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Serial: RA-19-0452
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10 CFR 50.4
10 CFR 50.54(f)

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

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261 / RENEWED LICENSE NO. DPR-23

SUBJECT: H.B. Robinson Steam Electric Plant, Unit No. 2 – Seismic Probabilistic Risk Assessment (SPRA), Response to March 12, 2012, Request for Information Regarding Recommendation 2.1: Seismic, of the Near-Term Task Force Related to the Fukushima Dai-ichi Nuclear Power Plant Accident

REFERENCES:

1. NRC Letter, *Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3 of the Near-Term Task Force review of Insights from the Fukushima Dai-ichi Accident*, dated March 12, 2012 (ADAMS Accession No. ML12053A340)

2. EPRI 1025287, *Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic*, dated November 2012 (ML12333A170)

3. Duke Energy Letter, *Submittal of Revision to Seismic Hazard Evaluation to Include New Ground Motion Response Spectra (GMRS) Using New Geotechnical Data and Shear-Wave Testing for H. B. Robinson Steam Electric Plant, Unit No. 2*, dated July 17, 2015 (ML15201A006)

4. NRC Letter, *H. B. Robinson Steam Electric Plant, Unit No. 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 Dai-ichi Accident*, dated October 19, 2015 (ML15280A199)

5. NRC Letter, *Final Determination of Licensee Seismic Probabilistic Risk Assessments Under the Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendation 2.1: Seismic, of the Near-Term Task Force Review of Insights from the Fukushima Daiichi Accident*, dated October 27, 2015 (ML15194A015)

6. Duke Energy Letter, *Request for Extension of Due Date for Seismic Probabilistic Risk Assessment Submittal*, for H.B. Robinson Steam Electric Plant, Unit No. 2, dated November 29, 2018 (ML18337A159)

7. NRC Letter, *H. B. Robinson Steam Electric Plant, Unit No. 2 - Response to Request for Extension of Seismic Probabilistic Risk Assessment Submittal (EPID No. L-2018-JLD-0017)*, dated January 10, 2019 (ML19004A356)

8. Duke Energy Letter, *Request for Extension of Due Date for Seismic Probabilistic Risk Assessment Submittal*, for H.B. Robinson Steam Electric Plant, Unit No. 2, dated October 21, 2019 (ML19294A028)

9. NRC Letter, *H. B. Robinson Steam Electric Plant, Unit No. 2 – Response to Request for Extension of Due Date for Seismic Probabilistic Risk Assessment Submittal (EPID NO. L-2019-JLD-0014)*, dated October 28, 2019 (ML19296C623)

Ladies and Gentlemen,

On March 12, 2012, the U.S. Nuclear Regulatory Commission (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) (i.e., Reference 1), requesting 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.

Industry guidance was developed by EPRI that provided the screening, prioritization and implementation details (SPID) for the resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic. The SPID (i.e., Reference 2) was used to compare the reevaluated seismic hazard to the design basis hazard. The H.B. Robinson Steam Electric Plant (HBRSEP) Unit No. 2, reevaluated seismic hazard (i.e., Reference 3) concluded that the ground motion response spectrum (GMRS) exceeded the design basis seismic response spectrum in the 1 to 10 Hz range, and therefore a seismic probabilistic risk assessment was required.

Reference 4 contains the NRC Staff Assessment of the HBRSEP Unit No. 2 reevaluated seismic hazard submittal and confirmed the conclusion that the GMRS for the Robinson site exceeds the design basis seismic response spectrum and a seismic risk evaluation is merited.

Reference 5 contains the NRC letter for the final determination of licensee seismic probabilistic risk assessments. In that letter, the NRC instructed HBRSEP Unit No. 2, to submit an SPRA by March 31, 2019.

Duke Energy requested an extension to that due date (i.e., Reference 6), and the NRC approved the due date extension to October 31, 2019 (i.e., Reference 7). Duke Energy requested a second extension request to the SPRA submittal due date (i.e., Reference 8), and the NRC approved the due date extension to December 12, 2019 (i.e., Reference 9).

Enclosure 1 of this letter contains the HBRSEP, Unit No. 2, Seismic Probabilistic Risk Assessment (SPRA) Summary Report which provides the information requested in Enclosure 1, Item (8) B. of Reference 1.

This letter contains no new Regulatory Commitments.

Should you have any questions regarding this submittal, please contact Art Zaremba – Director, Fleet Licensing, at (980) 373-2062.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 12, 2019.

Sincerely,

Ernest J. Kapopoulos, Jr.
Site Vice President

LJG/ljg

Enclosure: H.B. Robinson Steam Electric Plant, Unit No. 2, Seismic Probabilistic Risk Assessment (SPRA) in Response to 50.54(f) Letter Regarding NTTF 2.1: Seismic Summary Report

cc (with enclosure)

L. Dudes, Regional Administrator USNRC Region II
A. Hon, NRR Project Manager – RNP
M. Fannon, NRC Senior Resident Inspector – RNP

Enclosure

H.B. Robinson Steam Electric Plant, Unit No. 2, Seismic Probabilistic Risk Assessment (SPRA) in Response to 50.54(f) Letter Regarding NTTF 2.1: Seismic Summary Report

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

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

SUMMARY REPORT

December 2019

Table of Contents

Executive Summary

In response to the 10 CFR 50.54(f) letter issued by the NRC on March 12, 2012, a seismic PRA (SPRA) has been developed to perform the seismic risk assessment for H. B. Robinson Steam Electric Plant, Unit No. 2. The SPRA model shows the point estimate seismic Core Damage Frequency (SCDF) is $9.27x10^{-5}$/reactor-year and the seismic Large Early Release Frequency (SLERF) is $2.02x10^{-5}$/reactor-year. The SPRA reflects the as-built/as-operated Robinson Nuclear Power Plant as of the freeze date for the internal events model (June 2015). An assessment is included in Appendix A of the impact of the results of plant changes not included in the model since the model freeze date.

Due to the insights gained from the seismic risk assessment, Robinson plans to implement a means to provide Auxiliary Feedwater supplied by a modified FLEX strategy. The results of the corresponding sensitivity analysis show that the SCDF and SLERF can be reduced by approximately 40 percent and 30 percent, respectively. This modification will be implemented by the end of 2022.

1.0 Purpose and Objective

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

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

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

This report provides summary information regarding the SPRA as outlined in Section 2. The level of detail provided in the report is intended to enable NRC to understand the inputs and methods used, the evaluations performed, and the decisions made as a result of the insights gained from the Robinson Nuclear Power Plant seismic PRA.

2.0 Information Provided in This Report

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

(1) The list of the significant contributors to SCDF for each seismic acceleration bin, including importance measures (e.g., Risk Achievement Worth, Fussel-Vesely and Birnbaum)

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

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

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

Note that 50.54(f) letter Enclosure 1 paragraphs 1 through 6, regarding the seismic hazard evaluation reporting, also apply, but have been satisfied through the previously submitted Robinson Nuclear Power Plant Seismic Hazard Submittal [4]. Further, 50.54(f) letter Enclosure 1 paragraph 9 requests information on the Spent Fuel Pool. Duke submitted the Spent Fuel Pool Supplemental Report to the NRC for H. B. Robinson Nuclear Power Plant [57] and has received the final staff assessment [60].

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

The SPID [2] defines the principal parts of an SPRA, and the H. B. Robinson Nuclear Power Plant SPRA has been developed and documented in accordance with the SPID. The main elements of the SPRA performed for H. B. Robinson Nuclear Power Plant in response to the 50.54(f) Seismic letter correspond to those described in Section 6.1.1 of the SPID, i.e.:

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

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

The Robinson Nuclear Power Plant SPRA and associated documentation has been peer reviewed against the PRA Standard in accordance with the process defined in NEI 12-13 [6] as documented in the Robinson Nuclear Power Plant SPRA Peer Review Report. The

Robinson Nuclear Power Plant SPRA, complete SPRA documentation, and details of the peer review are available for NRC review.

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

The content of this report is organized as follows:

- Section 3 provides information related to the Robinson Nuclear Power Plant seismic hazard analysis.
- Section 4 provides information related to the determination of seismic fragilities for the Robinson Nuclear Power Plant SSCs included in the seismic plant response.
- Section 5 provides information regarding the plant seismic response model (seismic accident sequence model) and the quantification of results.
- Section 6 summarizes the results and conclusions of the SPRA, including identified plant seismic issues and actions taken or planned.
- Section 7 provides references.
- Section 8 provides a list of acronyms used.

Appendix A provides an assessment of SPRA Technical Adequacy for Response to NTTF 2.1 Seismic 50.54(f) Letter, including a summary of the Robinson Nuclear Power Plant SPRA peer review.

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

50.54(f) Letter Reporting Item	Description	Location in this Report
1	List of the significant contributors to SCDF for each seismic acceleration bin, including importance measures	Section 5
2	Summary of the methodologies used to estimate the SCDF and LERF	Sections 3, 4, 5
2i	Methodologies used to quantify the seismic fragilities of SSCs, together with key assumptions	Section 4
2ii	SSC fragility values with reference to the method of seismic qualification, the dominant failure mode(s), and the source of information	Tables 5.4-2 and 5.5-2 provide fragilities (A_m, β_r and β_u for fragilities following a log-normal distribution), failure mode information, and method of determining fragilities for the top risk significant SSCs based on Fussel-Vesely (F-V).
2iii	Seismic fragility parameters	Tables 5.4-2 and 5.5-2 provide fragilities information (A_m, β_r and β_u for fragilities following a log-normal distribution) for the top risk significant SSCs based on Fussel-Vesely (F-V).
2iv	Important findings from plant walkdowns and any corrective actions taken	Section 4.2 addresses walkdowns and walkdown insights.
2v	Process used in the seismic plant response analysis and quantification, including specific adaptations made in the internal events PRA model to produce the seismic PRA model and their motivation	Sections 5.1 and 5.2 provide this information.
2vi	Assumptions about containment performance	Sections 4.3 and 5.5 address containment and related SSC performance.
3	Description of the process used to ensure that the SPRA is technically adequate, including the dates and findings of any peer reviews	App. A describes the assessment of SPRA technical adequacy for the 50.54(f) submittal and results of the SPRA peer review.
4	Identified plant-specific vulnerabilities and actions that are planned or taken	Section 6 addresses this.

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

SPID Section 6.8 Item (1) Description	Location in this Report
A report should be submitted to the NRC summarizing the SPRA inputs, methods, and results.	Entirety of the submittal addresses this.
The level of detail needed in the submittal should be sufficient to enable NRC to understand and determine the validity of all input data and calculation models used	Entirety of the submittal addresses this. It identifies key methods of analysis and referenced codes and standards
The level of detail needed in the submittal should be sufficient to assess the sensitivity of the results to all key aspects of the analysis	Entirety of the submittal addresses this. Results sensitivities are discussed in the following sections: • 5.7 (SPRA model sensitivities) • 4.4 Fragility screening (sensitivity)
The level of detail needed in the submittal should be sufficient to make necessary regulatory decisions as a part of NTTF Phase 2 activities.	Entirety of the submittal template addresses this.
It is not necessary to submit all of the SPRA documentation for such an NRC review. Relevant documentation should be cited in the submittal, and be available for NRC review in easily retrievable form.	Entire report addresses this. This report summarizes important information from the SPRA, with detailed information in lower tier documentation
Documentation criteria for a SPRA are identified throughout the ASME/ANS Standard [5 and 37]. Utilities are expected to retain that documentation consistent with the Standard.	This is an expectation relative to documentation of the SPRA that the utility retains to support application of the SPRA to risk-informed plant decision-making.

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

3.0 Robinson Nuclear Plant Seismic Hazard and Plant Response

Section 3.0 provides a high-level summary site description for the H. B. Robinson Steam Electric Plant (HBRSEP). The subsections provide brief summaries of the site hazard and response characterizations for the HBRSEP and the Lake Robinson Dam as well as discussions of the potential liquefaction evaluation and impacts.

The HBRSEP is a soil site located in Darlington County South Carolina. The following description of the general geology at the site is adapted from [16]:

> The Robinson Plant is within the Coastal Plain physiographic province in South Carolina and within the upper portion of that province. At the western edge of the Coastal Plain, which is approximately 15 miles northwest of the site, pre-Cambrian basement rock of the Piedmont physiographic province is exposed. In the Piedmont, the basement rocks are covered with soil-like material weathered in place from the original granitic rocks, and the UFSAR indicates this weathered material may be present below the Coastal Plain formations at the site as well.
>
> The Middendorf Formation of Cretaceous age overlies the Piedmont at the site. The Middendorf Formation was formed by deposition of sediments transported by water from the west. A fluvial to deltaic depositional environment is described by Sohl and Owens, [17]. Both types of depositional environments are characterized by lateral and vertical variations in soil layers, both in composition and thickness. Such variations were observed in the boring logs from historical and current explorations. Overall, the Middendorf is described as a sequence of alternating clay and sand layers. The sand layers vary from clean sands with some gravel zones to sands with varying proportions of silt and clay. The soils are generally hard or dense, but loose zones can exist. Indurated to partly indurated layers of clay and sand are common within the Middendorf Formation. A layer of hard clay approximately 15 to 30 feet thick is consistently present across the plant site.
>
> A thin zone of recent soils caps the Middendorf Formation. The recent soils are sands of either alluvial, fluvial or aeolian deposition and vary in density and silt content both laterally and vertically. The boundary between the recent soils and the Middendorf Formation is not clearly identifiable and the recent soils are combined with the upper part of the Middendorf Formation for analysis based on similar shear wave velocities.

