60)
		1.9285E-01	SF-SLOCA-C-%G06	SEISMIC FRAGILITY FOR %G06: Small LOCA

		9.8645E-01	SF-TFP-PNLS-C-%G06	SEISMIC FRAGILITY FOR %G06: 250 VDC Distribution Panel (Unit 1)
8	4.9520E-07	2.1754E-06	%G06	Seismic Initiating Event (0.6g to <0.7g)
		1.0000E+00	0-SEQ-DLOOP-005	SEQUENCE DLOOP-005
		9.1000E-01	MODE1	PLANT OPERATING IN MODE 1
		5.5357E-01	RELAY_B_2_U1-C-%G06	SEISMIC FRAGILITY FOR %G06: Fragility Group Relay_B_2_U1
		5.1269E-02	SF-ACCUM-C-%G06	SEISMIC FRAGILITY FOR %G06: Accumulator (Room 68)
		5.5726E-03	SF-CP-C-%G06	SEISMIC FRAGILITY FOR %G06: Main Control Room Boards (Correlated Failure)
		9.9137E-01	SF-FLD-1N5M-C-%G06	SEISMIC FRAGILITY FOR %G06: Seismic-Induced Flood from Unit 1 Group N5
		9.2095E-01	SF-LSP-C-%G06	SEISMIC FRAGILITY FOR %G06: Offsite Power
		2.4047E-02	SF-MLOCA-C-%G06	SEISMIC FRAGILITY FOR %G06: Medium LOCA
		7.8400E-03	SF-POLARCRNC-%G06	SEISMIC FRAGILITY FOR %G06: Polar Crane Fragility
		2.3001E-03	SF-PZR-C-%G06	SEISMIC FRAGILITY FOR %G06: Pressurizer Support
		2.8786E-06	SF-RCP-C-%G06	SEISMIC FRAGILITY FOR %G06: Reactor Coolant Pump (Room 60)
		1.2550E-04	SF-RV-C-%G06	SEISMIC FRAGILITY FOR %G06: Reactor Vessel (Room 105)
		9.0678E-02	F-SCIB-AB-C-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Building - Auxiliary Building
		1.9365E-01	SF-SCIB-C-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Building
		6.5251E-03	SF-SCIB-CONTC-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Building - Containment Building
		5.2031E-03	SF-SCIC-CONT1-C-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Components in Containment up to EL. 625'.
		5.8703E-02	SF-SCIV-AB1-C-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Valves in Auxiliary Building - EL. 573-591
		7.0361E-07	SF-SG-C-%G06	SEISMIC FRAGILITY FOR %G06: Steam Generator (Room 60)
		1.9285E-01	SF-SLOCA-C-%G06	SEISMIC FRAGILITY FOR %G06: Small LOCA

		9.8645E-01	SF-TFP-PNLS-C-%G06	SEISMIC FRAGILITY FOR %G06: 250 VDC Distribution Panel (Unit 1)
9	4.9520E-07	2.1754E-06	%G06	Seismic Initiating Event (0.6g to <0.7g)
		1.0000E+00	0-SEQ-DLOOP-005	SEQUENCE DLOOP-005
		9.1000E-01	MODE1	OPERATING IN MODE 1
		5.5357E-01	RELAY_B_2_U1-C-%G06	SEISMIC FRAGILITY FOR %G06: Fragility Group Relay_B_2_U1
		5.1269E-02	SF-ACCUM-C-%G06	SEISMIC FRAGILITY FOR %G06: Accumulator (Room 68)
		5.5726E-03	SF-CP-C-%G06	SEISMIC FRAGILITY FOR %G06: Main Control Room Boards (Correlated Failure)
		9.9137E-01	SF-FLD-2N3M-C-%G06	SEISMIC FRAGILITY FOR %G06: Seismic-Induced Flood from Unit 2 Group N3
		9.2095E-01	SF-LSP-C-%G06	SEISMIC FRAGILITY FOR %G06: Offsite Power
		2.4047E-02	SF-MLOCA-C-%G06	SEISMIC FRAGILITY FOR %G06: Medium LOCA
		7.8400E-03	SF-POLARCRNC-%G06	SEISMIC FRAGILITY FOR %G06: Polar Crane Fragility
		2.3001E-03	SF-PZR-C-%G06	SEISMIC FRAGILITY FOR %G06: Pressurizer Support
		2.8786E-06	SF-RCP-C-%G06	SEISMIC FRAGILITY FOR %G06: Reactor Coolant Pump (Room 60)
		1.2550E-04	SF-RV-C-%G06	SEISMIC FRAGILITY FOR %G06: Reactor Vessel (Room 105)
		9.0678E-02	SF-SCIB-AB-C-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Building - Auxiliary Building
		1.9365E-01	SF-SCIB-C-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Building
		6.5251E-03	SF-SCIB-CONTC-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Building - Containment Building
		5.2031E-03	SF-SCIC-CONT1-C-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Components in Containment up to EL. 625'.
		5.8703E-02	SF-SCIV-AB1-C-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Valves in Auxiliary Building - EL. 573-591
		7.0361E-07	SF-SG-C-%G06	SEISMIC FRAGILITY FOR %G06: Steam Generator (Room 60)
		1.9285E-01	SF-SLOCA-C-%G06	SEISMIC FRAGILITY FOR %G06: Small LOCA
		9.8645E-01	SF-TFP-PNLS-C-%G06	SEISMIC FRAGILITY FOR %G06: 250 VDC Distribution Panel (Unit 1)

10	4.9520E-07	2.1754E-06	%G06	Seismic Initiating Event (0.6g to <0.7g)
		1.0000E+00	0-SEQ-DLOOP-005	SEQUENCE DLOOP-005
		9.1000E-01	MODE1	PLANT OPERATING IN MODE 1
		5.5357E-01	RELAY_B_2_U1-C-%G06	SEISMIC FRAGILITY FOR %G06: Fragility Group Relay_B_2_U1
		5.1269E-02	SF-ACCUM-C-%G06	SEISMIC FRAGILITY FOR %G06: Accumulator (Room 68)
		5.5726E-03	SF-CP-C-%G06	SEISMIC FRAGILITY FOR %G06: Main Control Room Boards (Correlated Failure)
		9.9137E-01	SF-FLD-2N4M-C-%G06	SEISMIC FRAGILITY FOR %G06: Seismic-Induced Flood from Unit 2 Group N4
		9.2095E-01	SF-LSP-C-%G06	SEISMIC FRAGILITY FOR %G06: Offsite Power
		2.4047E-02	SF-MLOCA-C-%G06	SEISMIC FRAGILITY FOR %G06: Medium LOCA
		7.8400E-03	SF-POLARCRNC-%G06	SEISMIC FRAGILITY FOR %G06: Polar Crane Fragility
		2.3001E-03	SF-PZR-C-%G06	SEISMIC FRAGILITY FOR %G06: Pressurizer Support
		2.8786E-06	SF-RCP-C-%G06	SEISMIC FRAGILITY FOR %G06: Reactor Coolant Pump (Room 60)
		1.2550E-04	SF-RV-C-%G06	SEISMIC FRAGILITY FOR %G06: Reactor Vessel (Room 105)
		9.0678E-02	SF-SCIB-AB-C-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Building - Auxiliary Building
		1.9365E-01	SF-SCIB-C-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Building
		6.5251E-03	SF-SCIB-CONTC-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Building - Containment Building
		5.2031E-03	SF-SCIC-CONT1-C-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Components in Containment up to EL. 625'.
		5.8703E-02	SF-SCIV-AB1-C-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Valves in Auxiliary Building - EL. 573-591
		7.0361E-07	SF-SG-C-%G06	SEISMIC FRAGILITY FOR %G06: Steam Generator (Room 60)
		1.9285E-01	SF-SLOCA-C-%G06	SEISMIC FRAGILITY FOR %G06: Small LOCA
		9.8645E-01	SF-TFP-PNLS-C-%G06	SEISMIC FRAGILITY FOR %G06: 250 VDC Distribution Panel (Unit 1)

Cutsets 6 through 10 are seismically-induced loss of offsite power sequences, together with a seismically-induced relay chatter event (RELAY_B_2_U1), a seismically-induced failure of a 250 VDC Distribution Panel (SF-TFPPNLS), and a seismically-induced flood (that varies between each cutset). The relay chatter event fails the automatic start of the motor-driven AFW pump. The 250 volt DC panel failure fails the automatic start of the turbine-driven AFW pump. Operator actions to start either pump, or to crosstie to the other unit, are set to 1.0 due to the high seismic bins. Feed and bleed fail due to operator failures to "restore control air W/ PAC after LOOP" and failure of "primary bleed and feed without SI actuated", both of which are also set to 1.0 failure due to the high seismic bin.

Table 5.4-2
SCDF Importance Measures Ranked by F-V

Event	Description	F-V	CDF Contribution	Median Capacity (g)	βr	βu	Failure Mode	Fragility Method
SF-LSP	Seismic Loss of Off-Site Power	0.142	3.47E-06	0.3	0.24	0.49	Functional	Generic 15C4313-RPT-007
SF-VSLOCA	Seismic Induced Very Small LOCA	0.054	1.31E-06	0.507	0.24	0.32	Functional	Judgement 15C4313-RPT-007
RELAY_D_1	Seismic Failure of Relay Group D_1	0.042	1.02E-06	0.470	0.27	0.49	Relay Chatter	SOV 15C4313-CAL-021
RELAY_D_2	Seismic Failure of Relay Group D_2	0.035	8.46E-07	0.470	0.27	0.49	Relay Chatter	SOV 15C4313-CAL-021
SF-A11-PNLS	Failure of Panel A11 due to Control Room Ceiling Failure	0.027	6.48E-07	0.740	0.24	0.38	Anchorage	Refined CDFM 15C4313-Cal-024
SF-BATCD	Seismic Failure of Train CD Battery Rack	0.023	5.57E-07	0.689	0.24	0.38	Anchorage	Refined CDFM 15C4313-Cal-026
SF-SDG	Seismic Failure of Supplemental Diesel Generator Components	0.021	5.22E-07	0.503	0.24	0.49	Functional	Judgement 15c4313-RPT-007
SF-NSCIB-TB1	Turbine Building Collapse	0.020	4.85E-07	0.677	0.24	0.26	Structural Collapse	Refined CDFM 15C4313-Cal-020

Table 5.4-3 lists the most significant non-seismic SSC random failures for Unit 1. These events were included in the FPIE model that formed the base for the SPRA Plant Response Model, and as such were carried over as valid SSC failure modes in the SPRA model. Note that these events are below the threshold for risk-significance, as defined above in Section 5.4.

Table 5.4-3
Contribution to Unit 1 SCDF by Random Failure

Random Failure Description	% of total SCDF	SCDF Contribution
Running failure of EDG 1-OME-150-AB	0.33%	7.99E-08
Running failure of EDG 1-OME-150-CD	0.28%	6.79E-08
Running failure of SDG 12-OME-250-SDG2	0.11%	2.72E-08
Start failure of TDAFP 1-PP-4	0.10%	2.35E-08

A summary of the Unit 1 SCDF results for each seismic hazard interval is presented in Table 5.4-4.

Table 5.4-4
Contribution to Unit 1 SCDF by Acceleration Interval

Hazard Interval Description	SCDF	% of Total SCDF	Cumulative CDF
%G01 – 0.1-0.2g	2.16E-07	0.89	2.16E-07
%G02 – 0.2-0.3g	2.38E-06	9.75	2.60E-06
%G03 – 0.3-0.4g	7.02E-06	28.76	9.62E-06
%G04 – 0.4-0.5g	6.02E-06	24.67	1.56E-05
%G05 – 0.5-0.6g	3.37E-06	13.81	1.90E-05
%G06 – 0.6-0.7g	1.90E-06	7.78	2.09E-05
%g07 – 0.7-0.8g	1.19E-06	4.88	2.21E-05
%G08 - >0.8g	2.31E-06	9.46	2.44E-05

5.5 SLERF Results

The seismic PRA performed for CNP shows that the point estimate is seismic LERF of 5.65E-06 for Unit 1 and 5.36E-06 for Unit 2. A discussion of the SLERF with uncertainty distribution reflecting the uncertainties in the hazard, fragilities, and model data is presented in Section 5.6 of this report. Important contributors are discussed in the following paragraphs.

The top Unit 1 SLERF cutsets are documented in the CNP SPRA Quantification Notebook [12]. These are briefly summarized in Table 5.5-1 of this report..

SSCs with the most significant seismic failure contribution to Unit 1 SLERF are listed in Table 5.5-2 of this report, sorted by F-V importance. The definition of risk-significance is provided in Section 5.4.

Note for the tables that follow, Unit 2 results were similar. While there may be small numerical differences between the units, risk insights from Unit 1 are applicable to Unit 2.

The seismic fragilities for each of the significant LERF contributors is also provided in Table 5.5-2, along with the corresponding limiting seismic failure mode and method of fragility calculation.

Table 5.5-1
Summary of Top SLERF Accident Cutsets

#	Cutset Probability	BE Probability	Input	Description
1	7.8053E-07	2.6936E-06	%G08	Seismic Initiating Event (>0.8g)
		9.1000E-01	MODE1	PLANT OPERATING IN MODE 1
		3.1844E-01	SF-SCIB-ABC-%G08	SEISMIC FRAGILITY FOR %G08: SC-I Building - Auxiliary Building
2	2.1590E-07	1.3468E-06	%G07	Seismic Initiating Event (0.7g to <0.8g)
		9.1000E-01	MODE1	PLANT OPERATING IN MODE 1
		1.7616E-01	SF-SCIB-ABC-%G07	SEISMIC FRAGILITY FOR %G07: SC-I Building - Auxiliary Building
Cutsets 1 and 2 are direct to LERF failures of the Auxiliary Building at the two highest seismic bins.				
3	1.8983E-07	3.8842E-06	%G05	Seismic Initiating Event (0.5g to <0.6g)
		1.0000E+00	0-SEQ-CET-025	SEQUENCE SBODD-030
		1.0000E+00	0-SEQ-SBODD-030	CTMT FAILURE EARLY RCS PRESS INIT HI THEN LO DIS FAILS - LATE DEPRESS
		9.1000E-01	MODE1	PLANT OPERATING IN MODE 1
		8.6100E-01	NO-SO-PZR-VLV	COMPLEMENT PZR PORV/SV SO DURING BOILDOWN
		9.8000E-01	RCS-DEP-LATE	RCS DEPRESSURIZED PRIOR TO REACTOR VESSEL BREACH
		2.2377E-02	SF-ACCUMC-%G05	SEISMIC FRAGILITY FOR %G05: Accumulator (Room 68)
		1.5463E-03	SF-CP-C-%G05	SEISMIC FRAGILITY FOR %G05: Main Control Room Boards (Correlated Failure)

		8.6503E-01	SF-LSP-C-%G05	SEISMIC FRAGILITY FOR %G05: Offsite Power
		1.1555E-02	-SF-MLOCAC-%G05	SEISMIC FRAGILITY FOR %G05: Medium LOCA
		2.6309E-03	SF-POLARCRNC-%G05	SEISMIC FRAGILITY FOR %G05: Polar Crane Fragility
		7.7917E-04	SF-PZR-C-%G05	SEISMIC FRAGILITY FOR %G05: Pressurizer Support
		5.7262E-07	SF-RCP-C-%G05	SEISMIC FRAGILITY FOR %G05: Reactor Coolant Pump (Room 60)
		2.7247E-05	SF-RV-C-%G05	SEISMIC FRAGILITY FOR %G05: Reactor Vessel (Room 105)
		3.4993E-02	-SF-SCIB-ABC-%G05	SEISMIC FRAGILITY FOR %G05: SC-I Building - Auxiliary Building
		9.0135E-02	SF-SCIB-C-%G05	SEISMIC FRAGILITY FOR %G05: SC-I Building
		1.5492E-03	SF-SCIB-CONTC-%G05	SEISMIC FRAGILITY FOR %G05: SC-I Building - Containment Building
		1.6607E-03	-SF-SCIC-CONT1-C-%G05	SEISMIC FRAGILITY FOR %G05: SC-I Components in Containment up to EL. 625'
		2.6187E-02	SF-SCIV-AB1-C-%G05	SEISMIC FRAGILITY FOR %G05: SC-I Valves in Auxiliary Building - EL. 573-591
		5.6634E-01	SF-SDG-C-%G05	SEISMIC FRAGILITY FOR %G05: Supplemental Diesel Generator System Components
		1.2767E-07	SF-SG-C-%G05	SEISMIC FRAGILITY FOR %G05: Steam Generator (Room 60)
		1.1432E-01	SF-SLOCAC-%G05	SEISMIC FRAGILITY FOR %G05: Small LOCA
Cutset 3 is a dual SBO scenario with failure of SDGs, and containment failure due to CFEH-C, which represents the likelihood of containment failure without hydrogen ignitors available in a scenario which begins with high pressure due to failure of AFW, but depressurizes later due to hot leg failure prior to vessel breach. Hydrogen ignitors are failed due to various seismic impacts, including seismically-induced fires.				
4	1.7951E-07	2.1754E-06	%G06	Seismic Initiating Event (0.6g to <0.7g)
		9.1000E-01	MODE1	PLANT OPERATING IN MODE 1
		9.0678E-02	SF-SCIB-ABC-%G06	SEISMIC FRAGILITY FOR %G06: SC-I Building - Auxiliary Building
Cutset 4 is another direct to LERF failure of the Auxiliary Building at the next highest seismic bin.				
5	1.6759E-07	3.8842E-06	%G05	Seismic Initiating Event (0.5g to <0.6g)
		1.0000E+00	0-SEQ-CET-025	SEQUENCE CET-025
		1.0000E+00	0-SEQ-DLOOP-012	SEQUENCE DLOOP-012

		1.7900E-01	CFEH-C	CTMT FAILURE EARLY RCS PRESS INIT HITHEN LO DIS FAILS - LATE DEPRESS
		5.0000E-01	ESW-1EAST2EAST	FRACTION OF TIME U1 AND U2 EAST ESW PUMPS OPERATING
		9.1000E-01	MODE1	PLANT OPERATING IN MODE 1
		8.6100E-01	NO-SO-PZR-VLV	COMPLEMENT PZR PORV/SV SO DURING BOILDOWN
		9.8000E-01	RCS-DEP-LATE	RCS DEPRESSURIZED PRIOR TO REACTOR VESSEL BREACH
		2.2377E-02	SF-ACCUMC-%G05	SEISMIC FRAGILITY FOR %G05: Accumulator (Room 68)
		1.5463E-03	SF-CP-C-%G05	SEISMIC FRAGILITY FOR %G05: Main Control Room Boards (Correlated Failure)
		8.6503E-01	SF-LSP-C-%G05	SEISMIC FRAGILITY FOR %G05: Offsite Power
		1.1555E-02	SF-MLOCAC-%G05	SEISMIC FRAGILITY FOR %G05: Medium LOCA
		2.6309E-03	SF-POLARCRNC-%G05	SEISMIC FRAGILITY FOR %G05: Polar Crane Fragility
		7.7917E-04	SF-PZR-C-%G05	SEISMIC FRAGILITY FOR %G05: Pressurizer Support
		5.7262E-07	SF-RCP-C-%G05	SEISMIC FRAGILITY FOR %G05: Reactor Coolant Pump (Room 60)
		2.7247E-05	SF-RV-C-%G05	SEISMIC FRAGILITY FOR %G05: Reactor Vessel (Room 105)
		3.4993E-02	SF-SCIB-ABC-% G05	SEISMIC FRAGILITY FOR %G05: SC-I Building - Auxiliary Building
		9.0135E-02	SF-SCIB-C-%G05	SEISMIC FRAGILITY FOR %G05: SC-I Building
		1.5492E-03	SF-SCIB-CONTC-G05	SEISMIC FRAGILITY FOR %G05: SC-I Building - Containment Building
		1.6607E-03	SF-SCIC-CONT1-C-%G05	SEISMIC FRAGILITY FOR %G05: SC-I Components in Containment up to EL. 625'.
		2.6187E-02	SF-SCIV-AB1-C-%G05	SEISMIC FRAGILITY FOR %G05: SC-I Valves in Auxiliary Building - EL. 573-591
		1.2767E-07	SF-SG-C-%G05	SEISMIC FRAGILITY FOR %G05: Steam Generator (Room 60)
		1.1432E-01	SF-SLOCAC-%G05	SEISMIC FRAGILITY FOR %G05: Small LOCA
6	1.6759E-07	3.8842E-06	%G05	Seismic Initiating Event (0.5g to <0.6g)
		1.0000E+00	0-SEQ-CET-025	SEQUENCE CET-025

		1.0000E+00	0-SEQ-DLOOP-012	SEQUENCE DLOOP-012
		1.7900E-01	CFEH-C	CTMT FAILURE EARLY RCS PRESS INIT HI THEN LO DIS FAILS - LATE DEPRESS
		5.0000E-01	ESW-1WEST2WEST	FRACTION OF TIME U1 AND U2 WEST ESW PUMPS OPERATING
		9.1000E-01	MODE1	PLANT OPERATING IN MODE 1
		8.6100E-01	NO-SO-PZR-VLV	COMPLEMENT PZR PORV/SV SO DURING BOILDOWN
		9.8000E-01	RCS-DEP-LATE	RCS DEPRESSURIZED PRIOR TO REACTOR VESSEL BREACH
		2.2377E-02	SF-ACCUMC-%G05	SEISMIC FRAGILITY FOR %G05: Accumulator (Room 68)
		1.5463E-03	SF-CP-C-%G05	SEISMIC FRAGILITY FOR %G05: Main Control Room Boards (Correlated Failure)
		8.6503E-01	SF-LSP-C-%G05	SEISMIC FRAGILITY FOR %G05: Offsite Power
		1.1555E-02	SF-MLOCAC-%G05	SEISMIC FRAGILITY FOR %G05: Medium LOCA
		2.6309E-03	SF-POLARCRN-C-%G05	SEISMIC FRAGILITY FOR %G05: Polar Crane Fragility
		7.7917E-04	SF-PZR-C-%G05	SEISMIC FRAGILITY FOR %G05: Pressurizer Support
		5.7262E-07	SF-RCP-C-%G05	SEISMIC FRAGILITY FOR %G05: Reactor Coolant Pump (Room 60)
		2.7247E-05	SF-RV-C-%G05	SEISMIC FRAGILITY FOR %G05: Reactor Vessel (Room 105)
		3.4993E-02	SF-SCIB-ABC-%G05	SEISMIC FRAGILITY FOR %G05: SC-I Building - Auxiliary Building
		9.0135E-02	SF-SCIB-C-%G05	SEISMIC FRAGILITY FOR %G05: SC-I Building
		1.5492E-03	SF-SCIB-CONT-C-%G05	SEISMIC FRAGILITY FOR %G05: SC-I Building - Containment Building
		1.6607E-03	SF-SCIC-CONT1-C-%G05	SEISMIC FRAGILITY FOR %G05: SC-I Components in Containment up to EL. 625'
		2.6187E-02	SF-SCIV-AB1-C-%G05	SEISMIC FRAGILITY FOR %G05: SC-I Valves in Auxiliary Building - EL. 573-591
		1.2767E-07	SF-SG-C-%G05	SEISMIC FRAGILITY FOR %G05: Steam Generator (Room 60)
		1.1432E-01	-SF-SLOCAC-%G05	SEISMIC FRAGILITY FOR %G05: Small LOCA
Cutsets 5 and 6 contain similar failures but with opposite initial alignments (ESW-1EAST2EAST vs ESW-1WEST2WEST). They follow sequence DLOOP-012 due to various ESW failures,				

including those caused by seismically-induced fires that fail ESW room cooling fans with a very high probability (so are set TRUE for the LERF calculation). Containment failure is due to the same conditions as in cutset 3.				
7	1.2953E-07	2.6936E-06	%G08	Seismic Initiating Event (>0.8g)
		9.1000E-01	MODE1	PLANT OPERATING IN MODE 1
		5.2843E-02	SF-SCIB-CONTC-%G08	SEISMIC FRAGILITY FOR %G08: SC-I Building - Containment Building
Cutset 7 is a direct to LERF failure of Containment at the highest seismic bin.				
8	1.2369E-07	3.8842E-06	%G05	Seismic Initiating Event (0.5g to <0.6g)
		9.1000E-01	MODE1	PLANT OPERATING IN MODE 1
		3.4993E-02	SF-SCIB-ABC-%G05	SEISMIC FRAGILITY FOR %G05: SC-I Building - Auxiliary Building
Cutset 8 is another direct to LERF failure of the Auxiliary Building at the next highest seismic bin.				
9	1.1949E-07	2.6936E-06	%G08	Seismic Initiating Event (>0.8g)
		1.0000E+00	0-SEQ-CET-025	SEQUENCE CET-025
		1.7900E-01	CFEH-C	CTMT FAILURE EARLY RCS PRESS INIT HI THEN LO DIS FAILS - LATE DEPRESS
		9.1000E-01	MODE1	PLANT OPERATING IN MODE 1
		8.6100E-01	NO-SO-PZR-VLV	COMPLEMENT PZR PORV/SV SO DURING BOILDOWN
		9.8000E-01	RCS-DEP-LATE	RCS DEPRESSURIZED PRIOR TO REACTOR VESSEL BREACH
		3.1844E-01	SF-SCIB-ABC-%G08	SEISMIC FRAGILITY FOR %G08: SC-I Building - Auxiliary Building
		5.0000E-01	SF-SCIB-C-%G08	SEISMIC FRAGILITY FOR %G08: SC-I Building
		5.2843E-02	SF-SCIB-CONT-C-%G08	SEISMIC FRAGILITY FOR %G08: SC-I Building - Containment Building
		1.1984E-05	SF-SG-C-%G08	SEISMIC FRAGILITY FOR %G08: Steam Generator (Room 60)
Cutset 9 is CDF cutset 1 (Screenhouse failure), with containment failure due to the same conditions as in LERF cutsets 3, 5, and 6.				
10	1.0763E-07	3.8842E-06	%G05	Seismic Initiating Event (0.5g to <0.6g)
		1.0000E+00	0-SEQ-CET-025	SEQUENCE CET-025
		1.0000E+00	0-SEQ-MSLB-007	SEQUENCE MSLB-007
		1.7900E-01	CFEH-C	CTMT FAILURE EARLY RCS PRESS INIT HI THEN LO DIS FAILS - LATE DEPRESS
		9.1000E-01	MODE1	PLANT OPERATING IN MODE 1

		8.6100E-01	NO-SO-PZR-VLV	COMPLEMENT PZR PORV/SV SO DURING BOILDOWN
		9.8000E-01	RCS-DEP-LATE	RCS DEPRESSURIZED PRIOR TO REACTOR VESSEL BREACH
		2.2377E-02	SF-ACCUMC-%G05	SEISMIC FRAGILITY FOR %G05: Accumulator (Room 68)
		1.5463E-03	SF-CP-C-%G05	SEISMIC FRAGILITY FOR %G05: Main Control Room Boards (Correlated Failure)
		1.1555E-02	SF-MLOCAC-%G05	SEISMIC FRAGILITY FOR %G05: Medium LOCA
		2.7049E-01	SF-NSCIB-TB1-C-%G05	SEISMIC FRAGILITY FOR %G05: Turbine Building Collapse
		2.6309E-03	SF-POLARCRN-C-%G05	SEISMIC FRAGILITY FOR %G05: Polar Crane Fragility
		7.7917E-04	SF-PZR-C-%G05	SEISMIC FRAGILITY FOR %G05: Pressurizer Support
		5.7262E-07	SF-RCP-C-%G05	SEISMIC FRAGILITY FOR %G05: Reactor Coolant Pump (Room 60)
		2.7247E-05	SF-RV-C-%G05	SEISMIC FRAGILITY FOR %G05: Reactor Vessel (Room 105)
		3.4993E-02	SF-SCIB-ABC-%G05	SEISMIC FRAGILITY FOR %G05: SC-I Building - Auxiliary Building
		9.0135E-02	SF-SCIB-C-%G05	SEISMIC FRAGILITY FOR %G05: SC-I Building
		1.5492E-03	SF-SCIB-CONT-C-%G05	SEISMIC FRAGILITY FOR %G05: SC-I Building - Containment Building
		1.6607E-03	-SF-SCIC-CONT1-C-%G05	SEISMIC FRAGILITY FOR %G05: SC-I Components in Containment up to EL. 625'
		1.2767E-07	SF-SG-C-%G05	SEISMIC FRAGILITY FOR %G05: Steam Generator (Room 60)
		1.1432E-01	SF-SLOCAC-%G05	SEISMIC FRAGILITY FOR %G05: Small LOCA
Cutset 10 is a seismically-induced Main Steam Line Break due to Turbine Building collapse, with failure of high pressure injection that leads to core damage. Containment failure is due to the same conditions as in several previous LERF cutsets with CFEH-C.				