The current ground surface in the main plant area is at approximately elevation 226 feet. Elevations in the seismic studies are referenced to the National Geodetic Vertical Datum of 1929 (NGVD29); the datum in effect when the plant was designed and constructed. Slight amounts of cut and fill were needed to reach the current grade. The primary Category 1 structures (Reactor Building, Auxiliary Building, Turbine Generator Class 1 Building, Fuel Handling Building and New Fuel Building) are supported on driven pile foundations embedded into a hard clay layer within the Middendorf Formation. Pile tips for the reactor building are generally within the range of elevations 155 feet to 163 feet. Other Category 1 structures are supported on shallow-depth foundations bearing in sands in the upper part of the Middendorf Formation. The intake structure is supported on a mat foundation bearing on the hard clay layer at approximately elevation 172 feet.

Detailed information regarding the HBRSEP site hazard was provided to NRC in the seismic information submitted to NRC in response to the NTTF 2.1 Seismic information request [4]. The response of [4] only considered the control point elevation in the main plant area (elevation 226 feet) for calculation of the ground motion response spectra

(GMRS). The probabilistic seismic hazard analysis (PSHA) described in this submittal expands on the response of [4] to consider four other control points for ground motion analyses:

- Elevation 216 feet for the base of the reactor containment building;
- Elevation 159.2 feet for the tip of the reactor building piles;
- Elevation 163 feet for the Lake Robinson Dam Spillway;
- Elevation 180 feet for the Lake Robinson dam original ground surface, and;
- Elevation 244 feet for the FLEX Building.

3.1 Seismic Hazard Analysis

This section discusses the seismic hazard methodology, presents the final seismic hazard results used in the SPRA, and discusses important assumptions and important sources of uncertainty. The work follows the general guidance provided in Regulatory Guide 1.208 [25] and the SPID [2] but differs in some of the implementation details. These differences are discussed in appropriate sections of this report, and detailed information is available in [4].

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 HBRSEP site hazard was provided to NRC in the seismic hazard information submitted to NRC in response to the NTTF 2.1 Seismic information request [4].

The analyses for the plant area used three alternative median shear wave velocity profiles (A, B and C) as described in [4] and listed in Table 3-1. The three profiles were weighted as described in [4] based on the relative amount of data available. A different set of alternative median shear wave velocity profiles was used for the Lake Robinson dam area, as shown in Table 3-1 and described in [16]; however, the methodology for using the profiles was the same as described for the plant area in [4].

As described in [4], the characterization of epistemic uncertainty in site median Vs used for HBRSEP differs somewhat from the approach described in Appendix B of [2] in which a best estimate median Vs profile is defined and epistemic uncertainty in median Vs is represented by upper and lower bound profiles with velocities assigned based on an epistemic sigma for ln(Vs).

However, the characterization using the three alternate profiles produces comparable epistemic uncertainty. As discussed in [4], Appendix B of [2] lists a recommended value for sigma ln(median Vs) of 0.35 for sites with limited Vs data and indicates that for sites with multiple detailed shear wave velocity profiles, the appropriate value may be significantly smaller. As there is a substantial amount of velocity data for the Robinson site, the modeled epistemic uncertainty in median Vs presented in [4] is consistent with the recommendations in [2].

Table 3-1. HBRSEP Alternative Median Shear Wave Velocity Profiles and Included SSCs	
Alternative Median Shear Wave Velocity Profile Label	**Represented SSCs**
A	All within the Plant Area
B	
C	
1-Dam	Lake Robinson Dam
2-Dam	
3-Dam	

<u>Control Point Elevations</u>

The site control point elevation is 226 feet (NGVD 29) as noted in [4]. This point is the ground surface elevation within the main plant area and defines the GMRS reference point.

The reactor building foundation is a mat supported by piles. The control point for the base of the mat is elevation 216 feet. A FIRS was developed for this elevation as a geological outcrop motion as described in [16].

The pile tip elevations for the reactor vary slightly; a control point to represent the pile tips was taken at elevation 159.2 feet using the approach described in [4]. A Soil Column Outcrop Response (SCOR) FIRS was developed for this elevation as described in [16].

The Auxiliary Building and New Fuel Building are pile-supported structures with the base of the pile cap at elevation 222 feet. The GMRS at the site control point elevation of 226 feet was used for these structures.

The Lake Robinson Dam spillway is a concrete structure founded at elevation 163 feet. That point was taken as the control point elevation for the spillway. A Truncated Soil Column Response (TSCR) FIRS was developed as described in [16].

The Lake Robinson Dam embankment is not a Category 1 structure, but consideration of potential liquefaction impacts required that a FIRS at the elevation of the original ground (elevation 180 feet) be developed. Details of that FIRS development are presented in [16].

<u>Ground Motion Parameters</u>

The GMRS and FIRS for the control point elevations noted above are computed using the weighted alternative profiles as described above for a range of spectral frequencies between 0.5 Hz and 100 Hz. Hazard curves for the SPRA (mean and fractiles of 0.05, 0.16, 0.5, 0.84 and 0.95) are provided for the various profiles and control points for peak ground acceleration (PGA, modeled as occurring at 100 Hz).

3.1.1 Seismic Hazard Analysis Methodology

For the HBRSEP, the following method is used:

- Conduct a hard-rock PSHA for the site and perform deaggregation;

- Develop alternative shear wave velocity profiles as discussed above to represent the plant site and the dam site;
- Develop site amplification functions for motions at the various control points using site response analyses;
- Combine the site amplification functions with the hard rock PSHA model to produce soil hazard curves for the various control points;
- Develop horizontal uniform hazard response spectra (UHRS) and GMRS or FIRS, and;
- Develop vertical GMRS or FIRS.

Each of the above steps is summarized below.

Hard Rock PSHA

A hard-rock PSHA was performed for the HBRSEP site following the guidance of NRC Regulatory Guide 1.208 [25] and the SPID [2]. The PSHA used the following input:

- Seismic Source Model

  The seismic source model used for the HBRSEP PSHA is based on the "Central and Eastern United States Seismic Source Characterization (CEUS-SSC) for Nuclear Facilities" project [18]. Distributed seismic source zones and Repeated Large Magnitude Earthquake (RLME) sources within 625 miles (1,000 km) and those RLME seismic sources at greater distances that contribute at least 1 percent to the hazard at the site are included. Two refinements to the CEUS-SSC model are incorporated as discussed in [4]. These are:

  - Mmax distributions for seismotectonic zones IBEB, MID-C (A through D), PEZ-N, PEZ-W and SLR are updated as included in [19] and accepted by NRC [20] for use at HBRSEP.
  - Revisions to the earthquake catalog in the south-eastern US including removal of reservoir-impounding causes and aftershocks of the Charleston earthquakes of 1886 [21]. Revised recurrence rates were subsequently generated for the south-eastern US and the overall result was included in [22].

- Ground Motion Characterization (GMC)

  The GMC used for the HBESEP is the GMC model of [23] which provides Ground Motion Prediction Equations (GMPEs) for seven reference spectral frequencies: PGA (100 Hz), 25, 10, 5, 2.5, 1 and 0.5 Hz. These GMPEs have been endorsed by the NRC for use in computing hazard at nuclear sites [24].

- Seismicity Catalog

  The seismicity catalog used for the HBRSEP is based on the CEUS-SSC catalog provided by [18] with revisions as noted above. This catalog was current through the end of 2008 and includes all known earthquakes of a magnitude relevant for assessing earthquake hazard. The CEUS-SSC catalog was updated to include earthquakes through the end of November 2014 as described in [16]. Testing of the updated catalog confirmed the adequacy of the earthquake recurrence rates provided in [18] and [22].

Hard rock hazard calculations, including deaggregation, are performed according to [25]. A minimum moment magnitude of 5.0 is used in the calculations in lieu of a cumulative absolute velocity filter. Results of the PSHA and deaggregation are used to perform the site response analysis.

<u>Site Response Analysis</u>

Site response analyses are performed using suites of input acceleration time histories representing the hard rock hazard at the site. The results of the site response analyses are used to develop soil amplification functions for the site control point (Elevation 226 ft.) The amplification functions are then used to develop soil hazard curves for the control point from which horizontal UHRS and GMRS are developed. The same process is applied for the Dam Original Ground control point (Elevation 180 ft.).

For control points at Elevations 216 ft., 159.2 ft. and 163 ft., the amplification functions are developed from the full column site response analyses following guidance given in [26] and [27]. Details are given in [16].

Inputs to the site response analysis include:

> *Vs profiles*. The analyses for the plant area used three alternative median shear wave velocity profiles (A, B and C) as described in [4] and shown in Figure 3-1. The three profiles were weighted as described in [4] based on the relative amount of data available.
>
> As described in [4], the characterization of epistemic uncertainty in site median Vs used for HBRSEP differs somewhat from the approach described in Appendix B of [2] in which a best estimate median Vs profile is defined and epistemic uncertainty in median Vs is represented by upper and lower bound profiles with velocities assigned based on an epistemic sigma for ln(Vs).
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> However, the characterization using the three alternate profiles produces comparable epistemic uncertainty. As discussed in [4], Appendix B of [2] lists a recommended value for sigma ln(median Vs) of 0.35 for sites with limited Vs data and indicates that for sites with multiple detailed shear wave velocity profiles, the appropriate value may be significantly smaller. As there is a substantial amount of velocity data for the Robinson site, the modeled epistemic uncertainty in median Vs presented in [4] is consistent with the recommendations in [2].
>
> The depth to bedrock (approximately 410 feet below site ground surface) is based on results of a deep boring performed for the PSHA study as well as historical site data and geologic publications [16].
>
> *Shear modulus and damping curves.* Alternative sets of non-linear material properties (G/Gmax and damping curves) were developed for the soil and weathered rock at HBRSEP using the sediment characteristics as described in [28]. The alternative G/Gmax and damping relationships represent epistemic uncertainty in the dynamic behavior of the subsurface materials. Following the approach described in [2] for treating epistemic uncertainty in G/Gmax and damping, the G/Gmax and damping relationships developed in [28] were grouped into two sets: a “Sand” set representing a greater degree of nonlinearity and a “Clay” set representing a lesser degree of nonlinearity. This is consistent with the use by [2] of the guidance in [29] and the more linear Peninsular Ranges subset of [29] described in [2].

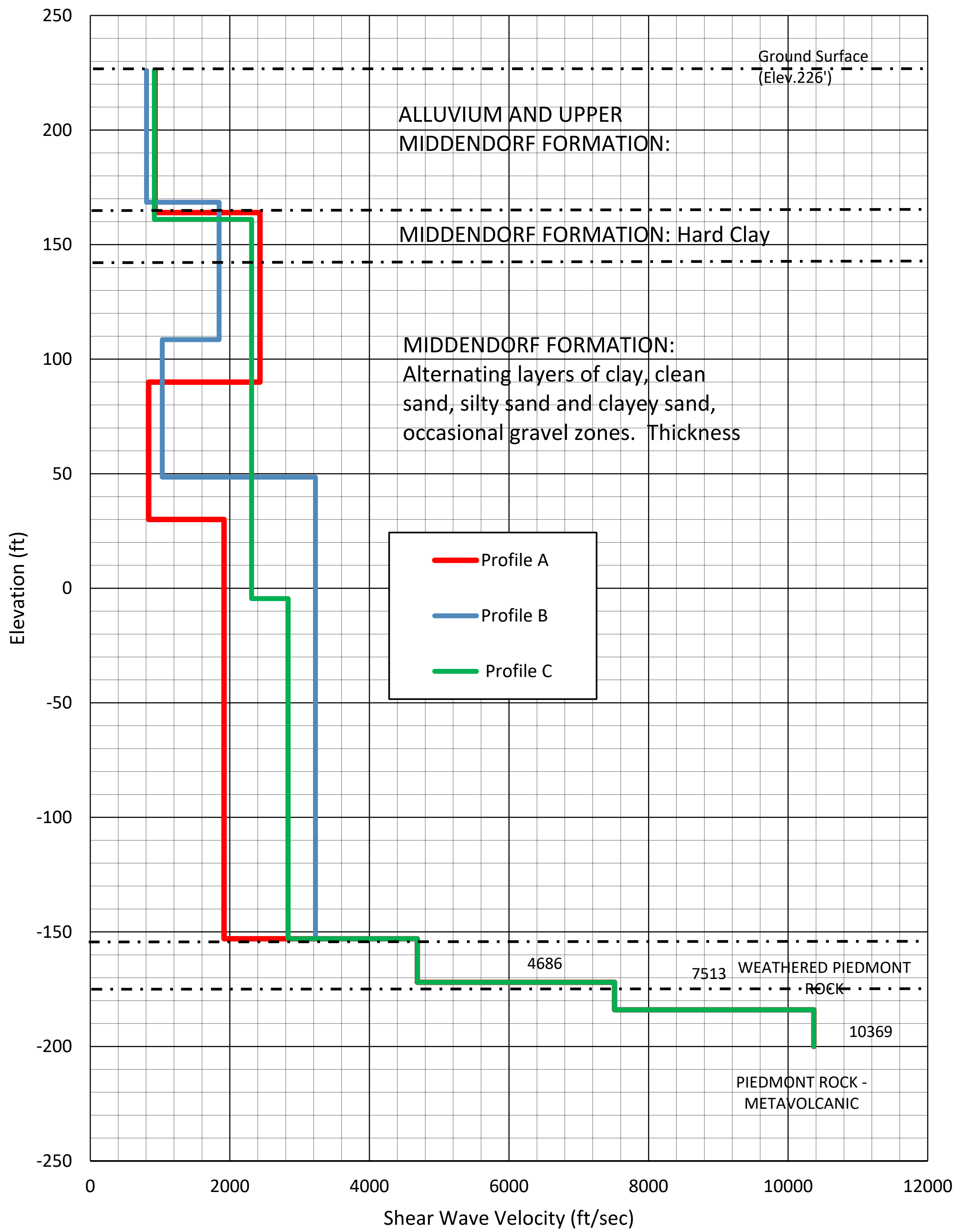

Figure 3-1. Plant Area Shear Wave Velocity Profiles A, B and C [4].