Table 5.5-2
Unit 1 SLERF Importance Measures Ranked by F-V

Event	Description	F/V	SLERF Contribution	Median Capacity (g)	βr	βu	Failure Mode	Fragility Method
SF-SCIB-AB	SC-I Building Auxiliary Building	0.157	8.85E-07	1.038	0.24	0.26	Structural	Screening 15C4313-RPT-001
SF-LSP	Loss of Offsite Power	0.096	5.41E-07	0.3	0.24	0.49	Structural	CDFM 15C4113-RPT-007
RELAY_D1	Relay Fragility Group Relay_D1	0.051	2.90E-07	0.470	0.27	0.49	Chatter	SOV 15C4313-CAL-021`
RELAY_D2	Relay Fragility Group Relay_D_2	0.046	2.58E-07	0.470	0.27	0.49	Chatter	SOV 15C43213-CAL-021
SF-SDG	Supplemental Diesel Generator System Components	0.038	2.14E-07	0.503	0.24	0.49	Structural	CDFM 15C4313-RPT-007
SF-NS-FIRE30	Seismic-Induced Fire Originating from Boiler	0.036	2.03E-07	0.327	0.24	0.18	Structural	Refined CDFM 15C4313-CAL-023
SF-MLOCA	Medium LOCA	0.036	2.01E-07	0.531	0.35	0.45	Functional	Generic 15C4313-RPT-007
SF-A11-PNLS	Relay Panels in Control Room	0.029	1.65E-07	0.740	0.24	0.38	Anchorage	CDFM 15C4313-CAL-024
SF-NS-FIRE35L	Seismic-Induced Fire Originating from Main Turbine Oil System - Limited	0.028	1.57E-07	0.327	0.24	0.18	Structural	Refined CDFM 15C4313-CAL-023
RELAY_B5_U1	Fragility Group Relay_B_5_U1	0.026	1.45E-07	0.489	0.24	0.38	Chattel	Refined CDFM 15C4313-CAL-021
SF-NSCIB-TB1	Turbine Building Collapse	0.024	1.33E-07	0.677	0.24	0.26	Structural	CDFM 15C4313-CAL-020
SF-VSLOCA	Seismic-Induced Very Small LOCA	0.015	8.54E-08	0.507	0.24	0.32	Functional	Judgement 15C4313-RPT-007

Table 5.5-2
Unit 1 SLERF Importance Measures Ranked by F-V

Event	Description	F/V	SLERF Contribution	Median Capacity (g)	βr	βu	Failure Mode	Fragility Method
RELAY_B4_U1	Fragility Group Relay_B_4_U1	0.014	8.15E-08	0.614	0.24	0.38	Functional	Refined CDFM 15C4313-CAL-021
SF-SCIB-CONT	SC-I Building - Containment Building	0.014	7.96E-08	1.557	0.24	0.26	Structural	Screening 15C4313-RPT-001
SF-FLD-W3L1	Seismic-Induced Flood from Other Non-Safety Related Systems	0.012	6.90E-08	0.25	0.24	0.32	Structural	Generic 15C43132-RPT-007
SF-NS-FIRE-FPTE1	Seismic-Induced Fire Originating from Main Feed Pump - 10% oil	0.011	6.46E-08	0.285	0.24	0.38	Structural	Generic 15C4313-RPT-007
SF-CR-CEIL-3	Control Room Ceiling Section 3	0.010	5.60E-08	1.074	0.32	0.41	Structural	SOV 15C4313-CAL-031
SF-CR-CEIL-2	Control Room Ceiling Section 2	0.009	5.36E-08	1.074	0.32	0.41	Structural	SOV 15C4313-CAL-031
SF-BATCD	Plant Battery CD (Room 201)	0.009	5.06E-08	0.689	0.24	0.38	Anchorage	CDFM 15C4313-CAL-026
RELAY_B2_U1	Fragility Group Relay_B_2_U1	0.009	4.85E-08	0.608	0.24	0.38	Chatter	Refined CDFM 15C4313-CAL-021

Table 5.5.3 list the most significant non-seismic SLERF SSC random failures for Unit 1. These events were included in the FPIE model that formed the base for the SPRA Plant Response Model, and as such were carried over as valid SSC failure modes in the SPRA model. Note that these events are below the threshold for risk-significance defined above in Section 5.4.

Table 5.5-3
Contribution to Unit 1 SLERF by Random Failure

Random Failure Description	% of total SCDF	SCDF Contribution
Running failure of SDG 12-OME-250-SDG2	0.13%	7.38E-09
Running failure of EDG 1-OME-150-AB	0.08%	4.59E-09
Running failure of EDG 1-OME-150-CD	0.07%	3.82E-09

A summary of the Unit 1 SLERF contribution for each seismic hazard interval is presented in Table 5.5-4.

Table 5.5-4
Contribution to Unit 1 SLERF by Acceleration Interval

Hazard Interval Description	SLERF	% of Total SLERF	Cumulative SLERF
%G01 – 0.1-0.2g	3.24E-08	0.57	3.24E-08
%G02 – 0.2-0.3g	2.60E-07	4.60	2.92E-07
%G03 – 0.3-0.4g	8.24E-07	14.59	1.12E-06
%G04 – 0.4-0.5g	8.76E-07	15.51	1.99E-06
%G05 – 0.5-0.6g	9.57E-07	16.94	2.95E-06
%G06 – 0.6-0.7g	6.76E-07	11.97	3.63E-06
%G07 – 0.7-0.8g	5.43E-07	9.61	4.17E-06
%G08 - >0.8g	1.48E-06	26.20	5.65E-06

5.6 SPRA Quantification Uncertainty Analysis

Parameter uncertainty in seismic PRA results comes from seismic hazard curve uncertainty, the SSC fragility uncertainties, and uncertainties in the human interaction and random failure calculations. SPRA model parameter uncertainty was quantified using the EPRI UNCERT code. The results are provided in Table 5.6-1. Figures 5.6-1 and 5.6-2, shows the SCDF and SLERF curves of cumulative probability and probability density function.

Table 5.6-1: SPRA CDF and LERF Parametric Uncertainty Results

	CDF	LERF
Mean	5.46E-05	9.72E-06
5%	4.23E-05	6.07E-06
Median	5.34E-05	9.26E-06
95%	7.13E-05	1.49E-05
StdDev	9.23E-06	2.81E-06
Skewness	1.247	1.339
Sample Size	10000	10000
Cutsets	10000	10000 (all, see note below)
Sampling Method	Monte Carlo	Monte Carlo
Point Estimate (ACUBE with specified number of cutsets)	3.42E-05	5.65E-05

The UNCERT runs were performed using the Monte Carlo method of sampling and a total of 10,000 samples. Both SCDF and SLERF runs solved 10,000 cutsets using ACUBE. Note that 10,000 cutsets comprise the entire SLERF population. The distribution for both SCDF and SLERF appears generally uniform. The uncertainty is generally dominated by the hazard uncertainty. Since much of the seismic risk comes from higher seismic intervals (greater ground motion), the failure probabilities at this ground motion are generally very high and therefore will not contribute much in the way of uncertainty. The point estimate SCDF / SLERF was calculated for each acceleration interval using mean values for the seismic hazard frequency, mean values for the seismic fragilities, and mean values for the random failures and human error probabilities. These acceleration interval point estimate means were then summed for the total SCDF and SLERF point estimate means.

NOTE: Performing ACUBE in combination with UNCERT can exceed the capabilities of typical computers. The LERF calculation above can process all the cutsets, but the CDF calculation was unsuccessful. Therefore, a corresponding point-estimate CDF ACUBE calculation with 10,000 cutsets is provided. Comparison with the point estimate values indicates that the point estimates provide a reasonable approximation of the mean values.

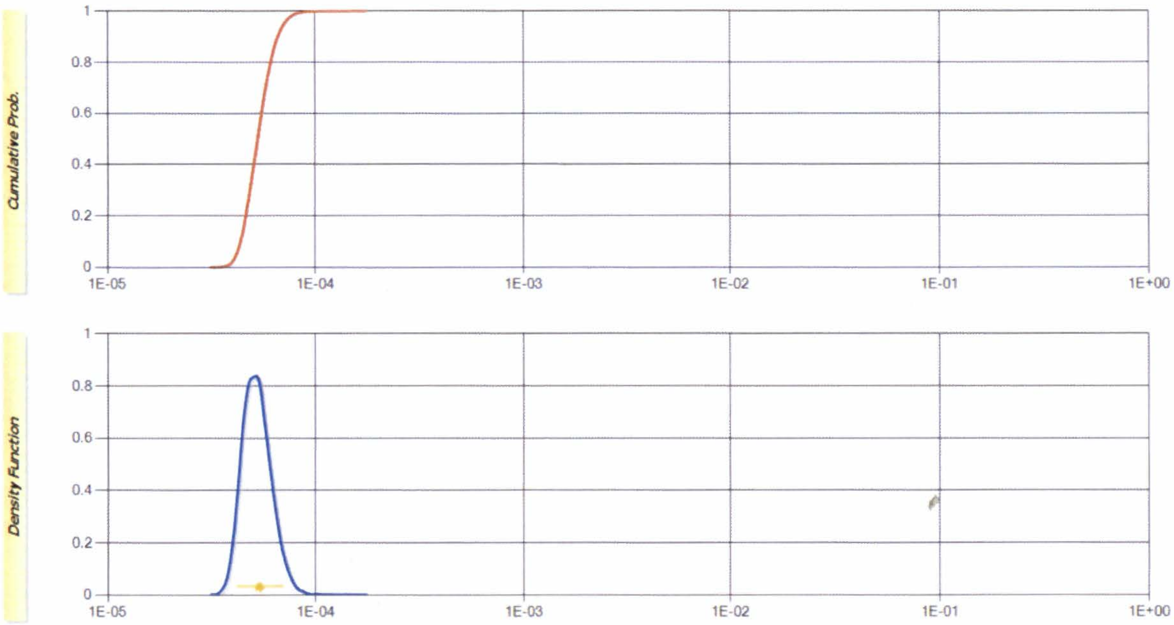


Figure 5.6-1: SCDF Parametric Uncertainty

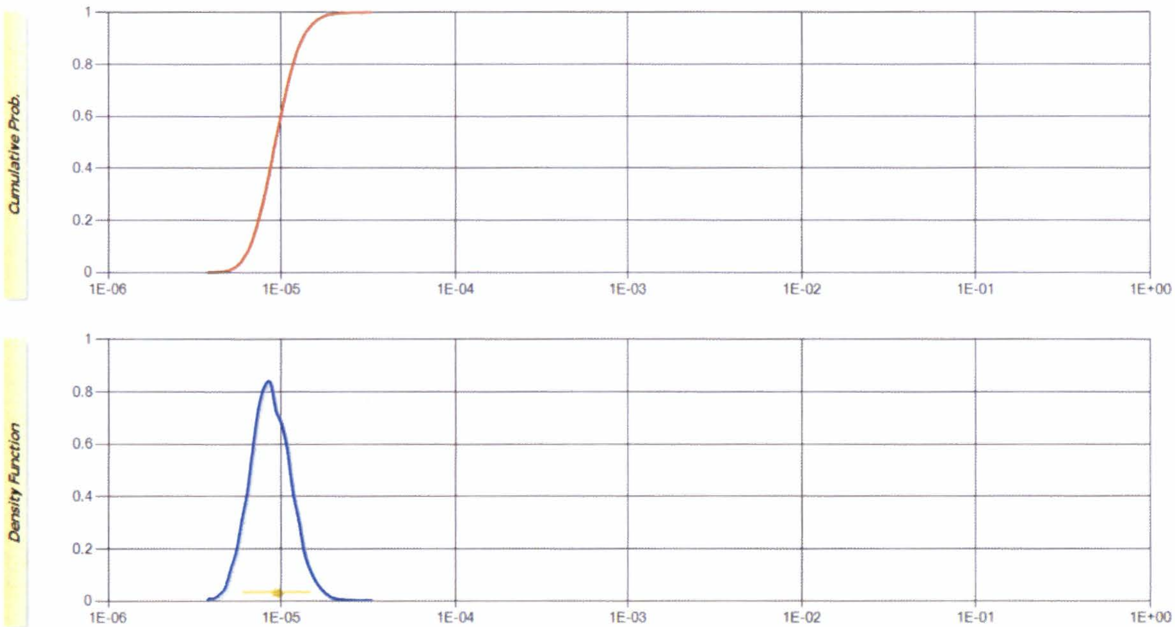


Figure 5.6-2: SLERF Parametric Uncertainty

Model uncertainty is introduced when assumptions are made in the SPRA model and inputs to represent plant response, when there may be alternative approaches to particular aspects of the modeling, or when there is no consensus approach for a particular issue. The following definition for identifying modeling uncertainty has been developed:

- Significant interpretations to infer behavior are required to develop a model (this is the case where some information is available, but is not sufficient to derive a definitive model or value),
- The phenomena or nature of the event or failure mode being modeled is not completely understood, or
- There is general agreement that the issue represents a potential source of modeling uncertainty.