Input Rock Motions. The input rock motions were developed using site-specific information rather than the generic inputs described in Appendix B of [2]. Conditional Mean Spectra (CMS) were developed to represent the hard ground motions corresponding to AFE of 10^{-3}, 10^{-4}, 10^{-5} and 10^{-6}. Two CMS were developed for each AFE level, one to represent earthquake scenarios contributing to hazard for high frequency motions (≥5 Hz) and one for earthquake scenarios contributing to hazard for low frequency motions (≤2.5 Hz). For each CMS, 30 time histories were loosely matched to the CMS. More details are in [16].

Kappa. As discussed in [4], the characterization of damping for the site soils does not make use of the parameter kappa. A comparison between an equivalent value of kappa computed using [30] and the value from an empirical relationship in [31] showed a difference that is less than the standard error reported in [31], showing that the equivalent value of kappa derived from the damping relationships assigned to the HBRSEP soils is considered consistent with the recommendations in [2].

Site response analyses were performed for six analysis cases representing the epistemic uncertainty in site Vs profile (3 alternatives) and non-linear properties (2 alternatives). For each case, analyses were performed for 12 levels of input motion. The resulting amplification values were fit by a piece-wise continuous function defining the variation in ln(amplification) and its standard deviation as a function of the level of input rock motion. Figures in [4] show the results.

The analysis for other control point ground motions followed the approach described above; details and results are presented in [16].

<u>Soil PSHA</u>

Following the approach described in [2], the ground motion values at the control point elevation are developed in a hazard-consistent manner by applying Approach 3 in [32]. Approach 3 involves characterizing the amplification of the site soils in terms of the median (mean log) amplification functions and their associated standard deviations. Rather than applying the amplification functions at a post processor on the hard rock hazard as described in [2], the PSHA for the site was recalculated by convolving the soil amplification functions with the ground motion predictions from the rock ground motion models within the hazard integral to produce mean and fractile soil hazard curves at the control point. More details on the approach are given in [4]. Analyses for other control points followed similar methodology and details are in [16].

<u>Horizontal UHRS and GMRS</u>

Results of the PSHA and site response analysis are used to develop the UHRS and GMRS for the site control point at elevation 226 feet. The development of the smooth UHRS for AFE of 10^{-4}, 10^{-5}, and 10^{-6} is performed in two steps. The first step involves interpolation of the mean soil hazard curves to obtain the ground motion levels at the desired AFE levels for the seven ground motion frequencies recommended in [23]. The second step involves developing smooth interpolation/extrapolation functions using the response spectra computed in the site response analyses to provide smooth UHRS for the ground motion frequency range of 0.1 to 100 Hz (PGA). The performance-based GMRS is then computed from the 10^{-4} and 10^{-5} UHRS using the formulation in [25] based on the approach given in [33] for defining a risk-consistent Design Response Spectrum (DRS).

A similar approach is applied for other control points; details are in [16].

Vertical GMRS and FIRS

The vertical response spectra for the HBRSEP SPRA are needed for the GMRS/FIRS. Vertical response spectra consistent with the horizontal hazard are developed from the horizontal response spectra using vertical-to-horizontal (V/H) spectral ratios as suggested by [34]. The HBRSEP site control point at elevation 226 feet is underlain by approximately 410 feet of firm to stiff soil above hard rock. An envelope of the V/H ratios reported in [35] and [36] for similar site conditions was used to develop the vertical GMRS for the HBRSEP site. More details are in [16].

The above approach was applied for control points at elevation 216 feet and elevation 163 feet. For the control point at the base of the reactor piles, elevation 159.2 feet, a slightly different approach is used because the piles are founded in the hard clay layer which is stiffer than the average soil and the approach described above may produce V/H ratios that are too low at low frequency. Therefore, the V/H ratios for the base of the piles are calculated using both the envelope spectral ratios discussed above and the ratios developed in [32] for CEUS hard rock conditions of PGA in the 0.2 to 0.5 g range interpolated to the frequency values at which horizontal FIRS are computed. These values and the envelope discussed above are then enveloped and used to develop a vertical FIRS for the elevation 159.2 ft. control point [48]

3.1.2 Seismic Hazard Analysis Technical Adequacy

The HBRSEP SPRA hazard methodology and analysis associated with the horizontal GMRS were submitted to the NRC as part of the HBRSEP Seismic Hazard Submittal [4] and found to be technically acceptable by NRC for application to the HBRSEP SPRA.

The analyses performed for the HBRSEP SPRA described in this Section were subject to in-process peer review against the pertinent requirements in the SPID [2] and the PRA standard [5 and 37]. Comments from the third-party reviewers were addressed and incorporated into the vendor deliverables. HBRSEP ownership of the calculations was assumed, and the vendor deliverables were issued as site calculations. Once complete, the HBRSEP hazard analysis was also subjected to an independent peer review against the pertinent requirements in the PRA standard [5 and 37]. The peer review assessment, and subsequent disposition of peer review findings, is described in Appendix A.

3.1.3 Seismic Hazard Analysis Results and Insights

This section provides the final seismic hazard results used in the HBRSEP SPRA. Mean and fractile soil hazard curves for PGA at the 226 ft. elevation control point are used for quantification in the SPRA. These PGA hazard curves are provided in Figure 3-2 and Table 3-2.

Table 3-2 HBRSEP Mean and Fractile Exceedance Frequencies for PGA at Elevation 226 ft.						
Peak Ground Acceleration (g)	**Annual Frequency of Exceedance**					
	Mean	**5th%**	**16th%**	**50th%**	**84th%**	**95th%**
1.00E-03	5.70E-02	2.75E-02	3.80E-02	5.62E-02	7.59E-02	8.91E-02
2.00E-03	3.87E-02	1.95E-02	2.63E-02	4.07E-02	5.50E-02	6.46E-02
3.00E-03	3.03E-02	1.59E-02	2.14E-02	3.24E-02	4.37E-02	5.25E-02
5.00E-03	2.17E-02	1.18E-02	1.62E-02	2.34E-02	3.16E-02	3.98E-02
1.00E-02	1.29E-02	6.92E-03	9.77E-03	1.41E-02	1.91E-02	2.63E-02
2.00E-02	6.95E-03	3.31E-03	4.68E-03	7.24E-03	1.05E-02	1.59E-02
3.00E-02	4.70E-03	1.91E-03	2.82E-03	4.68E-03	7.41E-03	1.18E-02
5.00E-02	2.73E-03	7.59E-04	1.26E-03	2.46E-03	4.47E-03	7.24E-03
7.00E-02	1.81E-03	3.63E-04	6.46E-04	1.48E-03	3.09E-03	5.13E-03
1.00E-01	1.08E-03	1.45E-04	2.82E-04	7.59E-04	1.95E-03	3.39E-03
2.00E-01	2.64E-04	1.62E-05	3.63E-05	1.20E-04	4.57E-04	1.05E-03
3.00E-01	8.93E-05	3.89E-06	9.33E-06	3.31E-05	1.35E-04	3.63E-04
5.00E-01	1.75E-05	5.62E-07	1.48E-06	5.89E-06	2.40E-05	6.46E-05
7.00E-01	5.08E-06	1.38E-07	3.98E-07	1.62E-06	7.08E-06	1.86E-05
1.00E+00	1.24E-06	2.82E-08	9.12E-08	4.17E-07	1.86E-06	4.47E-06
2.00E+00	6.72E-08	8.91E-10	3.39E-09	2.24E-08	1.07E-07	2.82E-07
3.00E+00	1.43E-08	7.76E-11	3.55E-10	3.31E-09	2.14E-08	6.46E-08
5.00E+00	1.72E-09	2.51E-12	1.51E-11	2.14E-10	2.19E-09	8.13E-09
1.00E+01	6.03E-11	8.71E-15	8.13E-14	2.24E-12	5.13E-11	2.40E-10
2.00E+01	1.15E-12	9.33E-18	1.48E-16	8.71E-15	5.37E-13	3.09E-12

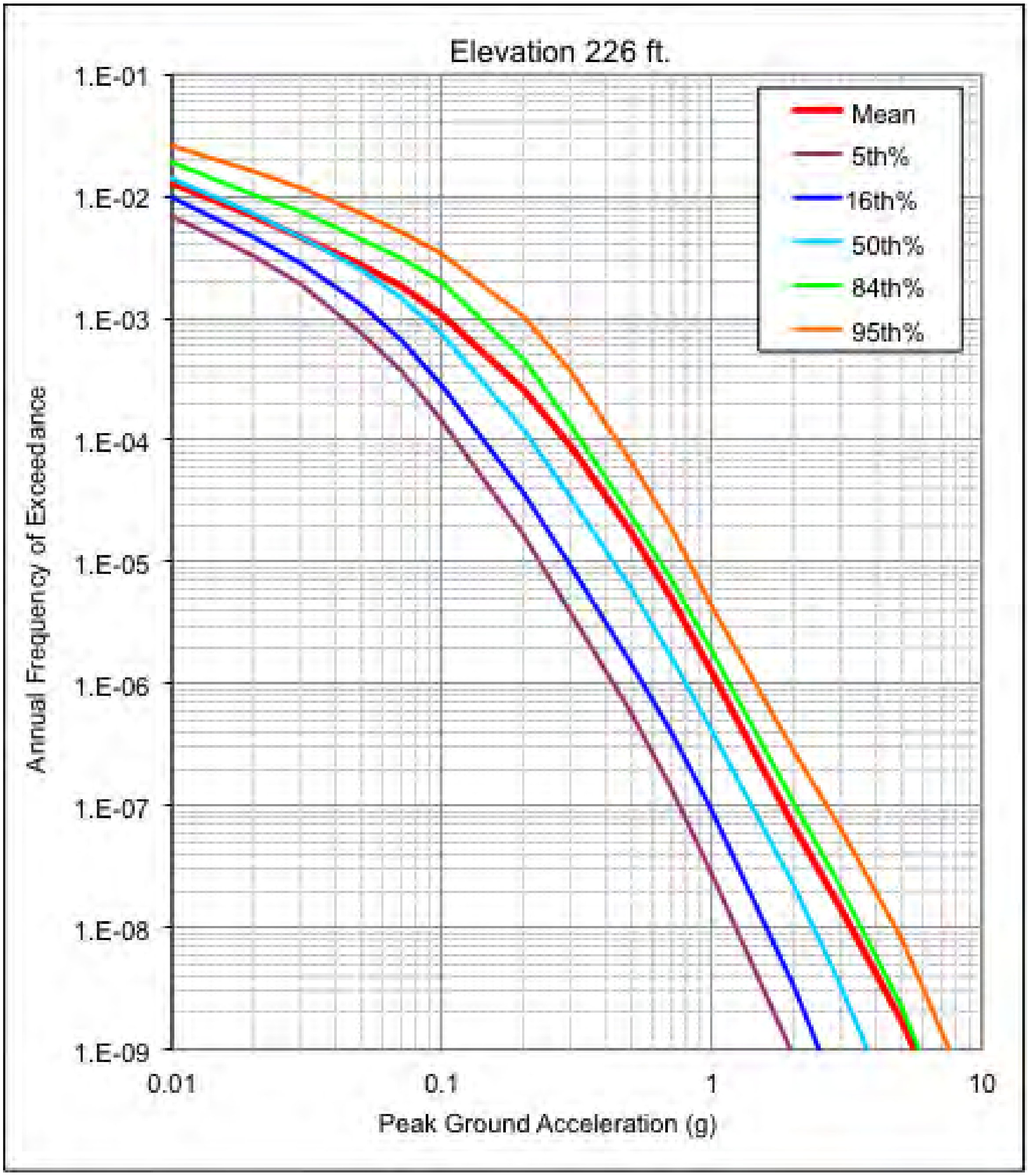

Figure 3-2. HBRSEP Mean and Fractile Soil Hazard Curves for PGA for Elevation 226 ft.

<u>Sources of Uncertainty</u>

The soil hazard fractiles are produced by the combination of the epistemic uncertainty in the CEUS rock hazard and the epistemic uncertainty in the site response model parameters characterized by the logic trees presented in [16]. The epistemic uncertainty is quantified by the variance in AFE computed for ground motion levels of AFE from 10^{-3} to 10^{-7} as shown on Figures 5-15 through 5-21 in [16].

For low frequency ground motions, the largest contributions to the uncertainty in the hazard are from the uncertainty in the RLME magnitudes and recurrence rates. For the HBRSEP site, this is primarily the uncertainty in the Charleston RLME characterization. The other major contributors are the uncertainty in the ground motion median models in [23]. For high frequency ground motions, the uncertainty in the ground motion median models in [23] becomes the largest contributor to the uncertainty in AFE. In addition, uncertainty in the distributed seismicity sources seismicity parameters has a contribution. The uncertainty in the characterization of the site dynamic properties has a small contribution to the total uncertainty in AFE.

3.1.4 Horizontal and Vertical GMRS

This section provides the control point horizontal and vertical GMRS.

The horizontal UHRS and GMRS and the vertical GMRS described in Section 3.1.1 above are provided in Table 3-3 and Figure 3-3. The V/H ratios used as described in Section 3.1.1 are shown in Figure 3-4 and tabulated in Table 3-4. Horizontal and vertical FIRS were also developed for the reactor foundation level at elevation 216 ft., the base of the Lake Robinson dam spillway at elevation 163 ft., and a vertical FIRS for the base of the reactor building piles at elevation 159.2 ft. These FIRS are provided in [16].

Table 3-3 Smoothed UHRS for Elevation 226 ft. and Horizontal and Vertical GMRS					
	Spectral Acceleration (g)				
Frequency (Hz)	Horizontal 10^{-4} UHRS	Horizontal 10^{-5} UHRS	Horizontal 10^{-6} UHRS	Horizontal GMRS	Vertical GMRS
100.000 (PGA)	2.88E-01	5.82E-01	1.05E+00	3.03E-01	3.03E-01
90.090	2.88E-01	5.91E-01	1.12E+00	3.07E-01	3.24E-01
83.333	2.88E-01	6.33E-01	1.20E+00	3.25E-01	3.57E-01
66.667	2.88E-01	7.43E-01	1.48E+00	3.69E-01	4.57E-01
60.241	2.88E-01	7.99E-01	1.64E+00	3.91E-01	4.81E-01
50.000	2.98E-01	9.35E-01	1.84E+00	4.46E-01	5.42E-01
40.000	3.34E-01	9.89E-01	1.95E+00	4.77E-01	5.57E-01
33.333	3.63E-01	9.87E-01	1.91E+00	4.85E-01	5.46E-01
25.000	4.45E-01	9.43E-01	1.78E+00	4.87E-01	4.98E-01
20.000	5.00E-01	1.01E+00	1.86E+00	5.24E-01	4.93E-01
16.667	5.42E-01	1.05E+00	1.92E+00	5.52E-01	4.83E-01
13.333	5.90E-01	1.12E+00	2.01E+00	5.92E-01	4.84E-01
11.111	6.15E-01	1.19E+00	2.10E+00	6.24E-01	4.89E-01
10.000	6.32E-01	1.23E+00	2.15E+00	6.44E-01	4.91E-01
8.333	6.00E-01	1.20E+00	2.11E+00	6.28E-01	4.62E-01
6.667	5.82E-01	1.17E+00	2.06E+00	6.11E-01	4.32E-01
5.882	5.95E-01	1.19E+00	2.11E+00	6.23E-01	4.31E-01
5.000	6.12E-01	1.22E+00	2.17E+00	6.36E-01	4.27E-01
4.000	6.39E-01	1.26E+00	2.27E+00	6.59E-01	4.28E-01
3.333	6.38E-01	1.22E+00	2.21E+00	6.45E-01	4.19E-01
3.000	5.96E-01	1.18E+00	2.13E+00	6.17E-01	4.01E-01
2.500	5.25E-01	1.07E+00	1.94E+00	5.59E-01	3.63E-01
2.000	4.70E-01	1.03E+00	1.85E+00	5.28E-01	3.43E-01
1.667	4.54E-01	9.71E-01	1.77E+00	5.01E-01	3.25E-01
1.333	4.05E-01	8.61E-01	1.59E+00	4.44E-01	2.89E-01
1.111	3.30E-01	7.29E-01	1.40E+00	3.73E-01	2.42E-01
1.000	2.84E-01	6.42E-01	1.30E+00	3.27E-01	2.13E-01
0.667	1.39E-01	3.94E-01	9.08E-01	1.92E-01	1.25E-01
0.500	8.38E-02	2.60E-01	6.78E-01	1.24E-01	8.09E-02
0.333	4.77E-02	1.47E-01	4.18E-01	7.03E-02	4.57E-02
0.250	3.66E-02	1.07E-01	2.90E-01	5.20E-02	3.38E-02
0.200	2.86E-02	8.53E-02	2.30E-01	4.11E-02	2.67E-02
0.167	2.27E-02	7.04E-02	1.86E-01	3.37E-02	2.19E-02
0.133	1.78E-02	5.39E-02	1.52E-01	2.59E-02	1.68E-02
0.111	1.35E-02	4.34E-02	1.24E-01	2.06E-02	1.34E-02
0.100	1.16E-02	3.67E-02	1.05E-01	1.75E-02	1.14E-02

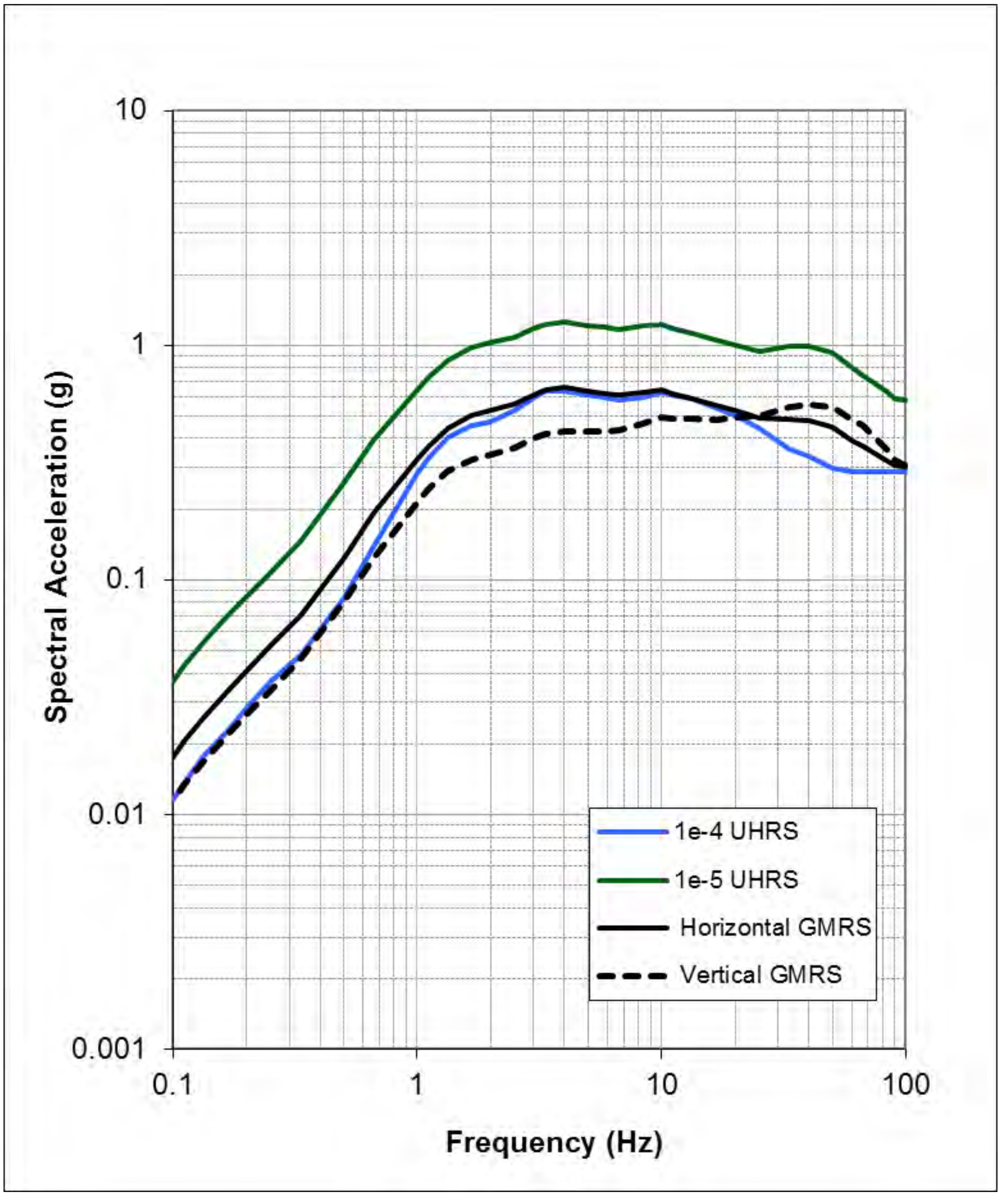

Figure 3-3. Horizontal Smoothed UHRS for MAFEs of 1e-4 and 1e-5, and the Horizontal and Vertical GMRS (5% damping) at Elevation 226 ft.

Table 3-4. V/H Ratios for GMRS at Elevation 226 ft.

Frequency (Hz)	V/H Ratio
100.000	1.000
90.090	1.056
83.333	1.101
66.667	1.238
60.241	1.230
50.000	1.216
40.000	1.166
33.333	1.128
25.000	1.023
20.000	0.939
16.667	0.875
13.333	0.817
11.111	0.783
10.000	0.763
8.333	0.737
6.667	0.708
5.882	0.692
5.000	0.672
4.000	0.650
3.333	0.650
3.000	0.650
2.500	0.650
2.000	0.650
1.667	0.650
1.333	0.650
1.111	0.650
1.000	0.650
0.667	0.650
0.500	0.650
0.333	0.650
0.250	0.650
0.200	0.650
0.167	0.650
0.133	0.650
0.111	0.650
0.100	0.650

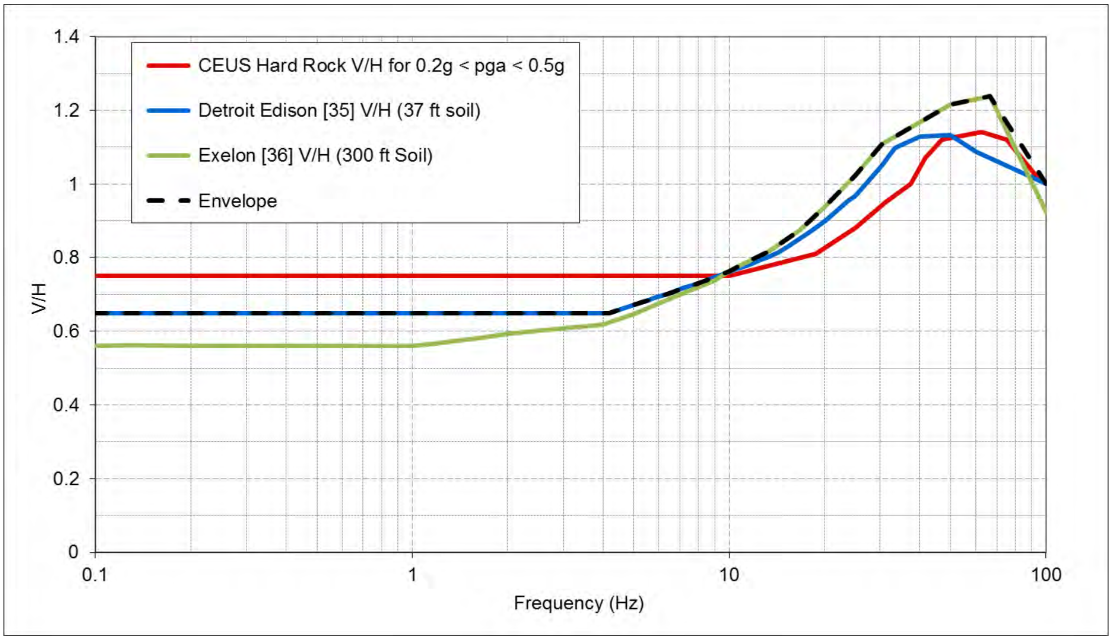

Figure 3-4. V/H Ratios for Two Soil Sites Compared to the Hard Rock V/H Ratios from [31] for PGA of 0.2 to 0.5 g.