Important model uncertainties were addressed through the sensitivity studies described in Section 5.7 to determine the potential impact on SCDF or SLERF.

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

5.7 SPRA Quantification Sensitivity Analysis

A series of sensitivity studies was performed to quantify the potential impact of various uncertainties or model variables on the SPRA results and insights. The intent was to evaluate uncertainties or variables to which the model may be sensitive, which may identify steps that may be taken to reduce significant uncertainties or quantify risk reduction of plant improvement considerations. It also supports applications of the SPRA that must consider uncertainty in context with the application. The truncation levels for each sensitivity case were kept the same as the baseline results, except where specifically noted (such as to not exceed computational limits). Unit 1 is used for all sensitivity calculations but the results were generally representative for both units.

5.7.1 Sensitivity Descriptions

Case 1: Recovery of DIS

Recovery of power to the distributed ignition system (hydrogen ignitors) is expected to be a valuable action to prevent containment failure and large early release. The power may come from a new, separate power system or from the supplemental diesel generator.

Case 2: SDG Fragility

This case modifies SF-SDG-AM to 0.33 in the type codes based on a lower bound of fragility estimated as 10% greater than SF-LSP.

Case 3: Train CD Battery Fragility
This case modifies SF-BATCD-AM to 0.89 in the type codes based on the fragility level used for BATAB.

Case 4: Relay Fragility
Various risk-significant relay fragilities were tested in the following subcases, with the last case being a combination of all of the subcases.

Case 4a: Relay Groups D1 & D2
This case removes all events for Relay_D_1 or Relay_D_2.

Case 4b: Relay Groups B2, B4, & B5 (non-chatter)
This case removes all events for Relay_B_2 or Relay_B_4 or Relay_B_5 that are not chatter-related.

Case 4c: All Case 4 Groups
This case removes all events for the previous cases 4a-4b.

Case 5: FLEX
This case models the potential impact of the FLEX equipment already set up in the PRA model, though this equipment is not currently credited.

Case 6: CST Recovery Action
The impact of the CST recovery action, 1D0--ESWEARLYHE, can be quickly determined by its F-V importance in the overall model results.

Case 7: No Rugged Components
Instead of including Rugged Components in the model with a screening fragility value, they were excluded from the model. This case removes all SF-RUGGED events.

Case 8: Soil Failure
Soil failure is a highly uncertain event that can dominate results and overwhelm other insights from the SPRA since it is modeled as a direct-to-CDF and direct-to-LERF event. The base SPRA model does not include SOIL failure in this manner. However, these subcases explore the effects of different potential modeling assumptions or soil fragility parameters.

Case 8a: No Soil Failure
This case activates the soil failure in the model calculation. The current parameters in the model were maintained based on HCLPF=0.34g.

Case 8b: Soil Failure Modification 1
This case changes the soil fragility values by increasing Am to reflect a HCLPF=0.5g (Am scales by the same factor).

Case 8c: Soil Failure Modification 2

This case changes the soil fragility values by increasing Am to reflect a HCLPF=1.0g (Am scales by the same factor).

Case 9: Quantification with High Fragilities set to True

The CNP SPRA quantification approach applies an assumption to simplify the LERF calculations using a flag file that sets most fragility events that have very high failure probabilities to TRUE. These subcases examine the impacts of that assumption on the model results.

Case 9a: High Fragilities set to True for CDF

This case applies the same assumption to the CDF calculations.

Case 9b: High Fragilities are not set to True

This case removes the flag file so that all CDF and LERF cases run without this assumption.

Case 10: Sensitivity on Hazard Interval Definition

The CNP SPRA discretizes the seismic hazard into eight intervals, each of uniform width; however, FRANX can support definition of more intervals. Using more intervals can be more accurate in the sense that, as more intervals are defined, the PGA range for each interval narrows, and the upper/lower bounds for each interval deviates less from its assigned representative ground motion.

The CNP hazard intervals were defined to discretize the region between 0.1 and 0.8g. Eight intervals were defined, with the first seven having a width of 0.1g, and the eighth covering everything above 0.8g. This provides a reasonable balance between SPRA resolution and computing limitations.

Case 10a: Additional Hazard Intervals at the Upper End

The SPRA model was rebuilt using ten seismic intervals by adding two additional intervals at the upper end (without changing the first seven intervals).

Case 10b: Additional and Redefined Intervals

The SPRA model was rebuilt adding four intervals increasing the seismic intervals to twelve. The intervals were redefined creating narrower ranges of eleven intervals, as the first interval was not narrowed.

Case 11: Large Correlation Groups

Some of the fragility groups contain quite a few SSCs that are different component types, are in different locations and elevations, and impact different accident sequence functions. These groups were reviewed to identify where significant differences exist that could justify refinement to reduce the size of the correlated group. One group for components (SF-SCIC-AB1) and one group for valves (SF-SCIV-AB1) was identified for analysis.

Case 11a: SF-SCIC-AB1

Seismic fragility group SF-SCIC-AB1 contains components from AC power, AFW, DG, DIS, & ECCS systems. This fragility group was refined to smaller groups by reassigning the corresponding components in the SEL to new fragility groups for AC, AFW, DG, & ECCS. Fragility parameters for the new groups were kept the same as the original group DIS was not removed from the original group.

Case 11b: SF-SCIV-AB1

Seismic fragility group SF-SCIV-AB1 contains components from DG, ECCS, & SG systems. This fragility group was refined by reassigning the corresponding components in the SEL to new fragility groups for DG & SG with the same fragility parameters as the original group. ECCS valves were left in the existing group.

Case 12: Delayed Evacuation

Two aspects of the SPRA impact the determination of early releases which could be affected by a delay in offsite evacuation. First, the CFE probabilities when hydrogen ignitors are not functional accounts for hydrogen burns that happen after effective evacuation. Therefore, the supporting analysis for CFEs was reviewed and cases where evacuation is credited to avoid a Large Early release were added back into the CFE calculations. Second, all Large Late release were assumed to now be "Early" by inserting the SPRA top Gate LL under SLERF. This addition greatly slows the model quantification, so truncations were adjusted as necessary to allow the calculation to complete.

Case 13: Containment Liner for LERF

The SF-SCIB-CONT failure is broken up such that CONT applies to only the containment internals, which were assumed to go direct to core damage (without a large or medium LOCA), while a new group SF-SCIB-LINER represents failure of the containment liner that would go direct to LERF. The new LINER group uses HCLPF 1.0g (Am=2.26).

Case 14: Generic Fragilities

The best available information for building and LOCA fragilities was determined to be generic values for the baseline SPRA model. To investigate the impacts of these generic fragilities, these values were adjusted, first by smaller groups, then by a combined calculation, to investigate their impact on the SPRA results.

Case 14a: Generic Building Fragilities

Increase SF-SCIB-CONT, SF-SCIB-AB, SF-NSCIB-TB1 Am by 50%.

Case 14b: Generic LOCA Fragilities

Increase SF-MLOCA, SF-SLOCA, SF-VSLOCA Am by 50%.

Case 14c: Combined Generic Fragilities

Combine case 14a and 14b.

Sensitivity on Ground Motion Quantity

SPRAs are most commonly built around the PGA metric; the CNP SPRA is based on PGA. The ground motion metric used for an SPRA may in fact be any spectral acceleration or any ground motion quantity of interest; the only requirement being that both the hazard and fragilities be based on the same quantity. An alternative would be to base the SPRA on a ground motion quantity that is highly correlated with damage to critical structures and equipment at the plant (this may be in the range of 2-10 Hz).

The SPRA calculated risk metrics (SCDF and SLERF) would only be impacted by using another spectral instead of PGA if the shape of the UHRS is significantly different for different hazard return periods (e.g., if the ratio of accelerations at 5 Hz / accelerations at PGA is significantly different between the UHRS 10-4 and 10-6 curves). For CNP, the UHRS shapes are the same; as such, use of PGA vs another SA is not a significant modeling uncertainty for CNP. For these reasons, this case was not run.

5.7.2 Sensitivity Results

Table 5.7.2 shows the results of the sensitivity cases. These results are for Unit 1. Unit 2 results would be similar.

Table 5.7.2 – Sensitivity Case Results

Case	SCDF	SLERF	% CDF Change	% LERF Change
Base Model	2.44E-05	5.65E-06	N/A	N/A
1-DIS Recovery	2.44E-05	2.80E-06	0.0%	-50.5%
2-SDG fragility	2.51E-05	5.94E-06	3.0%	5.1%
3-CD bat fragility	2.40E-05	5.61E-06	-1.7%	-0.8%
4a-RelayD1D2	2.29E-05	5.29E-06	-6.3%	-6.4%
4b-RelayB2B4B5noCH	2.34E-05	5.41E-06	-4.1%	-4.3%
4c-RelaysAllAbove	2.17E-05	5.08E-06	11.1%	-10.2%
5-FLEX	2.44E-05	5.64E-06	-0.1%	-0.2%
6-CST Recovery	2.41E-05	5.62E-06	-1.1%	-0.5%
7-No Rugged	2.44E-05	5.65E-06	0.0%	-0.1%
8a-Soil 1	2.46E-05	9.97E-06	0.8%	76.3%
8b-Soil 2	2.45E-05	6.61E-06	0.2%	17.0%
8c-Soil 3	2.44E-05	5.66E-06	0.0%	0.1%
9a-Use HighFragT for CDF	2.45E-05	5.65E-06	0.2%	0.0%
9b-No HighFragT	2.44E-05	5.02E-06	0.0%	-11.1%

Table 5.7.2 – Sensitivity Case Results				
Case	**SCDF**	**SLERF**	**% CDF Change**	**% LERF Change**
10a-Hazard Intervals at Upper End	2.40E-05	5.94E-06	-1.6%	5.0%
10b-Hazard Intervals Redefined	2.27E-05	5.54E-06	-7.1%	-2.0%
11a-Large Correlation Groups	2.44E-05	5.65E-06	0.0%	0.0%
11b-Large Correlation Groups	2.43E-05	5.62E-06	-0.6%	-0.5%
12-Delayed evacuation	2.44E-05	2.21E-05	0.0%	291.0%
13-Containment Liner for LERF	2.43E-05	5.53E-06	-0.5%	-2.1%
14a-Generic Building Fragilities	2.39E-05	4.65E-06	-2.2%	-17.8%
14b-Generic LOCA Fragilities	2.30E-05	5.41E-06	-5.9%	-4.3%
14c-Both Generics	2.25E-05	4.36E-06	-7.8%	-22.8%

5.8 SPRA Logic Model and Quantification Technical Adequacy

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

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

6.0 Conclusions

A seismic PRA has been performed for CNP Units 1 and 2 in accordance with the guidance in the SPID [2]. The SPRA shows that the CNP Unit 1 seismic CDF is 2.44 E-05 and the seismic LERF is 5.65 E-06. CNP Unit 2 seismic CDF is 2.38 E-05, and the seismic LERF is 5.36 E-06. Uncertainty, importance, and sensitivity analyses were performed. The sensitivity studies were performed to understand critical assumptions, test model stability to quantification parameters and seismic hazards, and determine risk reduction value in considering seismic plant modifications.

This SPRA report reflects the as-built/ as-operated CNP Unit 1 and Unit 2 configurations as of May 8, 2018. Section A.10, of Appendix "A," of this report addresses the impact of plant changes to the SPRA after May 8, 2018.

I&M reviewed the SPRA Phase 2 assessment guidance provided in [71] and [72]. Potential modifications for CNP noted in [72] were not identified as significant risk SCDF contributors, per Table 5.4-2 of this report, and were therefore not considered to be beneficial modifications. Table 27 of the CNP SPRA Quantification Notebook [12] lists sensitivity studies performed for possible modifications or actions considered by I&M and the effectiveness of each consideration. The results show that modifications or actions do not meet the 1E-5 delta CDF reduction criteria in the NRC guidance in [71] and additional consideration is not necessary.

No other significant seismic hazard vulnerabilities were identified for the CNP SSCs. However, the loss of offsite power risk was further investigated, with the recognition that I&M cannot influence industry established low offsite power fragilities or events affecting offsite power outside of the station's power block. The investigation determined that a considerable seismic LERF reduction could be gained by providing power to the containment DIS in the event of loss of offsite power. I&M plans to develop and implement a plant modification that will provide power to the DIS to mitigate the loss of offsite power event. This modification is estimated to reduce SLERF by approximately 50%, as shown in sensitivity Case 1 of Table 27 of the CNP SPRA Quantification Notebook [12]. The approximate Unit 1 SLERF, after this modification is implemented, will be about 2.80E-06 with similar benefits for Unit 2. Conceptual engineering has begun for this modification. Once initial engineering actions have been completed, an implementation schedule will be developed for this modification.

7.0 References

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53. Letter from Q. S. Lies, I&M, to NRC, "Donald C. Cook Nuclear Plant Units 1 and 2, Spent Fuel Pool Evaluation Supplemental Report, Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," dated October 12, 2016, ADAMS Accession No. ML16288A843.