3.1.5 Liquefaction Hazard Evaluation – Plant Area

In accordance with HLR-SHA-I [5], the HBRSEP is screened for potential secondary seismic hazards. Except for potential for soil liquefaction, no secondary seismic hazards are identified. This section briefly describes the liquefaction evaluation and results. More details are provided in [16].

<u>Liquefaction Screening</u>

The broad process described in [38] for screening and evaluation of liquefaction potential is used in the HBRSEP approach with some deviations. The liquefaction evaluation is part of the SPRA project which is not a design-basis assessment, but an assessment that includes behavior under extreme events. Thus, use of conservative parameter selection as would be required for the design basis approach described in [38] is not necessary for the SPRA. Instead, best estimates for the input parameters to calculate factor of safety of liquefaction are used based on the data, and the triggering factor of safety for initiation of liquefaction is taken as 1.00, instead of the value of 1.1 used in [38].

The upper part of the Middendorf Formation in the HBRSEP site profile described previously includes sands with variable densities below the water table. Such conditions cause the HBRSEP site to be screened in for potential for liquefaction; however, it is noted that available records do not show liquefaction as having occurred at the HBRSEP site or nearby. Details of the screening are in [16].

Liquefaction Evaluation Methodology

In summary, assessment of liquefaction triggering is done by developing a cyclic resistance ratio, CRR, and a cyclic stress ratio, CSR. The soil characteristics entering into both ratios are developed from site specific data using the best estimate parameters as listed in [16]. Adjustments to the CSR and CRR are made to account for differing earthquake magnitudes, overburden effects and geologic age as described in [16]. Earthquake ground motions representing annual frequencies of exceedance for the four target hazard levels (10^{-4}, 10^{-5}, 10^{-6} and GMRS) are used. Both high frequency motions and low frequency motions from the site response analysis discussed previously are used in the evaluation.

The CRR is computed using the methods of Boulanger and Idriss [39] with a probability of liquefaction of 50 percent. The calculation methodologies are summarized in [16] with details in [40] and [41].

Liquefaction Evaluation Results – Plant Area

Factors of safety were computed to evaluate liquefaction triggering for all exploration points in [42]. The explorations include 30 Standard Penetration Test borings (SPT), three Cone Penetration Test probes (CPT), and two Geophysics boreholes with shear wave velocity (V_s) measurements, for a total of 35 points. Samples above the water table, below the top of the hard clay stratum, or having factors of safety greater than 2.00 are represented with a factor of safety of 2.00.

Table 3-5 summarizes the results of the factor of safety computations. See [40] for detailed figures with results. Samples with factors of safety ≤ 1.00 occur with increasing frequency as the hazard level decreases. For example, there are only 12 exploration points at the GMRS hazard level (HF) that have points with factors of safety ≤ 1.00, while there are 32 exploration points at the 10^{-6} hazard level (LF). Instances of liquefaction triggering occur at variable vertical depths within the exploration points, separated by zones without liquefaction.

Table 3-5. Summary of Liquefaction Triggering Results in Plant Area					
Target Hazard Level	**Summary of Liquefaction Results (LF)**				
	Number of Liquefaction Zones				
	None	1	2	3	4 or more
	Number of Explorations (Max thickness, Min. thickness)				
10^{-4} AFE	30	5 (27, 2)	0	0	0
GMRS	25	7 (27, 2)	3 (6, 3)	0	0
10^{-5} AFE	3	14 (40, 3)	10 (19, 1)	7 (27, 1)	1 (9, 4)
10^{-6} AFE	3	4 (11, 5)	12 (45, 3)	6 (27, 1)	10 (17, 1)
Target Hazard Level	**Summary of Liquefaction Results (HF)**				
	Number of Liquefaction Zones				
	None	1	2	3	4 or more
	Number of Explorations (Max thickness, Min. thickness)				
10^{-4} AFE	28	6 (25, 2)	1 (6, 6)	0	0
GMRS	23	10 (25, 2)	2 (9, 3)	0	0
10^{-5} AFE	11	8 (27, 1)	11 (21, 1)	4 (9, 1)	1 (12, 2)
10^{-6} AFE	7	10 (40, 1)	12 (27, 1)	1 (8, 3)	5 (12, 1)

While liquefaction could potentially occur at a single exploration, not every sample in the exploration may indicate liquefaction triggering, and in adjacent explorations, liquefaction triggering may not be indicated at the same elevations. As shown in [42], there is not an indication at any of the four hazard levels of a liquefiable soil layer that is continuous across the protected area.

<u>Impacts of Liquefaction – Plant Area</u>

Potential impacts of liquefaction are ground settlement and lateral displacement of soils toward the lake.

Ground Settlement. Using the factors of safety for liquefaction determined as described above, probabilistic factors of safety and evaluation of seismic settlement are performed as described in [43]. The impacts to seismic fragilities are discussed in Section 4.

Lateral Displacement. Lateral spreading occurs when a soil mass slides laterally on a liquefied layer, and gravitational and inertial forces cause the layer, and the overlying non-liquefied material, to move in a downslope direction. In order for the lateral spreading to mobilize there must be a continuous layer of liquefying soils so that a failure surface can form and connect to an outlet, such as the free face of a slope. The locations of free faces are at the Lake Robinson shore and at the discharge canal. Cross sections between locations of the priority Structures, Systems and Components (SSC) and these two features were used to obtain lateral distances and vertical slope heights as discussed in [42].

Groupings of borings that could represent soil conditions along the cross sections are selected from the available site data and a criterion for determining if a cross section has a continuous layer of liquefiable soils is created. That criterion and the results of probabilistic factor of safety evaluations in [43] are used to compute Lateral Displacement Indices with the methodology described in [44].

Non-exceedance probabilities of lateral spreading displacements were evaluated in a step-wise manner as described in [46]. The probabilities of a continuous layer are shown in Table 3-6 and the numerical results are summarized in Tables 3-7, 3-8 and 3-9. The lateral displacement results are used to evaluate fragility estimates as described in Section 4.

Table 3-6. Probability of a Continuous Layer of Liquefaction Occurring, Plant Area			
	Prob. of a Continuous Layer		
Target Hazard Level	**3 to 4 borings (140-280 feet)**	**8 to 9 borings (490-630 feet)**	**13 to 14 borings (840-980 feet)**
10^{-4}	3.2%	0.0%	0.0%
GMRS	9.6%	0.4%	0.0%
10^{-5}	34.6%	5.7%	1.4%
10^{-6}	47.6%	9.7%	3.6%

Table 3-7. Lateral Displacement (LD) for 140 to 280-foot Cross Section (3-4 borings), Plant Area				
Statistical Value	**Target AFE**			
	10^{-4}	**GMRS**	**10^{-5}**	**10^{-6}**
	LD (in)	**LD (in)**	**LD (in)**	**LD (in)**
Triggering LD and Percentile	76.4 (96.9th percentile)	46.1 (90.4th percentile)	46.0 (65.4th percentile)	57.6 (52.4th percentile)
Minimum	0.0	0.0	0.0	0.0
2nd Percentile	0.0	0.0	0.0	0.0
16th Percentile	0.0	0.0	0.0	0.0
84th Percentile	0.0	0.0	309.1	370.2
98th Percentile	243.9	388.1	618.8	680.4
Maximum	592.4	739.1	903.3	949.1

Table 3-8. Lateral Displacement (LD) for 450 to 630-foot Cross Section (8-9 borings), Plant Area

Statistical Value	Target AFE			
	10^{-4}	GMRS	10^{-5}	10^{-6}
	LD (in)	LD (in)	LD (in)	LD (in)
Triggering LD and Percentile	None	50.3 (99.6th percentile)	30.7 (94.3th percentile)	20.0 (90.4th percentile)
Minimum	0.0	0.0	0.0	0.0
2nd Percentile	0.0	0.0	0.0	0.0
16th Percentile	0.0	0.0	0.0	0.0
84th Percentile	0.0	0.0	0.0	0.0
98th Percentile	0.0	0.0	78.3	86.0
Maximum	0.0	71.5	110.5	115.2

Table 3-9. Lateral Displacement (LD) for 840 to 980-foot Cross Section (13-14 borings), Plant Area

Statistical Value	Target AFE			
	10^{-4}	GMRS	10^{-5}	10^{-6}
	LD (in)	LD (in)	LD (in)	LD (in)
Triggering LD and Percentile	None	None	73.3 (98.6th percentile)	63.8 (96.4th percentile)
Minimum	0.0	0.0	0.0	0.0
2nd Percentile	0.0	0.0	0.0	0.0
16th Percentile	0.0	0.0	0.0	0.0
84th Percentile	0.0	0.0	0.0	0.0
98th Percentile	0.0	0.0	0.0	124.5
Maximum	0.0	0.0	160.6	169.0

3.1.6 [redacted]

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[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
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	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
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	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]

[redacted]

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 RNP SPRA. The subsections provide brief summaries of these elements.

4.1 Seismic Equipment List

For the RNP SPRA, a seismic equipment list (SEL) was developed that includes those SSCs that are important to achieving safe shutdown following a seismic event, and to mitigating radioactivity release if core damage occurs, and that are included in the SPRA model. The methodology used to develop the SEL is generally consistent with the guidance provided in Seismic Probabilistic Risk Assessment Implementation Guide, EPRI 3002000709, December 2013 [11] and Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic, EPRI Repor1025287 [2].

4.1.1 SEL Development

The RNP SPRA SEL is developed by using the RNP existing full-power PRA models as the starting point. Use of the PRA models as a starting point for SSCs to consider for fragility analysis is a rational starting point as the PRA models have already identified and modeled SSCs that cover all the critical safety functions and are appropriate for modeling in PRA core damage frequency (CDF) and release frequency models. The process begins by extracting the basic events from the respective models. A tabular list of all the RNP PRA basic events was used as input to this SEL development [64]. In addition, reviews of RNP drawings and RNP PRA System Notebook model boundary diagrams were performed as part of the initial SEL development to confirm that the PRA models are a sufficiently detailed input for the SEL development. These drawing reviews also assisted in locating equipment and identifying various passive failure items not contained in the PRA models.

Once the unique basic event list is generated, the basic events are then reviewed to disposition from further detailed consideration those basic events that need not be carried further in the SEL development process. Such events include:

- Non-applicable initiating events
- Type A and B HEP basic events
- Dependent HEP basic events
- Functional recovery and repair basic events
- Test and Maintenance basic events
- Common cause failure (CCF) basic events
- Flag basic events
- Other basic events

SSCs not to be credited in the SPRA include BOP equipment not powered by emergency AC (e.g., Main Feedwater, Condensate and Main Circulating Water). The SEL line items related to these systems are screened out from consideration of future seismic fragility analysis activities.

On-site structures and passive equipment was also reviewed for potential inclusion on the SEL. Structures that house or spatially interact with identified SSCs, as well as those that involve ex-Control Room actions credited in the SPRA, are included in the SEL for future consideration. The following buildings and structures are identified for inclusion on the SEL:

- Reactor Containment Building (RCB)
- RB Internal Structure
- Reactor Auxiliary Building (RAB)
- RHR Pump Pit
- Turbine Building Class 1 Bay
- Turbine Building Class 3 Portion
- Fuel Building (#215)
- Maintenance Fab Shop (#390)
- Switchyard
- Unit 2 Intake Structure
- DS Diesel Generator Bldg
- AFW "C" Diesel Generator Bldg
- SW Pipe Enclosure
- Radwaste Building (#210)
- RCB Sump

The following earthen structures are identified for inclusion on the SEL:

- Intake Pool Submerged Dike
- Discharge Canal
- Robinson Dam

The following large above ground storage tanks are identified for inclusion on the SEL:

- Condensate Storage Tank (CST)
- Refueling Water Storage Tank (RWST)
- Diesel Fuel Oil Storage Tank (DFOST)

The following buried items are identified for inclusion on the SEL:

- Deepwell Pumps and Buried Discharge Piping
- Deepwell Pumps A, B and C Buried Electrical
- Deepwell Pump D Buried Electrical
- AFW Pump C DG Buried Electrical
- DS DG Buried Electrical
- Service Water Buried Piping
- Service Water Buried Electrical
- Fire Water Buried Piping
- EDG Fuel Oil Transfer Buried Piping
- EDG Fuel Oil Transfer Buried Electrical

In addition, per the EPRI SPRA Implementation Guide, items associated with reactor scram function (reactor internals), offsite power and primary system LOCA were reviewed and added to the SEL.