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55. Email from J. Waters, I&M, to B. Titus, NRC, "Plans re. GMRS for D. C. Cook," dated August 3, 2017.

56. Email from B. Titus, NRC, to J. Waters, I&M, "Plans re. GMRS for D. C. Cook," dated August 07, 2017.

57. Electric Power Research Institute document EPRI 3002012980, "Methodology for Seismically Induced Internal Fire and Flood Probabilistic Risk Assessment," dated June 27, 2018.

58. Letter from B. E. Thomas, NRC, to R. J. Budnitx, Lawrence Berkley National Laboratory, and C. R. Grantom, "U.S. Nuclear Regulatory Commission Acceptance of ASME/ANS RA-S Case 1," dated March 12, 2018, ADAMS Accession No. ML18017A963.

59. NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3, dated January 2018, ADAMS Accession No. ML17317A256.

60. CNP Document – "2015 Internal Events Peer Review Disposition Report," Revision 0.

61. CNP Document SD-121023-001 – "In Response to the 50.54(f) Information Request Regarding Fukushima Near-Term Task Force Recommendation 2.3: Seismic for D.C. Cook Unit 1 and Unit 2," Revision 2.

62. CNP Document SD-140218-001 - "EPRI Seismic Hazard, GMRS and Screening Report for D. C. Cook Nuclear Plant, Calculation of Seismic Hazards for CEUS Sites," Revision 1.

63. CNP document 15C4313-RPT-003, "Summary of Building Response Analysis for the Cook Nuclear Plant (CNP) Unit 1 & Unit 2 SPRA," Revision 4.

64. Report AEPDCC-00058-REPT-001, "SPRA F&O Independent Assessment and Focused-Scope Peer Review, Donald C. Cook Nuclear Plant Units 1 and 2," Revision A.

65. CNP document DC Cook-PR-09, "Evaluation of secondary hazards associated with vibratory ground motion at DC Cook Nuclear Plant for SPRA," Revision 0.

66. Letter from M. J. Ross-Lee, NRC, G. Krueger, NEI, "U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 07-12, and 12-13, Close-Out of Facts and Observations (F&Os)," dated May 3, 2017, ADAMS Accession Number M27L17079A4.

67. Electric Power Research Institute document EPRI 3002011627 "Updated Equipment Seismic Capacities from Experience Data for Use in the Fragility Calculations," dated November 2017.

68. Electric Power Research Institute document EPRI 3002013017 "Updated Equipment Seismic Capacities from Experience Data for Use in Fragility Calculations: Phase II," dated November 2018.

69. CNP document 15C4313-CAL-010 "Response Analysis of Auxiliary Building," Revision 3.

70. Memorandum from J. Fong, NRC, to J. G. Gitter, NRC, "Guidance for Determination of Appropriate Regulatory Action Based on Seismic Probabilistic Risk Assessment Submittals in Response to Near Term Task Force Recommendation 2.1: Seismic," dated August 29, 2017, ADAMS Accession No. ML17146A200.

71. Copy of Phase 2 Worksheet (Guidance for Determination of Appropriate Regulatory Action Based on NTTF 2.1 SPRA Submittals), dated May 26, 2017, Adams Accession No. ML17146A206.

8.0 Abbreviations and Acronyms

AC	Alternating Current
ACI	American Concrete Institute
ACS	American Chemical Society
ACUBE	Advanced Cutset Upper Bound Estimator
ADAMS	Agencywide Documents Access and Management System
AFE	Annual Frequencies of Exceedance
AFW	Auxiliary Feed Water
Am	Median Fragility seismic capacity
ANS	American Nuclear Society
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
βc	Combined uncertainty of βr and βu
βr	Logarithmic standard deviation of randomness of capacity
βu	Logarithmic standard deviation of uncertainty of median value
BE	Best Estimate
bldg.	Building
BWR	Boiling Water Reactor
CAFTA	Computer Aided Fault Tree Analysis
CC	Capability Category
CCW	Component Cooling water
CCWT	Loss of RCP Seal Cooling
CDF	Core Damage Frequency
CDFM	Conservative Deterministic Failure Margin
CEUS	Central-Eastern United States
CEUS-SSC	Central-Eastern United States Seismic Source Characterization
CFE	Containment Failure Event
CFR	Code of Federal Regulations
CNP	Donald C. Cook Nuclear Plant
CST	Condensate Storage Tank
CVCS	Chemical and Volume Control System
DC	Direct Current
DESL	Dual-Unit Loss of ESW After Single-Unit LOOP
DG	Diesel Generator
DIS	Distributed (Hydrogen) Ignition System
DLOOP	Dual Unit Loss of Offsite Power
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EPRI	Electric Power Research Institute
ESEP	Expedited Seismic Evaluation Program
ESW	Essential Service Water
FEM	Finite Element Model
FLDC	Internal-Flooding-Induced Loss of CCW
FLDT	Internal-Flooding-Induced Transient
FLEX	Flexible and diverse mitigation strategies implemented per NEI 12-06
FIRS	Foundation Input Response Spectra
F&Os	facts and observations
FPIE	Full Power Internal Event

FRANX	Software for spatial dependences in PRA modeling
ft.	feet
F-V	Fussell-Vesely
g	Acceleration due to gravity
GERS	Generic Equipment Ruggedness Spectra
GIP	General Implementation Procedure
GMC	Ground Motion Characterization
GMM	ground-motion model
GMPE	Ground Motion Prediction Equation
GMRS	Ground Motion Response Spectra
gpm	Gallons per minute
GRS	ground response spectra
HCLPF	High Confidence of a Low Probability of Failure
HEP	Human Error Probability
HFE	Human Failure Event
HLR	High Level Requirement
HPI	High Pressure Injection
HRA	Human Reliability Analysis
HVAC	Heating, Ventilation and Air Conditioning
Hz	Hertz
kV	kilovolt
I&M	Indiana Michigan Power Company (Licensee for CNP)
in.	inch
IPEEE	Individual Plant Examination for External Events
ISLOCA	Interfacing System Loss of Coolant Accident
ISFSI	Independent Spent Fuel Storage Installation
ISTH	In-Structure Time History
ISRS	In-Structure Response Spectra
JCNRM	Joint Committee on Nuclear Risk Management
KEPCO	Korea Electric Power Corporation
lb.	Pound
LB	Lower Bound
LERF	Large Early Release Frequency
LMSM	Lumped Mass Stick Model
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LSP	Loss of Site Power
MAAP	Modular Accident Analysis Program
MAFEs	Mean Annual Frequencies of Exceedances
MFLB	Main Feedwater Line Break
MSLB	Main Steam line break
MUP	Make-up Plant
NEI	Nuclear Energy Institute
NEP	Non-Exceedance Probability
NESW	Non-Essential Service Water
NFPA	National Fire Protection Association
NMFS	New Madrid Fault System
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NTTF	Near Term Task Force

PAC	Plant Air Compressor
PGA	Peak Ground Acceleration
P&ID	Piping and Instrumentation Diagram
PRA	Probabilistic Risk Assessment
PSHA	Probabilistic Seismic Hazard Analysis
psi	Pounds per square inch
PWR	Pressurized Water Reactor
RAW	Risk Achievement Worth
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RLME	Repeat Large Magnitude Earthquake
RLE	Review Level Earthquake
ROB	Rule-of-the-Box
RRW	Risk Reduction Worth
RVT	Random Vibration Theory
SA	Spectral Acceleration
SAREX	KEPCO PRA software
SASSI	System Analysis for Soil-Structure Interaction
SBO	Station Blackout
SBODD	Dual unit SBO after dual unit LOOP with SDGs available
SBOSD	Single unit SBO after dual unit LOOP without SDGs available
SBOSL	Single unit SBO after single unit LOOP without SDGs available
SC	Seismic Class
SCDF	Seismic Core Damage Frequency
SDGDD	Dual unit SBO after dual unit LOOP with SDGs available
SDGSD	Single unit SBO after dual unit LOOP with SDGs available
SDGSL	Single unit SBO after single unit LOOP with SDGs available
SG	Steam Generator
SI	Safety Injection
SIET	Seismic Initiating Event Tree
SDG	Supplemental Diesel Generator
SDS	Shutdown (RCP) Seal
sec.	Second
SEL	Seismic Equipment List
SFP	Spent Fuel Pool
SFR	Seismic Fragility Requirements from ASME/ANS PRA Standard
SGTR	Steam Generator Tube Rupture
SHA	Seismic Hazard Analysis Requirement contained within ASME/ANS PRA
SLERF	Seismic Large Early Release Frequency
SOV	Separation of Variables
SPRA	Seismic Probabilistic Risk Assessment
SPRAIG	Seismic PRA Implementation Guide
SPID	Screening, Prioritization and Implementation Details
SPR	Seismic PRA Modeling Requirements contained within ASME/ANS PRA Standard
SPRA	Seismic Probabilistic Risk Assessment
SQUG	Seismic Qualification Utility Group
SR	Supporting Requirement
SRT	Seismic Review Team

SRP	Standard Review Plan (NUREG-0800)
SSE	Safe Shutdown Earthquake
SSC	Structure, System or Component
SSHAC	Senior Seismic Hazard Analysis Committee
SSI	Soil Structure Interaction
TDAFP	Turbine Driven Auxiliary Feed Pump
TH	Time History
TRA	Plant Transient with power conversion
TRAN	Transient
UB	Upper Bound
UFSAR	Updated Final Safety Analysis Report
UHRS	Uniform Hazard Response Spectra
U.S.	United States
VDC	Volt Direct Current
V/H	Vertical-to-Horizontal
Vs	Shear Wave Velocity
VSLOCA	Very Small Loss of Coolant Accident

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 CNP SPRA was subject to an independent peer review against the pertinent requirements of the PRA standard [4]. The peer review assessment [6], and subsequent disposition of peer review findings, are summarized here. The scope of the review encompassed the set of technical elements and SRs for the SHA, SFR, and SPR elements for seismic CDF and LERF. The peer review therefore addressed the set of SRs identified in Tables 6-4 through 6-6 of the SPID [2].

The information presented in this appendix establishes that the CNP SPRA has been peer reviewed by a team with adequate credentials to perform the assessment, establishes that the peer review process meets the intent of the peer review characteristics and attributes in Table 16 of RG1.200 [11] and the requirements in Section 1-6 of the PRA Standard [4], and presents the significant results of the peer review.

The CNP SPRA peer review was conducted during the week of November 5, 2018, at I&M engineering offices in Buchanan Michigan. As part of the peer review, a walk-down of portions of CNP Units 1 and 2 was performed on November 6, 2018, by two members of the peer review team who have the appropriate training.

A.2 - Summary of Peer Review Process

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

The 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 PRA Standard [4] to ensure the robustness of the model relative to all requirements.

Implementing the review involved 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 was found during the initial review. The SRs provided a structure which, in combination with the peer reviewers' PRA experience, provided the basis for examining the various PRA technical elements. If a reviewer identified a question or discrepancy, additional investigation was conducted until the issue was resolved or an F&O was written describing the issue and its potential impacts, and suggesting possible resolution.

For each area, (i.e., SHA, SFR, or SPR), a team of at least two peer reviewers were assigned, one having lead responsibility for that area. For each SR reviewed, the team reached consensus regarding which of the Capability Categories defined in the Standard that the PRA met for that SR. The assignment of the Capability Category for each SR was ultimately based on the consensus of the full review team. The PRA Standard [4] also specifies HLRs. Consistent with the guidance in the PRA 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. F&Os were prepared as part of the review team's assessment of capability categories. There are three types of F&Os summarized from NEI 12-13 [5]:

- Findings, which identify issues that may need to be addressed in order for an SR (or multiple SRs) to "met" or 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 of this Appendix is on findings and their disposition.

A.3 - Peer Review Team Qualifications

The members of the peer review team were:

Rachel Christian - Westinghouse (Team Lead)
Dr. Annie Kammerer - Annie Kammerer Consulting
Dr. Richard Quittmeyer – RIZZO Associates
Frederic Grant - Simpson Gumpertz & Heger
Dr. Ram Srinivasan - independent technical consultant
Dr. Asa Bassam - SC Solutions
Thomas John – Dominion Generation
Paul Farish - Duke Energy Corporation

The specific qualifications of each team member are as follows:

Ms. Rachel Christian, the team lead, has over 8 years of experience at Westinghouse in the nuclear safety area generally and PRA specifically for both existing and new nuclear power plants. She is engaged in various SPRA projects with Westinghouse and is the primary SPRA model owner for multiple sites. Her peer review experience includes defense of Internal Events, External Events Screening, and Seismic PRA for the AP1000 design plant. Ms. Christian also served as a working observer for an Internal Events PRA peer review and a peer review lead in training for the Sequoyah nuclear plant SPRA peer review.

Dr. Annie Kammerer was the lead for the review of the SHA technical element. Dr. Kammerer is an expert in seismic hazard and risk, and integrated performance based, risk-informed engineering, particularly as applied to nuclear facilities. She has over 15 years of experience. She is an independent consultant, as well as a visiting scholar at the Pacific Earthquake Engineering Research Center at the University of California, Berkeley. She was employed by the NRC for 7 years, where she developed and coordinated the NRC Seismic Research Program. Dr. Kammerer has served on several national and international-level committees and working groups. She led the seismic hazard working group for update of the ASME/ANS JCNRM SPRAs standard. Dr. Kammerer has also served as peer reviewer for multiple SPRAs.

Dr. Richard Quittmeyer has over 35 years in project studies of seismicity and seismic hazards and site characterization. Much of Dr. Quittmeyer's work has been carried out in the nuclear regulatory environment. He managed and provided technical integration for, and peer reviewed hazard studies following the SSHAC process at all four levels of implementation. He has also managed and provided technical integration for large-scale site characterization projects and supervised the 10-year operation of a 45-station network for seismic monitoring. Dr. Quittmeyer has also served as peer reviewer for the Diablo Canyon SPRA PSHA.

Mr. Frederic Grant was the lead for the review of the SFR technical element. He has over 13 years of structural mechanics engineering experience, the majority of which has been in commercial and government nuclear industries. His work in the nuclear industry involves seismic probabilistic risk assessments, seismic fragility analysis, seismic margin assessments, experience-based seismic qualification methods, probabilistic seismic response analysis of structures, and analysis of damage indicating ground motion parameters. He has participated in SPRA peer reviews for North Anna, Indian Point, Watts Bar, and Vogtle nuclear plants and is a member of the ASME/ANS JCNRM Working Group maintaining Part 5 of the PRA Standard [4].

Dr. Ram Srinivasan has 42 years of experience in the nuclear industry, principally in the design, analysis (static and dynamic, including seismic), and construction of nuclear power plant structures, spent fuel cask systems including ISFSI design. Dr. Srinivasan is actively involved in the Post-Fukushima Seismic Assessments (NRC NTTF 2.1 and 2.3) and is a member of the NEI Seismic Task Force. He is also actively involved in the ASME JCNRM Working Group 5 (External Hazards) responsible for the maintenance of the PRA Standard [4] used in the SPRA for Seismic Events. He was the lead SFR Reviewer of the SPRA peer reviews of Diablo Canyon, V.C. Summer, H.B. Robinson nuclear plants, and SFR reviewer for the SPRA peer review of the Callaway nuclear plant and supported the S-PRA Peer Review of the Watts Bar Plant and Sequoyah nuclear plants.

Dr. Asa Bassam has 13 years of structural engineering experience, the majority of which has been in the commercial nuclear industry. His work involves structural analysis and design for dynamic load events, seismic probabilistic risk assessment, soil-structure interaction analysis, ground motion development, seismic fragility, and seismic margin assessments. He is a member of ACI 349 committee for design of nuclear concrete structures and ACI 447 committee for Finite Element analysis of concrete structures. Over the course of his career, he has provided consulting services to more than seventeen nuclear power plants, including the SPRA for the Sequoyah, Watts Bar, Browns Ferry, Peach Bottom, Palisades, and Dresden nuclear plants.

Mr. Thomas P. John was the lead reviewer for the SPR technical element. Mr. John has 27 years of experience in the nuclear industry and 20 years experience in the PRA of Dominion Power nuclear plants. He was the SPRA modeling lead for the Surry nuclear plant pilot SPRA, and is currently the lead for the North Anna nuclear plant SPRA.
Mr. John has supported previous peer reviews including Salem PRA peer review and Indian Point seismic PRA peer review.

Mr. Paul Farish assisted in the review of the seismic plant response technical elements. He has over 25 years of experience in the field of PRA. He has been involved in the development of the Oconee, McGuire and Catawba nuclear plant SPRAs. Mr. Farish served was a SHA reviewer for the Davis-Besse nuclear plant SPRA.

Additionally the team had two working observers: Sami Syed – Enercon, and Mustafa Ozkan - Westinghouse. Any observations and findings these working observers generated were given to the peer review team for their review and "ownership." As such, Sami Syed and Mustafa Ozkan assisted with the review but were not formal members of the peer review team.

I&M reviewed the Peer Review Team's qualifications and confirmed them to be consistent with requirements in the PRA Standard [4] and NEI-12-13 [5] guidelines. The Peer Review Team had no involvement with the development of the CNP SPRA.

A.4 - Summary of Peer Review Conclusions

Section 5 of the PRA Standard [4] contains 95 SRs under three technical elements. Twelve (12) of the SRs were judged to be not applicable, and the remaining 83 SRs were therefore reviewed. Table A.4-1 provides high-level summary of SR review results.

Table A.4-1
Overall Summary of CNP 1 & 2 SR Review

Seismic PRA Element	Number of SRs* Meeting Each Capability Category						
	Not Met	CC-I	CC-II	Met[1]	N/A	Not Reviewed	Total
SHA	2	0	0	30	1	0	33
SFR	1	2	4	13	6	0	26
SPR	1	1	8	21	5	0	36
Totals	4	3	12	64	12	0	95

* This includes all the SRs without any CC differentiation

The review team's conclusions regarding their assessment of the SPRA technical elements are summarized below. Issues identified by the review team were captured in peer review findings. The dispositions of those issues are summarized in section A.6 of this appendix.

A.4.1 - SHA Conclusions

- As required by the PRA Standard [4], the seismic hazard input to the SPRA is determined based on a site-specific PSHA. The site-specific PSHA consists of an Seismic Source Characterization model, a GMC model, and site response evaluations. The PSHA model components must be based on current geological, seismological, and geophysical data; local site topography; and surficial geologic and geotechnical site properties. The CNP PSHA met these requirements and applied structured processes for evaluation; thereby meeting HLRs SHA-A through SHA-E.

- The CNP PSHA:
    - o Used the existing regional Seismic Source Characterization model for the CEUS [19]
    - o Used the regional GMC models from CEUS [19]; and
    - o Evaluated the effects of local site conditions on the ground motions that would be experienced by plant structures, systems and components.

- The existing regional-scale EPRI Seismic Source Characterization model [19] for the CEUS largely meets the intent of the standard for sites in the CEUS. The CNP PSHA gathered and evaluated current earth science information, and new models (developed since the CEUS Seismic Source Characterization model was developed) that could potentially impact the estimate of the seismic hazard at the site.

- With respect to the GMC part of the PSHA, the ground motion prediction equations in EPRI 3002000717 [20] were used because they are suitable for estimating the rock site hazard. Reference [20] is an update of the EPRI SSHAC Level 3 study from 2004. The SSHAC process provides a structured method for the evaluation and integration of available information, including information coming from expert evaluations and provides minimum technical requirements to complete a PSHA.

- The PSHA requirements associated with incorporating the effects of local site conditions on ground motions are defined in SHA-E. The effects of site conditions were modeled by means of amplification factors derived from site response analyses that incorporate site-specific information on surficial geologic deposits and site geotechnical properties. Sources of epistemic uncertainty and aleatory variability were identified, quantified, and carried throughout the site response analyses and PSHA using Method 3 of NUREG/CR-6728 [24].

- HLR-SHA-F addresses the identification, evaluation and determination of relative importance of uncertainties in the SHA evaluations. The evaluations performed and documentation developed addressed the requirements of the PRA standard [4]. SHA-G requires that quantification of the hazard be site specific and a form that supports the SPRA activities. The evaluations performed met the requirements for HLR-SHA-G.

- The PRA Standard [4] requires that a screening analysis be performed to assess whether in addition to vibratory ground motion, other seismic hazards, such as fault displacement, landslide, soil liquefaction, or soil settlement, need to be included in the SPRA. A systematic evaluation was not carried out for other hazards. In the case of liquefaction, the evaluation did not meet the requirements of the PRA standard [4], leading to findings and an assessment of "not met" for SHA-I1 and SHA-I2.

- SHA-J defines the requirements for documentation of the PSHA. The documentation of the SHA is a collection of documents that describes the PSHA methodology, the rock site hazard results, the soil profile hazard results, geotechnical investigation, and the evaluation of other seismic hazards. Overall, the documentation for the PSHA is complete and met the intent of the PRA Standard [4].

A.4.2 – SFR Conclusions

- The seismic fragility analysis performed for CNP meets all SFR SRs of the PRA Standard [4] pertaining to seismic response analysis, fragility evaluation, walkdown, and documentation, except for SFR-B3, which was "not-met."
- Six SFR SRs were not applicable because the fragility analysis did not utilize response scaling, probabilistic response analysis, capacity-based screening, or screening based on inherent ruggedness.
- Two SFR SRs (SFR-E2 and SFR-E3) were assessed to meet CC-I, but not CC-II, because the fragilities for several risk-significant SSCs are conservative and based on generic capacities, and therefore the corresponding failure modes are generic and may be unrealistic.
- The seismic fragility analysis developed fragility curves for all the potentially important SSCs identified by the systems analysis. Fragility documentation lists the median ground motion capacities, epistemic and aleatory variabilities, and associated failure modes.
- To simplify the systems model, quantification, and fragility analysis effort, seismic fragilities were initially developed for large groups of SSCs that are treated as correlated. As quantification iterations identified need for refinement, the fragility groups were broken apart into more realistic subsets and separate fragilities were developed for the potentially important SSCs.
- The fragility analysts provided information that was used to make modeling decisions regarding correlating SSCs in the systems analysis. However, some groups of SSCs that were correlated in the final quantification lacked adequate justification for the correlation modeling approach.
- The structure seismic response analysis included generation of new structure models and performance of SSI analysis. The input motion was defined by a RLE which was found to be reasonably consistent with the ground motion levels that dominate both SCDF and SLERF. However, the soil and structure properties used in the response analyses were in some cases inconsistent with the RLE ground motion level.
- Several structure modeling decisions were not adequately justified, including issues concerning live load masses, openings in walls and slabs, truncation of basement floors, mass and stiffness of non-structural elements, structure cracking and effective stiffness, and structure damping.
- Deterministic SSI analyses were performed to produce median response and associated variability, but they neglected variability associated with structure frequency uncertainty and potential bias introduced by using a single time history.
- The CNP SPRA included a walkdown of the SSCs included in the SEL. The walkdown observations, including the anchorage of components, structural support, systems interaction, seismic vulnerabilities, identification of seismic induced flood and fire sources, etc., are properly documented.
- The walkdown was used to help ensure that the fragility evaluations were realistic, plant specific, and consistent with the as designed, as-built, and as-operated plant conditions.
- Except in a handful of instances, the peer review team assessed that the walkdown generally identified the important failure mechanisms, including functional, structural, and anchorage failure. The walkdown team reviewed seismic interactions, including II/I spatial interactions, seismic-fire, and seismic-flood interactions

- Relevant failure mechanisms were identified for most SSCs. The failure mechanisms identified were based on prior seismic evaluations, walkdown observations, and new analysis. The evaluations considered structural failure mechanisms of buildings, impact of adjacent structures, soil failures, equipment anchorage failures, functional failures of equipment, and seismic interaction failures. However, the failure modes for several risk significant SSCs were unrealistic or generic. Accordingly, several risk-significant SSCs were assigned conservative or generic fragilities, and adequate justification was not provided.
- Non-risk significant SSCs were generally represented by conservative fragility estimates. However, several sources of potential non-conservatism were identified.
- Fragilities were calculated for risk-significant relays, seismic-induced flood sources, and seismic-induced fire sources; fragilities were estimated for those that were not risk-significant. Several sources of conservatism were identified.

A.4.3 – SPR Conclusions

- As required by the PRA Standard [4], the seismic plant-response model developed for CNP meets all of the SPR SRs of the Code Case Standard except for SPR-F3. SPR-F3 was judged "not met" because sources of uncertainty were not documented.
- SPR-E6 was judged to meet CC-I because the FPIE LERF model meets only CCI and therefore the SPRA LERF is considered to be CC-I since it uses the FPIE LERF model
- The development of the SEL, which forms the foundation of the SPRA, was found to be comprehensive and thorough. The SEL database contained the key information on SSCs and was very useful.
- The seismic fire and flood evaluations followed the latest industry guidance in EPRI report 3002012980 [57]. The CNP seismic fire and flood evaluations were part of the pilot for confirming the process of this guidance.
- The relay chatter evaluation was also found to be comprehensive with a number of relay chatter events included in the SPRA model.
- The SPRA model was developed by modifying the FPIE PRA model to incorporate specific aspects of seismic analysis that are different from the FPIE. The logic model appropriately includes seismic-caused initiating events and other failures including seismic-induced SSC failures, non-seismic-induced unreliability and unavailability failure modes (based on the FPIE model), and human errors.
- FRANX was used effectively in the modification of the logic model.
- The grouping and correlation were generally appropriate, although there were some questions about some of the groups having a large number of different types of SSCs.
- The HRA modeling and documentation was judged to meet the supporting requirements.
- The EPRI seismic HRA methodology used the latest industry guidance [52] to adjust the Human Error Probability (HEP) basic events to account for earthquake intensity, time delays, and other performance shaping factors.
- The HRA module in FRANX was used effectively to add the conservative screening HEPs to the model and detailed HRA was performed for the significant HFEs.
- SRs for seismic HRA were judged to meet CCII. However, there were some questions the peer review team had with the determination of significant HEPs since HEPs were set to true in the ACUBE cutset processing.
- The use of CAFTA and ACUBE in the seismic model development and quantification fully met the requirements for integrating a seismic risk model. In these aspects, the

quantification of the CNP SPRA is judged to meet the CCII. Although, there were some quantification results that were not adequately documented such as significant accident sequences.

- The mean CDF and LERF and their distributions were quantified using UNCERT. But Sources of model uncertainty were not adequately documented and characterized. However, a significant number of sensitivities were performed, which provides insights of the impact of the various modeling and screening assumptions.

A.5 - Peer Review Conclusion

The review team concluded that the CNP seismic PRA model integrates the seismic hazard, the seismic fragilities, and the systems-analysis aspects appropriately to quantify core damage frequency and large early release frequency. The seismic PRA analysis was documented in a manner that facilitates applying and updating the SPRA model, but a number of documentation gaps were identified.

A.6 - Peer Review F&O Resolution

Following the Peer Review, CNP focused on resolving F&Os. The first step of this effort was to identify F&Os that could impact the SPRA results and influence significant risk contributors, and to resolve these F&Os or perform an analysis to determine the impact on the SPRA results. The SPRA model and documentation was updated to reflect technical changes made as a result of the F&O resolution process. The discussions and results in Section 5 of this report, reflect the results of the of the F&O resolution effort.