Previous analyses have been completed in support of determining the seismic risk at all nuclear power plants. As a result, these analyses are used to supplement the basic event review performed in the previous steps to determine if the additional SEL line items are warranted. The following seismic evaluations were reviewed to identify any potential additional SSCs not yet included in the previous steps:

- RNP IPEEE Seismic Equipment List [51]
- RNP NTTF 2.3 Seismic Walkdown Equipment List [3]
- RNP ESEP List [10]
- RNP Final Implementation Plan (FLEX) [9]

Seismic-induced fragility of the main control room ceiling has been considered in past SPRAs as a potential impact on the human error probabilities used in the PRA. As such, the main control room ceiling is added to the SEL for future consideration. The identification of the Ex-Control Room actions support a review of the pathway availability to the action location and the location itself was performed. This review is documented in the SEL notebook. The following items were added to the SEL for consideration for operator ex-MCR post-initiator action access issues:

- Sliding door to DG Room A
- Sliding door to DG Room B
- CO2 tanks next to MCC-1

Certain types of equipment are inherently rugged such that it need not be considered for fragility modeling in an SPRA. This includes check valves, manually operated valves, disconnect switches and inline piping items (e.g., nozzles, orifices, filters). These items are not included in the SEL based on their very high seismic capacity and their passive nature.

Equipment that is captured through “rule-of-the-box” considerations, e.g., equipment contained on a skid or in a cabinet that can be subsumed into the major skid equipment or into the cabinet, was also not explicitly included on the SEL. For such equipment, the seismic fragilities for the containing equipment consider all of the equipment in the “box.”

A walkdown for the RNP SPRA was performed in support of the initial SEL development. This walkdown covered confirmation of SSC locations, identification of missing items, “Rule of the Box” grouping, Ex-Control Room Operator Action Access Blockage items, identification of inherently rugged equipment, Seismic, fire, and flood sources identification. In addition, separate seismic capacity walkdowns were performed and documented in the station calculation for the seismic capacity walkdown [53]. These walkdowns identified additional SEL items that were fed back into the SEL.

As a final check, the RNP SEL was compared for reasonableness with the SEL developed for the Surry Power Station during the EPRI Surry SPRA Pilot study (Reference [89], EPRI 1020756, Surry Seismic Probabilistic Risk Assessment Pilot Plant Review, Electric Power Research Institute, July 2010.). This comparison did not identify SSCs inappropriately omitted from the RNP SEL development.

The resulting SEL includes about 339 component entries (Table 4 of the station calculation for the SEL development) [64]. The final SEL was documented for the SPRA.

4.1.2 Relay Evaluation

During a seismic event, vibratory ground motion can cause relays to chatter. The chattering of relays potentially can result in spurious signals to equipment. Most relay chatter is either acceptable (does not impact the associated equipment), is self-correcting, or can be recovered by operator action. An extensive relay chatter evaluation was performed for the Robinson SPRA, in accordance with SPID [2], Section 6.4.2 and ASME/ANS PRA Standard [5], Section 5-2.2. The results of this assessment show that the chatter of the vast majority of devices is of no consequence to the accident mitigation function of the components supported by the devices, or may require reset of a component from the Main Control Room or locally [72]. Chatter of some devices, typically in a lockout or seal-in type of application, can result in the need to reset a component outside the Main Control Room. The operator action to reset a component outside the Main Control Room may have an impact on the seismic risk and may need to be modeled in the seismic PRA. The results of the assessment found a relatively small number of unique combinations of relays requiring further assessment in the RNP SPRA. This unique set is summarized in the station calculation for the essential relay development [72]. This listing reflects a total of 90 individual devices/relays involving a total of 114 contact pairs.

4.2 Walkdown Approach

This section provides a summary of the methodology and scope of the seismic walkdowns performed for the RNP SPRA. Walkdowns were performed by personnel with appropriate qualifications as defined in the SPID [2]. Walkdowns of those SSCs included on the seismic equipment list were performed to assess the as-installed condition of these SSCs for determining their seismic capacity and performing initial screening.

Previous walkdowns from USI A-46 [48] and IPEEE [51] were used to inform the walkdowns for the RNP SPRA. The components of the RNP A-46 Safe Shutdown Equipment List (SSEL) required a thorough walkdown and review. The essential information for the equipment included in the USI A-46 SSEL had been assembled into data files using the Seismic Qualification Utility Group, *Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment*, Revision 2 [50], screening evaluation worksheets (SEWS), anchorage calculations, outlier sheets, photographs, drawings, test reports, and other background documentation. In addition, RNP completed a Seismic Margin Assessment (SMA) for the IPEEE [51]. As part of the IPEEE effort, extensive walkdowns were conducted in 1993 and 1994 for RNP Unit 2 consistent with the intent of the guidelines described in EPRI NP-6041-SL [8]. Because the screening rules for the USI A46 walkdown per the SQUG GIP [50] are similar to the rules for seismic margin walkdown per EPRI NP-6041-SL [8], the components common to both USI A-46 and IPEEE did not need an additional detailed walkdown for IPEEE. The IPEEE seismic capability team reviewed the USI A-46 equipment data files and then performed a walk-by of the equipment for seismic/fire, seismic/flood, and spatial interactions applicable to beyond design basis seismic events, including block walls upgraded under I&E Bulletin 80-11 [52]. The previous walkdowns were particularly useful to obtain information for components with restricted access. Information from those prior walkdowns was used where the appropriate level of detail needed for the SPRA was available.

The walkdowns were performed in accordance with the EPRI SMA methodology report (EPRI NP-6041-SL [8]. Based on EPRI NP-6041-SL guidelines, the Seismic Review Teams (SRTs) reviewed a reasonable sample from each group of similar SEL items in full detail (“full scope walkdown”) and reviewed the remaining reasonably accessible SEL items via “walk-by.”

In full scope walkdowns, the SRT collected detailed notes on the equipment configuration and tag information (e.g., weight, model number), and performed a detailed evaluation against the walkdown caveats and criteria from Appendix F to EPRI NP-6041-SL [8]. The team also collected photographs, measurements, sketches, and any other data that could be used to inform fragility calculations. The walkdowns focused on the seismic ruggedness of the equipment, anchorage capacity, mounting of internal devices, and potential spatial systems interaction concerns. The walkdowns were documented in the Robinson Unit 2 Walkdown Database containing the detailed walkdown findings and observations and SEWS forms.

After a lead item or a sample of similar equipment had been thoroughly reviewed via full scope walkdown, other similar components were reviewed to confirm similarity with the lead items and to verify that there are no anomalies in installation or interaction concerns. The SRT members judged items to be similar based on equipment construction, dimensions, location, seismic qualification requirements, anchorage type, and configuration. The abbreviated, confirmatory walkdowns are termed walk-bys. The level of detail of the review depended on what was observed during the walk-by. For example, if the anchorage was found to differ substantially from the lead item, the team may have taken new measurements or made new sketches as necessary to adequately describe the differences. In accordance with EPRI NP-6041-SL [8] the walk-by components were documented on the SEWS forms by identifying the equipment number of the component (lead item) to which it is similar and briefly describing any unique seismic interaction issues and/or differences from respective lead items.

EPRI NP-6041-SL [8] guidance states that 100% walk-by of all SEL items is not necessary for equipment classes that have “excessively large numbers of like elements,” which would include classes such as local instruments and distribution systems such as piping, cable trays, and HVAC ducting. The SRTs performed the walk-bys for these items on an area or sampling-basis as described in EPRI NP-6041-SL [8] Section 2 and Appendix D. The SRTs reviewed selected samples of such items in the vicinity of SEL components to establish the consistency of construction and general robustness of their supports.

4.2.1 Significant Walkdown Results and Insights

Consistent with the guidance from EPRI NP-6041-SL [8], no significant findings were noted during the RNP seismic walkdowns. The walkdowns are documented in the station calculation for the seismic capacity walkdowns [53].

Components on the SEL were evaluated for seismic anchorage and interaction effects, effects of component degradation, such as corrosion and concrete cracking, for consideration in the development of SEL fragilities. The potential for seismic-induced fire and flooding scenarios was assessed. Potential internal flood scenarios were incorporated into the RNP SPRA model. The walkdown observations were appropriate for use in developing the SSC fragilities for the SPRA and any seismic interactions which were determined to be credible failure modes were documented and included as failure modes for the associated SEL components.

To support the iterative process for the risk assessment, the seismic capacity of each component was ranked by expert judgment as to whether the component is Rugged or exhibits a High, Medium, or Low seismic capacity. The capacity ranking of the SSCs is based on whether the SSC itself meets the guidelines in EPRI NP-6041-SL [8], engineering judgment regarding the capacity of the anchorage, and engineering judgment regarding any potential spatial systems interactions. SSCs ranked Rugged are judged by the SRT to have very high seismic capacity, such that they are not expected to significantly contribute to seismic risk. SSCs ranked High satisfy the applicable EPRI NP-6041-SL [8] walkdown criteria for 1.2g Sa ground motion and have robust anchorage. SSCs ranked Medium satisfy the applicable EPRI NP-6041-SL [8] walkdown criteria for 1.2g Sa ground motion, but their failure is likely to be governed by anchorage failure modes. SSCs ranked Low have obvious seismic deficiencies as judged by the SRT. The capacity estimates were used for preliminary screening and prioritization.

Electrical cabinets typically met EPRI NP-6041-SL [8] criterion, but anchorage frequently controlled. Heating, Ventilating & Air Conditioning (HVAC) and cable raceways generally have robust support systems and were found to have relatively high seismic capacities. General observations on potential interaction sources include that lighting fixtures in safety-related areas are generally well supported, HVAC ducting and non-safety piping is generally well supported, fire protection piping is of good construction and generally well supported.

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

4.2.2 Seismic Equipment List and Seismic Walkdowns Technical Adequacy

The RNP SPRA SEL development and walkdowns were subjected to an independent peer review against the pertinent requirements (i.e., the relevant SFR and SPR requirements) Capability Category II of Part 5 of the ASME/ANS PRA Standard [37].

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

4.3 Dynamic Analysis of Structures

This section summarizes the dynamic analyses of structures that contain systems and components important to achieving a safe shutdown. Modeling of structures is documented in the station calculation for structural modeling [54] and response analyses are documented in the station calculation for seismic response analysis [55]. Table 4-2 lists the structures that support systems and components on the SEL and the type of model used to perform the dynamic analysis, whether finite element model (FEM) or lumped mass stick model (LMSM). The table also lists whether effects of soil-structure interaction (SSI) were included in the dynamic response analysis, the structure damping used, and the parameters varied in the analysis. Effects of ground motion incoherence (GMI) on structure responses were evaluated in the station calculation for the RAB incoherence sensitivity study [56] and found to be insignificant.

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

Building	Model Type	SSI	Best-Estimate Structure Damping	Varied Parameters
Reactor Containment Building (RCB)	New FEM*	Yes	4%	Soil properties, structure stiffness, pile-to-soil interface stiffness, damping
Reactor Auxiliary Building (RAB)	New FEM	Yes	4%	Soil properties, structure stiffness, pile-to-soil interface stiffness, damping
Class I Turbine Building	Revised FEM	No	2%	Structure stiffness
Class III Turbine Building	New FEM**	No	7%**	Structure stiffness
Unit 2 Intake Structure	Partial Building FEM Model***	No	N/A	None

* The new FEM incorporates an existing LMSM of the containment shell that was revised to meet SPID modeling criteria

** A separate model and damping was used to compute the building fragility

*** A modal analysis was performed on a model of the Unit 2 Intake Structure roof slab to demonstrate that the load path to the SSCs mounted in the structure is not flexible enough to significantly amplify the input motion

In addition to the buildings listed above, analysis was performed on the RNP site to determine probabilistic distributions of liquefaction-induced settlement to support fragility analyses of SSCs sensitive to ground settlement. This analysis is documented in the station calculation for liquefaction settlement [43].

4.3.1 Cracking Analysis

Before performing response analyses to compute in-structure response spectra (ISRS), each uncracked building model is evaluated for effects of concrete cracking.

In a preliminary study on pile modeling for the RAB and RCB, the fragility vendor noted that the lateral flexibility in the pile foundations causes these buildings to behave as rigid structures on a flexible pile foundation in an earthquake. Due to this behavior, stresses in the RAB and RCB superstructures are low, and cracking of concrete structural elements does not occur at the site reference earthquake.

Cracking in the Turbine Building structures is evaluated using a response spectrum or time history analysis with the applicable input motion and the appropriate damping level based on Table 3-2 of ASCE/SEI 43-05 [58]. Stresses from the cracking analysis are reviewed to identify areas in the building that are expected to crack due to the reference earthquake and result in softening that could affect the input motion seen by SEL equipment. Structure stiffnesses are determined following Section 3.3 of ASCE/SEI 4-16 [59] with stiffness reductions in elements where cracking is expected.