F&Os considered resolved by CNP were subjected to a NRC accepted independent F&O close-out process [66]. This review utilized the NEI 12-13 [5] Appendix X process. This process and outcomes of the F&O close-out process are summarized in this section.

A.6.1 - Summary of Supporting Requirements and Findings

Table A.6-1 lists SRs identified during the Peer Review that where “Not-Met” or did not achieve Capability Category II, and the status of the finding. Section 6.4 below addresses F&Os that remain open after the close-out peer review process. Full details of all original F&Os can be found in Appendix “C” the Peer Review Report [6]. Three new F&Os were created by the close-out team. Details regarding these new F&Os are provided in Appendix C of the F&O close-out review report [64].

Table A.6-1: Summary of SRs Graded as Not-Met or Capability Category I From CNP Peer Review			
SR	**Assessed Capability Category**	**Associated Finding F&Os**	**F&O Disposition to Achieve Met or Capability Category II**
SHA Technical Element			
SHA-I1	Not Met	20-5 20-6	20-5 has been resolved via the close out review process. 20-6 was dissolved by the close-out review team and assigned a new F&O 1-1. This SR is now graded as Met. F&O 1-1 is partially resolved.
SHA-I2	Not Met	20-7	20-7 was dissolved by the close-out review team and assigned a new F&Os 1-1 and 1-2. This SR is now graded as Met. F&O 1-1 is partially resolved. F&O 1-2 has been resolved via the close-out process.
SFR Technical Element			
SFR-B3	Not Met	28-2, 28-3, 28-4, 28-5, 28-6, 28-7, 28-10	28-3 has been dissolved and replaced with new F&O 2-1. 2-1 remains open and will be addressed in future close-out activities. 28-5, 28-6, 28-7, and 28-10 have been resolved through the close-out process. 28-2 and 28-4 are partially resolved and will be addressed in future F&O close-out activities. This SR has been graded as Met through the close-out process.
SFR-E2	CC-I	20-8, 22-5	20-8 has been resolved via the close-out process. 22-5 is technically resolved but remains open for documentation. This SR has been graded as CC1 through the close-out process.
SFR-E3	CC-I	22-2, 22-3	22-2 is technically resolved but remains open for documentation. 22-3 has been resolved via the close-out process This SR has been graded as CCI through the close-out process. This SR has been graded as Met through the close-out process.

Table A.6-1: Summary of SRs Graded as Not-Met or Capability Category I From CNP Peer Review			
SR	**Assessed Capability Category**	**Associated Finding F&Os**	**F&O Disposition to Achieve Met or Capability Category II**
SPR Technical Element			
SPR-B4	Not Met	24-4	24-4 has been resolved via the close-out process and is graded as met.
SPR-E6	CC-I	25-8, 25-9	25-8 has been resolved via the close out process. 25-9 will remain open and this SR graded CC-I until the FPIE model achieves CC-II for LE SRs.

A.6.2 - Summary of Independent Close-out Process

NEI 12-13 [5] Appendix "X" criteria were reviewed by I&M and addressed in recruiting and approving the closure review team members, defining the review scope, and determining the schedule for the review. Reviewer independence was established, approved, and documented in the F&O close-out review report [64]. Reviewer experience meets the criteria specified in NEI 12-13 [5] and the PRA standard [4]. Overall review team experience was such that there were two qualified reviewers for each F&O.

I&M provided the F&O close-out documentation to the independent close-out team prior to the start of the onsite review. This provided sufficient time for the reviewers to prepare and conduct a more efficient technical review. As input to the review, I&M provided a copy of the peer review report, the list of the findings I&M considered resolved, and I&M's resolution of each finding. The on-site review was held August 13, 14, and 15 of 2019. All current and historical SPRA documentation was made available to the Close-out Review Team to facilitate an in-depth review of each F&O considered resolved by I&M.

In accordance with the guidance in NEI 12-13 [5], Appendix X, a lead reviewer and supporting reviewer were assigned for each Technical Element. They assessed each proposed resolved finding for their Technical Element and made the initial determination regarding adequacy of resolution of each finding. The consensus process was followed during which the full team present considered and reached consensus on the adequacy of resolution of each finding using the appropriate SRs of the PRA Standard [4] for the review criteria.

A.6.3 - Independent Close-out Team Qualifications

Mr. Eric Jorgenson was the independent close-out team lead and an SPR reviewer. The close-out team and review areas were as follows:

- Dr. Osman El-Menchawi – SHA
- Alfredo Fernandez – SHA
- Kenneth Whitmore – SFR
- Sami Syed – SFR
- Steven Phillips – SPR
- Heather Morgan – SPR

The qualifications of the close-out team members were as follows:

Mr. Eric Jorgenson has over 30 years of experience performing PRAs for BWR and PWR nuclear plants. He has over twenty years of experience performing and developing external and spatially-dependent event PRA models (fires, floods, earthquakes, other external events). Mr. Jorgenson was the lead responsible engineer for the upgrade of the Columbia seismic PRA to address Fukushima Task Force Recommendation 2.1.

Dr. Osman El-Menchawi has over 20 years of extensive geotechnical consulting, seismic hazard, and construction experience. Dr. El Menchawi is the Vice President of Western United States and Nuclear Services managing Fugro's offices in California and Colorado, which represent FUGRO's Nuclear Center of Excellence in the Americas, in addition to earthquake engineering and geo-hazard assessment servicing of the international market, and geotechnical engineering serving the local region. He has extensive experience performing and managing seismic hazard evaluations at various sites including but not limited to: Browns Ferry, Sequoyah, Peach Bottom, Dresden, Comanche Peak Units 3 and 4, and Savannah River.

Dr. Alfredo Fernandez has over 10 years of experience in managing and performing seismic geotechnical calculations, including site response analyses, PSHA for soil conditions, development of design response spectra and associated ground motion time histories, SSIs, liquefaction hazards, and nonlinear finite element/finite difference methods. He is also an author or co-author of multiple papers focusing on the latest updates in the fields of SSI and PSHA, and has participated in the further development of FUGRO's FRISK88 code under nuclear Quality Assurance programs. Dr. Fernandez has full understanding of the ASME standards and SPID [2] as related to the SHA. Dr. Fernandez's seismic geotechnical experience includes applications to nuclear power plants, dams, bridges, oil and gas processing plants, port and harbor facilities, and offshore platforms.

Mr. Kenneth Whitmore is a Registered Professional Engineer with more than 30 years of experience in civil/structural analysis and design, including seismic analysis of structures, systems and components. He is a recognized expert in seismic qualification of equipment and structures using experience data, including use of SQUG methodology, as well as traditional analytical and test methods. He also has expertise in the evaluation and design of post-installed anchors in concrete. Mr. Whitmore was the overall technical lead for seismic PRAs at Peach Bottom, Dresden, Sequoyah and Brown's Ferry nuclear plants, and a subject matter expert for a seismic PRA at Columbia nuclear plant. He has performed peer review closure for Fermi nuclear plant and defended the Sequoyah nuclear plant SPRA peer review with no open items and no required follow-up actions.

Mr. Sami Syed has more than ten years of experience in structural engineering, structural dynamics, performance-based earthquake engineering, finite element modeling, linear and nonlinear response analyses of buildings and engineering systems, seismic risk assessment, seismic fragility analysis and statistical modeling. Mr. Syed is trained in SQUG walkdown screening and seismic evaluation procedures and is a certified SQUG seismic capability engineer. He has led SPRA and Mitigating strategy Assessment walkdowns for both BWR and PWR power plants. Mr. Syed has participated in industry peer reviews led by PWR and BWR Owner's Groups, serving various roles of defending

SPRA projects, performing peer reviews and closure reviews. Mr. Syed was a working observer of the CNP SPRA peer review.

Mr. Steven Phillips has 17 years of experience as an analyst on PRAs for nuclear power plants. He performed seismic PRA modeling and analyses of system components with respect to seismic events for the Vogtle, Hatch, and Farley nuclear plants using FRANX software and KEPCO's APR1400 plant design using SAREX software. He also performed reviews for the development of the SEL for the Sequoyah nuclear plant.

Ms. Heather Morgan has over seven years of experience in the nuclear industry; her primary focus during that time has been electrical circuit analysis. Ms. Morgan has conducted relay chatter originations and reviews to support Seismic PRA efforts for Exelon. While employed with E Group Engineering, Inc., Ms. Morgan was assigned as a member of a team working to develop Seismic PRA Models for each of Southern Nuclear Operating Company (SNC)'s three Nuclear Plants. Work included developing the Seismic Equipment List (SEL) using available system documentation and training materials. During this time Ms. Morgan also assisted in preliminary relay reviews. Ms. Morgan also provided electrical engineering services to SNC. She performed Circuit Analysis of the Electrical Systems of SNC-owned and operated Nuclear Power Plants.

I&M reviewed the resumes of the review team and found all reviewers to be competent and qualified to perform the independent close-out review. I&M also verified that the review team was independent of the CNP SPRA development.

A.6.4 - Independent Close-out Team Conclusions

As indicated in Table A.6-1, the Close-out Review Team dissolved three (3) F&Os: 20-6, 20-7, and 28-3. The Close-out Review Team created three (3) new F&Os (1-1, 1-2, and 2-1) that replaced the dissolved F&Os. F&O 1-1 is applicable to both SRs SHA-I1 and I2, F&O 1-2 is applicable to SHA-I2 only, and F&O 2-1 is applicable to SR SFR-B3. The SRs associated with the new F&Os were classified as "upgrades" that require a focused peer review. The close-out team performed the focused review for these new F&Os, and is documented in the F&O close-out report [64]. All new F&Os are graded as "Met."

Table A.6-4 presents a summary of the F&Os that remain open after the close-out process. Table A.8 summarizes the impact that open F&Os may have on the SPRA results.

Table A.6-4: Summary of CNP Open F&Os			
SR	**Assessed Capability Category**	**Associated Finding F&Os**	**F&O Disposition to Close Open F&O**
SHA Technical Element			
SHA-I1 SHA-I2	Met	1-1	This F&O had 9 subparts. 6 of the 9 have been resolved through the focused review process. I&M does not intend to take action on the remaining open subparts.
SHA-J2	Met	20-3	20-3 has been technically resolved and remains open only for documentation revisions. Additional documentation revisions are not planned at this time.
SFR Technical Element			
SFR-B3	Met	2-1	I&M plans on addressing this item in the future via a sensitivity study of the AB concrete cracking as suggested in the F&O close-out report [64].
SFR-B3	Met	28-2	I&M plans on addressing this item in the future via a sensitivity study of the AB concrete cracking as suggested in the F&O close-out report [64].
SFR-B3	Met	28-4	This F&O is tied to resolution of 2-1 above, and will be addressed with 2-1.
SFR-B4	Met	28-11	I&M plans on addressing this item in the future via a sensitivity study of a shift of structural peak response as suggested in the F&O close-out report [64].
SFR-B4	Met	28-13	I&M plans on addressing this item in the future via a sensitivity study of a shift of structural peak response as suggested in the F&O close-out report [64].
SFR-E2	CC-II	22-5	22-5 has been technically resolved and remains open only for documentation revisions. Additional documentation revisions are not planned at this time.
SFR-E3	CC-II	22-2	22-2 has been technically resolved and remains open only for documentation revisions. Additional documentation revisions are not planned at this time.
SFR-F2	Met	28-19	28-19 has 14 subparts. 13 of the 14 subparts have been resolved. The 1 remaining subpart is documentation only. Additional documentation revisions are not planned at this time.

Table A.6-4: Summary of CNP Open F&Os			
SR	**Assessed Capability Category**	**Associated Finding F&Os**	**F&O Disposition to Close Open F&O**
SPR Technical Element			
SPR-E3	CC-II	25-7	25-7 has 5 subparts. 4 of the 5 subparts have been resolved. The 1 remaining subpart is documentation only. Additional documentation revisions are not planned at this time.
SPR-E6	CC-I	25-9	This F&O will remain open and CC-I as it is driven by the FPIE model CC-I grade for LE SRs.

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

Supporting requirements from the PRA Standard [4] that are identified in Tables 6-4 through 6-6 of the SPID [2] define the technical attributes of a PRA model required for a SPRA used to implement the requirements of the 10 CFR 50.54(f) [1] letter. The conclusions of the peer review discussed above and summarized in this report demonstrates that the CNP SPRA model meets the expectations for PRA scope and technical adequacy as presented in RG 1.200 [11] as clarified in the SPID [2].

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

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

Detailed archival information for the SPRA consistent with the listing in Section 4.1 of RG 1.200 [11] is available if required to facilitate the NRC staff's review of this report.

The CNP SPRA reflects the as-built and as-operated plant as of the cutoff date for the SPRA, March 5, 2018. Beyond Design Basis equipment and mitigation strategies (FLEX) were not credited in the SPRA, except for the upgraded (leakage control) RCP seals. These seals are permanently installed plant equipment. All other permanent plant changes that could affect the SPRA results have been reflected in the SPRA model.

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

The PWR Owners Group performed a full scope peer review of the CNP 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 [26] and RG 1.200 [11]. FPIE peer review findings were documented for all SRs which failed to meet at least Capability

Category II, as noted in FPIE Peer Review Report [60]. All of the internal events and internal flooding PRA peer review findings that may affect the SPRA model have been addressed, as noted in Section 5.1 of this report.

The PWR Owners Group performed a peer review of the CNP SPRA in November 2018. The results of this peer review are discussed above, including resolution of SRs assessed by the peer review as not meeting CC-II, and resolution of peer review findings pertinent to this report. The peer review team expressed the opinion that the CNP SPRA model is of good quality and integrates the seismic hazard, the seismic fragilities, and the systems-analysis aspects which appropriately quantify core damage and large early release frequency. The general conclusion of the peer review was that the CNP SPRA is suitable for use for risk-informed applications.