Judgment is exercised when assessing whether cracking is sufficient to warrant stiffness reduction. Very localized cracking that does not significantly affect overall structure seismic response and force distribution may not require stiffness reduction. Conversely, stiffness reduction may be appropriate for localized cracking that affects local structure response of significance to SSC fragilities.

4.3.2 Fixed-base Analyses

Fixed-base dynamic response analyses are performed for the Class I and III Turbine Building structures. Response uncertainties are considered by analyzing models with best estimate (BE), lower bound (LB), and upper bound (UB) structure stiffness.

To obtain ISRS, each of the three structure stiffness models are analyzed for each of the applicable five earthquake input acceleration time history sets.

For each of the three structure stiffness models, the average of ISRS for the five sets of earthquake acceleration time histories is determined. The median ISRS is taken as the average of values for the three structure stiffness cases and the 84% non-exceedance probability (NEP) ISRS as the envelope of values for the three structure stiffness cases with valleys between the peaks of the median and 84% NEP ISRS filled in.

The ISRS are generated at an array of damping values ranging from 0.5% to 15% to cover damping values that might be needed for system and component fragility evaluation.

4.3.3 Soil Structure Interaction Analyses

SSI analyses are performed for the RCB and RAB using computer program SASSI. These structures are supported on piles and the SSI effects on their seismic responses are non-negligible. Pile elements are modeled with horizontal pile-to-soil springs connect the pile beam elements to the soil to represent the local flexibility and horizontal interaction between the piles and soil. The stiffness of these springs is based on pile lateral load test data.

The pile test data is judged to be stiffer than the expected response during seismic loading. To account for the local soil degradation due to cyclic loading effect, the computed scale factors are reduced to obtain the stiffness for the pile-to-soil springs.

Separate sets of SSI analyses are performed to permit determination of structure response variability due to uncertainty in soil properties, structure stiffness, pile-to-soil spring stiffness, and damping. For each varied parameter, median and 84% NEP ISRS are obtained similar to the process described for varying structural stiffness in fixed-base analyses using BE properties for all other modeling parameters. Only BE and LB damping values are considered for structural damping. For both buildings, effects of structural damping are found to be insignificant.

The median ISRS for seismic fragility evaluation is the average of (1) the median ISRS considering only variability in soil properties, (2) the median ISRS considering only variability in structure stiffness, and (3) the median ISRS considering only variability in pile-to-soil interface stiffness.

Spectra are computed for several damping levels, as required for the subsequent fragility evaluations. In addition to the ISRS calculated for seismic fragility evaluation, spectra are computed in the free field for evaluation of SSSI effects on the buildings adjacent to the RCB and RAB. It is found that the only building that is affected by the SSSI effects is the Class 1 Turbine Building.

4.3.4 Structure Response Models

The type of models used for dynamic response analyses are listed in Table 4-2. To meet SPID [2] modeling criteria for these structures, we developed new state of the art FEMs with the following exceptions. The model of the RCB included a new FEM representation of the internal structure as well as a recreated FEM of the Nuclear Steam Supply System (NSSS) and a revised version of an existing LMSM representation of the containment shell. The recreated NSSS model meets SPID [2] criteria and the original LMSM of the containment shell was modified to bring it into compliance with SPID [2] requirements by revising the modulus of elasticity, refining model discretization, and including mass moments of inertia. We used an existing FEM of the Class I Turbine Building with certain modifications to make it a median-centered model satisfying SPID [2] modeling criteria.

4.3.5 Seismic Structure Response Analysis Technical Adequacy

The RNP SPRA Seismic Structure Response and Soil Structure Interaction Analysis were subjected to an independent peer review against the pertinent requirements of Capability Category II of Part 5 of the ASME/ANS PRA Standard [37].

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

4.4 SSC Fragility Analysis

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

This section summarizes the fragility analysis methodology, and 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). Important assumptions and important sources of uncertainty, and any particular fragility-related insights identified, are also discussed.

4.4.1 SSC Screening Approach

A screening level that would result in a seismic core damage frequency (SCDF) of 5E-7/reactor-year was developed in accordance with SPID [2] guidance. This screening level for rugged SSCs is a high-confidence-of-low-probability-of-failure (HCLPF) capacity of 0.75g. Rugged SSCs are not removed from the SPRA model but are modeled with their respective fragilities, at least the screening level fragility. The screening-level fragility is assigned to several seismically rugged SSCs whose seismic capacity is determined to be equal to or greater than the screening level based on analysis or engineering judgement coupled with walkdown observations. Items commonly ranked rugged include local instruments, sensors, transmitters, and large-diameter valves. The items must be obviously well anchored and free of interaction concerns. The screening level is validated in Section 4.2 of the station calculation for the logic model quantification notebook [13].

For the RNP SPRA, the initial quantification was performed with representative fragilities that were conservatively biased, resulting in conservative SCDF and SLERF estimates. Then, for the remainder of the RNP SPRA project, significant efforts were expended to eliminate or minimize conservatisms in top risk contributor fragilities to avoid a situation where overall risk results and insights were unnecessarily masked by conservatively biased fragilities. This necessitated numerous refinement iterations in risk quantification.

For each iteration, risk insights were gained, and top contributors were identified. For the newly identified top contributors that were conservatively biased, more detailed fragilities were developed as SOV fragilities. The revised fragilities enabled a new iteration of risk quantification. The refinement process continued until reasonably converged quantification results were obtained with respect to risk results and insights, conditional upon a holistic review of fragility quality that top contributor fragilities were either realistic or, if they were not, the degree of conservatism was justified for its relative impact on the overall risk results (i.e., quantification results are not sensitive to potential fragility improvements).

4.4.2 SSC Fragility Analysis Methodology

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

The fragility evaluation effort began with the development of representative fragilities for all SSCs on the SEL as documented in the station calculation for representative fragility development [61]. These representative fragilities were used in an initial risk quantification to identify the most important SSCs and thus focus subsequent fragility evaluation efforts.

In many cases, the representative fragilities are conservatively biased and/or based on generic data (e.g., earthquake experience). Following each risk quantification conducted for the RNP SPRA, meetings were scheduled with the SPRA team to jointly determine the optimal path forward in terms of fragility refinements. Fragilities were refined for the most dominant risk contributors using increasingly sophisticated techniques to make them more realistic and plant-specific. Using the refined fragilities, the systems analyst then re-quantified the seismic risk and reprioritized the SSCs. This iterative process was repeated until the top risk contributors were ultimately characterized with a realistic and plant-specific fragility.

The final fragilities used to quantify the seismic risk of the plant are a combination of representative fragilities and more detailed fragilities (SOV fragilities).

4.4.2.1 Representative Fragilities

Representative fragilities are based on a combination of the following:

- Design information and calculations
- Seismic evaluations from the Unresolved Safety Issue (USI) A-46 [48] and Individual Plant Examination for External Events (IPEEE) [51] programs
- Judgements and rankings made during the seismic walkdowns
- Past fragility estimates and experience

- Recent assessments made at RNP as part of the Near-Term Task Force (NTTF) requirements of all U.S. nuclear plants (NTTF 2.1 and 2.3 requirements)
- Response analyses described in Section 4.3

In developing the initial representative fragilities, a slight conservative bias was incorporated to reduce the potential for changes to the risk ranking during subsequent risk quantifications.

4.4.2.2 Detailed fragilities

In most cases, detailed fragilities were performed following the SOV methodology documented in EPRI TR-103959 [62] along with more recent guidance from EPRI 1019200 [63] and EPRI 1025287 [2]. This involves identifying critical failure modes, computing median capacities (A_m) including effects of inelastic energy absorption, and computing lognormal standard deviations for randomness and uncertainty (β_r and β_u).

Liquefaction-induced settlement or deformation related fragilities do not fit a double lognormal distribution and required to be expressed as mean conditional probabilities of failures at the multiple hazard levels.

4.4.3 SSC Fragility Analysis Results and Insights

Refer to Section 5, Tables 5.4-2 and 5.5-2, for a tabulation of the fragilities for those SSCs (or correlated SSC groups) determined to be risk important, based on the final SPRA quantification. Tables 5.4-2 and 5.5-2 provide the risk important fragilities for SCDF and SLERF, respectively. The tables provide for each listed fragility, the median capacity and uncertainties (e.g., A_m, β_r, β_u), the calculation method, and the failure mode(s) addressed in the model.

4.4.4 SSC Fragility Analysis Technical Adequacy

The RNP SPRA SSC Fragility Analysis was subjected to an independent peer review against the pertinent requirements Capability Category II of Part 5 of the ASME/ANS PRA Standard [37].

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

5.0 Plant Seismic Logic Model

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

5.1 Development of the SPRA Plant Seismic Logic Model

The Robinson Nuclear Power Plant seismic response model was developed by starting with the Robinson Nuclear Power Plant internal events at power PRA model of record as of June 2015, and adapting the model in accordance with guidance in the SPID [2] and PRA Standard [5 and 37], including adding seismic fragility-related basic events to the appropriate portions of the internal events PRA, eliminating some parts of the internal events model that do not apply or that were screened-out, and adjusting the internal events PRA model human reliability analysis to account for response during and following a seismic event.

The general PRA modeling elements addressed in the development of the SPRA model are listed below, followed by a short description summarizing the treatment of each element.

- Initiating Event Analysis (IE)
- Accident Sequence Analysis (AS)/Success Criteria (SC)
- Systems Analysis (SY)
- Human Reliability Analysis (HR)
- Data Analysis (DA)
- Quantification (QU)

5.1.1 Initiating Event Analysis (IE)

Initiating events and consequential events that can be caused by a seismic event were considered by examining the SEL. Each SSC was examined to determine the plant impact from its failure. Since the plant impacts are already addressed by existing event trees, no new initiators needed to be considered. Passive failures, which may not have been represented in the internal events PRA, were given special attention, especially building failures.

Additionally, plant-specific seismic risk evaluations for the Surry [89] and Oconee [90] nuclear plants were reviewed to ensure all applicable initiating events have been accounted for in the RNP SPRA. These evaluations considered events, such as, seismic-induced flooding, seismic-induced fires, and seismic-induced failures of structures and dams, which are similarly evaluated in the RNP SPRA. No new initiating events or accident sequences were included in the RNP SPRA based on the review of the Surry and Oconee SPRAs.

Plant-specific seismic events are required to be considered in the RNP SPRA, if applicable. The intent of the requirement is to ensure the SPRA includes plant-specific seismic events and appropriately models the plant response to the event. No plant-specific events have occurred.

5.1.2 Accident Sequence Analysis (AS)/Success Criteria (SC)

This element consists of modeling the plant response using event trees, and identifying those functions and systems used to mitigate the modeled initiator. This PRA element starts with the review of the initiators identified in element (IE). No new accident sequences were developed, and so it was not necessary to modify the success criteria of the base model for seismic events. However, there is one variance in the success criteria for the SPRA due to crediting the Diverse and Flexible Coping Strategies (FLEX) systems and strategies. As discussed in the FLEX PRA system notebook, for events mitigated by FLEX strategies, the success criteria are limited to Phase 1 and Phase 2 FLEX strategies. Per MAAP analyses, core damage is prevented, and the plant is stable for at least 36 hours following the implementation of the Phase 2 FLEX strategies [73 and 66].

The PRA Standard requires examining the effect of including a small-small (also known as a very small) LOCA as an additional fault within each sequence in the Seismic PRA model. Very small LOCA (VSLOCA) break sizes of less than 0.35" were screened out of the internal events PRA because they could be mitigated by the highly reliable charging system. In the SPRA, these very small breaks are considered because the charging pumps are highly correlated and can fail concurrently during a seismic event. Thermal-hydraulic analysis [73] demonstrated a VSLOCA can be conservatively modeled with the same timing and success criteria as a Small LOCA, and no new event tree development was necessary. FLEX strategies are credited with mitigating a VSLOCA in the SPRA. In addition, FLEX is also credited for CST makeup and alternate suction to the SDAFW pump from the Condenser Inlet Waterbox.

Some SSCs included in the internal events PRA are not credited for the SPRA model; for example, Circulating Water pumps, primary water equipment, Fire Water equipment, and primary instrument air are not credited for the SPRA model due to their power source not being backed by a diesel generator and no credit is given for LOOP recovery for the first 24 hours.

5.1.3 Systems Analysis (SY)

The Internal Events PRA [88] system fault trees developed for the event tree functions were used as a starting point for the SPRA. Seismic-related events were inserted into the fault tree, and they were linked to 10 seismic acceleration levels. In general, at least 6 intervals are used in SPRAs and using 10 intervals for RNP should accurately represent the hazard curve. For SSCs which have no corresponding Internal Events PRA Basic Event (BE), the SSC was tied to events or gates which represented the same plant impact as the seismically failed SSC. Many SSCs are seismically rugged, and therefore are unchanged from the internal events PRA: e.g., check valves, manual valves, filters, dampers, and sensors; no seismic failure mode is added, only the nominal failure probability is included.