- Table A.6-1 provides a summary of the disposition of SRs judged by the peer review team to be not met, or not meeting CC-II.
- Table A.6-4 provides a summary of the disposition of F&Os remaining open after the NEI 12-13 Close out peer review.
- Table A.8 provides an assessment of the expected impact on the SPRA results of those F&O remaining open after the NEI 12-13 Close-out peer review.

Table A.8-1: Summary of CNP Open F&Os Impact to SPRA Results		
F&O	**Summary of Issue Not Fully Resolved**	**Impact on SPRA Results**
	SHA Technical Element	
1-1	Partially Resolved 1) - Only a single method was considered to evaluate the liquefaction triggering potential, liquefaction susceptibility, and post-liquefaction volumetric strains. However, in F&O 20-7 item 2, more than one method was requested to conduct the liquefaction hazard evaluation as "the choice of any single method does not address the epistemic uncertainty in the field (which is the underlying motivation of a recent National Academy study and report)". 2) - Lateral spreading hazard at the site does not address the evaluation of this potential hazard. Lateral spreading can occur in slope gradients as flat as 0.5% (without a free face) (see NA report). Additionally, Figures 6-7 and 6-9 shows that there is continuous layer of potentially liquefiable soils (in direction towards the lake) on borings B120, B124, B133, B142, and B141 between elevations of about 560 and 555 ft. Therefore, the potential of lateral spreading and/or flow slides at the site should be evaluated. 3) - Provide a full reference to all citations included in the report.	SPRA results are not expected to be impacted. I&M (2014) initially performed a liquefaction triggering (using Youd et al., 2001) and settlement (using Tokimatsu and Seed, 1987) analyses using the RLE and obtained comparable results. I&M considers that Figure 6-7 shows liquefaction at some boreholes for 1E-6 motions, but shows no lateral continuity of the liquefiable boreholes. Based on this information, I&M has concluded that the site can be screened out for site-wide lateral spreading.
20-3	Resolved with open documentation 1)- Include additional justification on why V/H ratios should be used instead of vertical GMPEs in report DC COOK-PR-02, Section 7.1 (e.g. inconsistency of controlling earthquakes between horizontal and vertical spectra if vertical GMPEs were used). 2) - Perform a thorough editorial review of the reference citations and list of references.	SPRA results are not expected to be impacted as this F&O has been technically resolved.
	SFR Technical Element	
2-1	Open While the cracking assessment for the AB and TB/SH has been resolved the cracking assessment for Auxiliary Building (AB) has not been fully resolved. Several changes were made to the AB structural model in 15C4313-CAL-010, "Response Analysis of Auxiliary Building," Revision 2 in response to other SFR	SPRA results are not expected to be impacted. The studies performed in 15C4313-RPT-003 "Summary of Building Response Analysis for the Cook Nuclear Plant (CNP) Unit 1 & Unit 2 SPRA," Attachment E show that while there may be some

	F&Os. The updated AB model was used in the cracking assessment with un-cracked section properties. The SPRA team performed cracking assessment at earthquake levels corresponding to 0.5*RLE and 1.0*RLE. Figures 1 through 8 in Attachment E present the shear stress contour plots on isometric views of the AB model showing the exterior walls. The stress contour plots only suggest that the building is overly stressed in certain regions. For a complex structure such as the AB, this is not sufficient to conclude that cracking will or will not occur in the building especially under dynamic loads. The SPRA development team has not assessed or documented the cracking assessment for the AB interior walls in a way that resolves the concern identified in the initial F&O issued by the peer review team.	cracking, it is not widespread at the RLE-level. Additionally, I&M engineering judgement is that with the studies performed, cracking in the structure will decrease the stiffness and increase the damping. These two effects tend to affect the structural response in opposite ways. Finally, many significant contributors have low fragilities for which consideration of a cracked model would be non-conservative.
22-2	Resolved with open documentation Perform a sensitivity study to address items determined to be risk significant based on F-V importance greater-than or equal-to 0.005.	SPRA results are not expected to be impacted. I&M position is that additional studies for risk-items not considered by risk significant as (defined in the SPRA quantification notebook [12]) will not change risk insights.
22-5	Resolved with open documentation Perform a sensitivity study to address items determined to be risk significant based on F-V importance.	SPRA results are not expected to be impacted. I&M position is that sensitivity studies documented in the SPRA quantification notebook [12] envelope any small fragility changes that may be discovered by the additional sensitivity recommend here and will not change risk insights.
28-2	Partially Resolved SPRA team has used the ASCE 4-16 [32], Section 3.7.2 dynamic coupling criteria for single-point attachment to show that the current CB modeling approach and response are realistic. While the modeling approach use probably does not have an effect on overall response of the structure but that conclusion has not been demonstrated adequately.	SPRA results are not expected to be impacted. The I&M position is that the simplified method used to demonstrate that the CB modelling simplifications have no impact on the response in 15C4313-RPT-003 Attachment B is sufficient to address the F&O. The close-out team requested more detailed studies be performed to close the F&O, however the team stated that they believe the conclusion will most likely not change as a result.

28-4	Partially Resolved Appropriate damping was used for cracked and un-cracked building sections in the building response sensitivity studies following the current industry and standard ASCE 4-16 [32]. The sensitivity practice studies are documented in Attachments B and F of 15C4313-RPT-003, respectively, for Containment Building and Turbine Bulding/Screen House. Appropriate damping is also used for AB response analysis model documented in 15C4313-CAL-010 Rev. 2. However, the focused scope peer review F&O 2-1 would require to reassess the cracking assessment of AB and appropriate damping should be used if cracking is assessed to be of significance.	SPRA results are not expected to be impacted. The position of I&M is that the conclusion provided in 15C4313-RPT-003 Attachment E is sufficient to justify the use of un-cracked damping for the AB model. See F&O 2-1 for further information.
28-11	Open The SPRA development team added an argument that due to the way that fragilities were developed, including the application of uncertainty with respect to frequency was sufficient to allow no variation in structural properties. The variation in frequency is intended to reflect uncertainty in the value of the calculated frequency. The variation in structural properties is intended to reflect uncertainty in those properties. Both effects must be considered when developing fragilities.	SPRA results are not expected to be impacted. The sensitivity studies performed in 15C4313-RPT-003 between un-cracked and cracked properties show that structural variability has a minor impact on response compared to the soil property variability. I&M will review the small number of impacted risk-significant components on a case by case basis, adjusting the FROI by an additional +/-15% to ensure structural variability is captured in the fragility calculations.
28-13	Partially Resolved The gap in PSD as described in the F&O should be addressed per latest fragility guidance document. If it is confirmed that there is a gap in PSD at FROI of structure, then it is recommended to perform a sensitivity study to assess the impact of the gap in energy. The SPRA development team can perform this by comparing the PSD functions of the five time histories that were generated by resolution of F&O 28-09 to the PSD function of the artificial time history, or the development team can integrate the PSD function to show that a smooth curve is generated.	SPRA results are not expected to be impacted. I&M position is that there are not significant gaps in energy near frequencies that are important to risk-significant fragilities. The PSDs as presented were developed using a logarithmic frequency interpolation which tends to emphasize magnitude variation at low frequencies. A review of the non-interpolated PSDs and PSDs developed using a linear frequency interpolation supports the determination that the gaps identified in the F&O are not significant.
28-19	Resolved with open documentation items	SPRA results are not expected to be impacted, as

	The documentation needs to be further updated to provide a basis for not considering SSSI effects. Subsequent to the closure review, additional documentation was added to the calculations. However, the closure review team does not consider this additional documentation to be sufficient to address the concern originally identified.	this F&O has been technically resolved. Additional quantitative justification added to Section 4.4 of 15C4313-RPT-003 is adequate in showing that SSSI effects do not control over RLE demand for applicable components. Also note that components associated in this documentation item are not risk-significant.
SPR Technical Element		
25-7	Resolved with Open Documentation In the SPRA Model Quantification Notebook, Section 8.2.2, Revision1, the cutset review included a statement on non-significant cutsets – samples are covered by the examination of G1 and G2 bins. G1 and G2 bins contain relatively fewer seismic-induced failures and the cutsets have features more like the internal events PRA. A recommendation is made to expand the review to other ground motion bins so that model logic related specifically to the SPRA can be confirmed to be appropriate and as intended.	SPRA results are not expected to be impacted as this F&O has been technically resolved.
25-9	Open Some of the supporting internal events LE SRs were met at CC-I only; therefore, this SR is met for CC-I only.	No impact to SPRA results – The LERF modeling is built upon the internal events LERF model and is essentially unchanged. The SPRA LERF model includes seismic-specific aspects such as unique containment failure probabilities. Whereas there are some internal events LE supporting SRs that meet both CC-I and CC-II, the majority of the SRs are met at CC-I. Therefore, this SR is considered to be met at CC-I only.

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

The PRA Standard [4] includes a number of requirements related to identification and evaluation of the impact of assumptions and sources of uncertainty on the PRA results. NUREG-1855 [13] and EPRI 1016737 [14] provide guidance on assessment of uncertainty for applications of a PRA. As described in NUREG-1855, sources of uncertainty include "parametric" uncertainties, "modeling" uncertainties, and "completeness" (or scope and level of detail) uncertainties.

- Parametric uncertainty was addressed as part of the CNP 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 CNP 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 of this report.

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

A summary of potentially important sources of uncertainty in the CNP SPRA is listed in Table A.9-1.

Table A.9-1
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 CNP SPRA peer review team noted that both the aleatory and epistemic uncertainties were identified, quantified, and carried throughout the site response analyses and PSHA using Method 3 of NUREG/CR-6728 [24].	The seismic hazard reasonably reflects sources of uncertainty.
Seismic Fragilities	Deterministic SSI analyses were performed to produce median response and associated variability, but they neglected variability associated with structure frequency uncertainty and potential bias introduced by using a single time history. The processes involved in developing the fragilities provided a general discussion of sources of uncertainty and related assumptions is Report 15C4313-RPT-007 [40].	As part of F&O resolution, additional analysis was performed, comparing the use of a single time history with additional "ad-hoc" time histories. This comparison showed insignificant changes to the results. The comparison was reviewed by the independent review team and found to be acceptable. Therefore, fragility uncertainty is reasonably reflected in the SPRA.
Seismic PRA Model	Sources of model uncertainty are documented and characterized in Reference 12, including a significant number of sensitivities which provide insights of the impact of the various modeling and screening assumptions.	The CNP SPRA is judged to have the appropriate level of detail to address the foreseeable applications. Sensitivity calculations include variations in Fragility parameters, recovery actions (e.g., for hydrogen ignitors), and overall modeling approaches (e.g., hazard interval definitions). These sensitivity calculations provide a basis to judge particular impacts on SPRA results and are discussed in Section 5.7.

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

The CNP SPRA reflects an as-built / as-operated of date May 8, 2018. Plant modifications are reviewed by the PRA group to determine potential impacts on the plant risk analysis. An evaluation was performed of pending modifications that were initiated after May 8, 2018, and determined to potentially impact a site PRA model.

Table A.10-1 lists those plant modifications and provides a qualitative assessment of the likely impact of those changes on the SPRA results and insights. As detailed in that table, there have been no plant modifications made since the SPRA cutoff date that would affect the results of the SPRA.

Table A.10-1
Summary of Significant Plant Changes Since SPRA Cutoff Date

Description of Plant Change	Impact on SPRA Results
EC-54649 – Replace emergency power transformers 12-TR12EP 1 and 2	No effect, as all the equipment is functionally being replaced with similar designs or is being removed. PRA impact evaluated under GT 2016-13860.
EC-54611 – Containment Hydrogen Recombiners, abandon in place	No effect hydrogen recombiner are not credited in SPRA. PRA impact evaluated under GT 2016-2293.
EC-54911 - Bellow seal removal on NESW piping and removal of protective covering to 1-CPN-85. A temp mod was also installed over the removed NESW piping	The mod resulted in a functionally similar design for this containment penetration, and thus there was no model impact. PRA impact evaluated under GT 2016-4232.
EC-54959 - Changes the normal position for several containment isolation valves to open instead of closed: 1(2)-DCR-201 and 203	The mod resulted in a functionally similar design for this containment penetration, and thus there was no model impact. PRA impact evaluated under GT 2016-6325.
EC-55458 – Unit 2 up-flow conversion	MAAP models are simplified enough so that the up-flow conversion has no effect on the analysis for the PRA model. PRA impact evaluated under GT 2017-3234.
EC-54651 – MUP system upgrade	MUP is not modeled in PRA. PRA impact evaluated under GT 2017-5028.
EC-55034 – Unit 1 reserve feed transformers 1-TR101AB and CD and main generator step-up transformer 1-TR-Main, Single phase detection and protection	No impact, cable failure are not a risk contributor to SPRA. PRA impact evaluated under GT 2017-6398.
EC-55035 – Unit 2 reserve feed transformers 2-TR201AB and CD and main generator step-up transformer 2-TR-Main, Single phase detection and protection	No impact, cable failure are not a risk contributor to SPRA. PRA impact evaluated under GT 2017-9196.

Table A.10-1 Summary of Significant Plant Changes Since SPRA Cutoff Date	
Description of Plant Change	**Impact on SPRA Results**
EC-55973 – Temporary installation of pumps to support the MUP	The MUP is not modeled in PRA. PRA impact evaluated under GT 2018-2832.
EC-56066 – Unit 1 up-flow conversion	MAAP models are simplified enough so that the up-flow conversion has no effect on the analysis for the PRA model. PRA impact evaluated under GT 2018-5632.
EC-53363 – Radiation monitor system upgrade	No impact as the RMS is not modeled in SPRA. PRA impact evaluated under GT 2018-6616.
EC-53364 – Radiation monitor system upgrade	No impact as the RMS is not modeled in SPRA. PRA impact evaluated under GT 2018-6616.