Relay chatter during a seismic event may cause SSCs to inadvertently actuate or lockout from actuating during or following the seismic-induced shaking. The plant specific assessment of relay chatter is provided in the station calculation for relay chatter systems evaluation [72].

FLEX equipment has been credited and modeled in the Robinson SPRA for mitigating seismic-induced failures. The key functions of FLEX credited for the Robison SPRA are the SG makeup and RCS makeup capabilities [77].

5.1.4 Human Reliability Analysis (HR)

Pre-initiator actions are not affected by seismic events and so their assessments were not changed from the internal events PRA model. The list of post initiator human actions for the internal events model was analyzed for modification due to seismic impacts. Only human failure events (HFE) associated with the sequence models used to represent seismic initiators were retained in the model and any new operator actions added for the SPRA.

Post-initiator HFEs retained in the SPRA cutsets were evaluated for the impacts of seismic events. The degree of impact was assumed dependent on the seismic acceleration level. At very high accelerations, the human error probabilities (HEPs) were set to 1.0. The seismic impacts on every post-initiator HFE in the SPRA models is accounted for by the HFE specific, performance shaping factors, and selected minimal values that increase with acceleration as a function of plant damage as documented in the station calculation for HRA [65].

5.1.5 Data Analysis (DA)

Equipment failure data for random failures, test and maintenance unavailabilities, and plant configuration data are unchanged from the internal events PRA model. The increasing SSC seismic failure probabilities with acceleration interval are computed from the fragility curves developed in the fragility analysis. Equations are developed in terms of the A_m, β_r, and β_u parameters of the SSC seismic fragility curves, and the failure probability of each new seismic event is evaluated using these equations as a function of the seismic acceleration level that applies.

5.1.6 Quantification (QU)

The 10 seismic interval frequencies are included as separate initiating events. Each SSC seismic failure is combined in a single OR gate with the associated seismic initiating event under an AND gate. The placements of these OR gates in the single linked fault tree is dictated by the BE(s) associated with the SSC. An SSC may impact more than one BE, there may be more than one failure mode for a basic event, and a single BE in the linked fault tree may be impacted by more than one SSC failed by a seismic event.

The quantification of the SPRA fault tree is accomplished using standard EPRI developed software used to analyze risk from external events at power plants. The seismic PRA quantification notebook [13] describes the use of this software for seismic events in detail.

5.1.7 Seismic Sequence Model

Seismic Initiating Event Impacts

The seismic initiating events are addressed by the seismic master event tree in the station calculation for plant logic modeling [77]. This event tree maps a seismic initiating event to an internal events initiating event based on the availability of certain SSCs. The seismic master event tree transfers to the following internal events initiating events or combination of internal events initiating events:

- Reactor / Turbine Trip (%T1)
- Loss of Main Feedwater (%T4)
- Loss of Offsite Power (%T5G)

- Loss of Service Water (%T9)
- S1LOCA (%S1)
- S2LOCA (%S2)
- Medium LOCA (%M)
- Large LOCA (%A)
- Vessel Rupture (%EXLOCA) (surrogate for direct core damage from failure of NSSS components beyond mitigation capability of the ECCS)

Seismic Initiating Event Frequencies

The seismic hazard curve is shown in Figure 3-2. The 100 Hz spectral acceleration is selected to represent the zero period acceleration or PGA. From the hazard curve the RNP Safe Shutdown Earthquake (SSE) at 0.20g has a mean hazard exceedance frequency of 2.64E-4 per year. The hazard exceedance frequency is at 1.75E-5 at 0.5g and 1.24E-6 per year at 1.0g.

The seismic initiating event frequencies and their associated acceleration intervals are found in Table 5.1-1. The lowest acceleration chosen was 0.1g. Relatively narrow acceleration intervals were selected for those ranges of acceleration where the conditional core damage probability was expected to change most quickly, and to aid in the demonstration that adding new SSC fragilities with higher capacity would not significantly impact the computed CDF. The higher end of the range of accelerations were retained to evaluate LERF.

Table 5.1-1: Seismic Initiating Event Intervals

IE	PGA Lower	PGA Upper	IE Frequency
%G01	0.10	0.15	5.67E-04
%G02	0.15	0.20	2.49E-04
%G03	0.20	0.25	1.16E-04
%G04	0.25	0.30	5.86E-05
%G05	0.30	0.35	3.26E-05
%G06	0.35	0.40	1.94E-05
%G07	0.40	0.50	1.98E-05
%G08	0.50	0.60	8.47E-06
%G09	0.60	0.90	7.16E-06
%G10	0.90	---	1.88E-06

Fault Tree Model Changes

Using FRANX minimizes the need to make changes to the RNP fault tree to model the plant response to a seismic event. However, some logic adjustment is needed to control the fault tree logic used by FRANX for the SPRA.

- Basic Event S-DAM is added to model dam failure failing fire water and service water.

- Basic Event DUMMY_CD was inserted as input to the core damage top event to serve as a placeholder to map seismic failures that lead (or are assumed to lead) directly to core damage, such as structural failure of the containment building.
- Logic was added to the fault tree to credit the FLEX system. The FLEX system is credited with providing backup suction for the SDAFW pump, alternate reactor coolant system (RCS) boration and makeup, and CST makeup. Logic was also added to the fault tree for FLEX alternate auxiliary feedwater to the steam generators.

A complete list of fault tree changes is documented in the station calculation for the plant logic modeling [77].

5.1.8 SSC Correlation

SSCs retained in the model were assigned to correlation sets by their seismic capacities and failure modes. Equipment is considered correlated for seismically induced failures if they meet all of the following four conditions: 1) located in the same building, 2) located on the same elevation in the building, 3) identical or essentially identical equipment and 4) the same orientation.

The model assumes complete correlation, which means that if one equipment item in the fragility group fails seismically, all others in that set are also assumed to fail. This 100% correlation approach conservatively minimizes the advantages of redundancy. Groups of equipment with different failure modes were split into different fragility groups because different failure modes are not correlated; for example, a functional failure of a panel and failure due to failure of the block wall on which it is mounted were put in different fragility groups. Further, potentially risk-significant fragility groups with significantly different fragilities were split into multiple fragility groups based on their seismic capacity and analyst judgement, supported by justifications provided by the fragility analysts. Once fragility groups were finalized, fragility values larger than the lowest value within the group were changed to match the most conservative value in the group. The final list of fragility groups and their numerical characterization are located in the station calculation for the plant logic modeling [77].

Relays were correlated, if they were 1) located in the same host equipment, 2) were the same relay make and 3) were the same relay model. If relays failed any of these checks, they were considered uncorrelated.

5.1.9 Seismic-Induce Floods and Fires

The fire PRA analysis existing at the time of the fire/flood seismic walkdowns was used to develop the list of fire ignition sources for the seismic fire assessment for RNP SPRA. The approach taken for seismic, focused on a sub-set of fire ignition sources identified from the Fire PRA that have the potential to become fire sources in the seismic context. This took advantage of insights gained from a systematic review of fire PRA results and past fire evaluations with respect to the type of ignition sources relevant to seismic and resulted in identifying flammable gases and liquids as representative seismic-fire ignition sources to be considered for SPRA. These flammable gas/liquid sources were determined to be Hydrogen Piping, Station Transformers, Diesel Fuel Oil Storage Tank, Diesel Fuel Oil Day Tank, Turbine Lube Oil Reserve and Lube Oil Storage Tank, Engine-Driven Fire Pump and Flammable Material Storage Cabinet. In addition, general plant areas and rooms were walked down as documented in the station calculation for the SPRA walkdowns [53] to assess whether any additional flammable gas/liquid sources existed in the plant that were not already identified from the fire PRA analysis existing at the time. When the SRT walked by each area, they looked for any credible seismic fire concerns related to flammable gas/liquid sources and documented their observations and findings in a SEWS. Based on the results of the seismic induced fire assessment, there were no seismic induced fire scenarios identified for inclusion with the SPRA.

Seismic component walkdowns, area walkdowns and reviews of past internal flood analysis [80] were systematically utilized to support the identification of potential seismic-induced flood sources to be added to the SEL. However, the SPRA approach was not to assess/screen flood sources just based on the Internal Flooding PRA, but to use the walkdowns to identify piping with the unusual features that could cause failure from a seismic event. In other words, the intent is not necessarily to itemize and disposition every passive component in the plant. For example, small tanks of limited volume that pose no significant flood hazard, and do not fail systems of interest to the SPRA, need not be itemized and individually dispositioned. To this end, area flooding walkdown reviews were conducted by the SRT and SEWS developed for Fire protection piping and other flooding sources beyond the SEL items. It should be noted that, based on experience with other SPRAs, particular attention was given to the Fire Water lines in RAB, since there can be the potential for threaded joints and piping interactions for the non-seismic design piping, and the size of the flood can be large. The seismic capacity walkdown team observed that this piping is adequately supported and free of vulnerabilities such as flexible headers with stiff branch lines and insufficient clearance between sprinkler heads and adjacent objects. For sprays, the equipment walkdowns included assessing potential piping failures near the SEL equipment. The SEL has entries for flood and fire potential for all of the area walkdowns. After a series of qualitative and quantitative screening steps taken to disposition all identified flooding sources on the SEL, the Fire Water lines in RAB was identified as one of flooding sources that were ultimately retained in the SPRA for quantification purposes as documented in the station calculation for the Seismic Induced Fire/Flood Assessment [69].

5.2 SPRA Plant Seismic Logic Model Technical Adequacy

The Robinson Nuclear Power Plant SPRA seismic plant response methodology and analysis were subjected to an independent peer review against the pertinent requirements in the PRA Standard [37].

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

5.3 Seismic Risk Quantification

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

5.3.1 SPRA Quantification Methodology

The Robinson SPRA model has been created using the linked fault tree approach. The EPRI R&R workstation code package is used to support development and quantification of the model. The analytic tools for the development of a quantified model are the EPRI CAFTA code suite augmented by EPRI's ACUBE Binary Decision Diagram (BDD) software.

5.3.2 SPRA Model and Quantification Assumptions

The following assumptions were used in development of the RNP SPRA model:

- The Internal Events PRA is used as the technical basis for both CDF and LERF. All assumptions and success criteria in the Internal Events PRA are retained in the SPRA for the portions of the sequence models that apply [77]. This assumption provides continuity between the Internal Events PRA and the SPRA. Any future changes to the Internal Events PRA success criteria would be addressed as part of the maintenance and update process of the integrated PRA.
- A plant trip is assumed for all seismic initiating events.
- The portions of the internal events PRA model that apply to seismic events are the following: transients (including loss of offsite power (LOOP), secondary line breaks, loss of service water, loss of CCW, loss of Instrument Air, loss of CVCS, and loss of feedwater), loss of coolant accidents (LOCAs) (small, medium, large, and interfacing-system), reactor vessel rupture, internal fire, and internal flood.
- The screening criterion for excluding structures, systems, and components (SSCs) from the SPRA model is an SSC High Confidence of a Low Probability of Failure (HCLPF) peak ground acceleration (PGA) of 0.75g or higher as documented in the station calculation for the logic modeling notebook [77].
- Seismic SSC failures are assumed to be complete failures, in that the SSC fails to perform its function. Degraded states of equipment for the period following the seismic initiator are not represented.
- The assumed SSC seismic failure mode depends on the SSC type and whether the fragility applies to functional failure, structural failure, or block wall or other interaction failure.
- Many SSCs are seismically rugged, and therefore the failure probability is unchanged from the internal events PRA (e.g., check valves, manual valves, filters, dampers, and sensors).
- If the seismic event results in a LOOP, recovery of offsite power is not credited in the SPRA during the first 24 hours after the seismic event.
- Building fragilities are included in the model. Failure of key structures (Auxiliary Building, Reactor Containment Building, and Reactor Containment Building) are conservatively assumed to lead to core damage. Failure of key structures is also assumed to lead to a large early release. Other building failures lead to failure of the equipment function in the building (e.g., failure of the dam leads to loss of service water and fire water).
- Non-rugged/offsite power dependent equipment screened from the SEL [64] is not credited in the model. Therefore, this equipment is assumed failed in the model. These include such SSCs as Circulating Water pumps, Main Feedwater and Condensate equipment, primary instrument air train, “D” instrument air train, Fire Water equipment, and primary water equipment.

5.4 SCDF Results

The seismic PRA performed for Robinson Nuclear Power Plant shows that the point estimate mean seismic CDF is 9.27xE-05/reactor-year. A discussion of the mean SCDF with uncertainty distribution reflecting the uncertainties in the hazard, fragilities, and model data is presented in Section 5.6. Important contributors are discussed in the following paragraphs.

The top SCDF cutsets are documented in the SPRA quantification report [13]. These are briefly summarized in Table 5.4-1.

SSCs with the most significant seismic failure contributions to SCDF are listed in Table 5.4-2, sorted by Fussell-Vesely (FV). 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.

The FV for each fragility basic event, has been calculated by summing the criticality in each interval from the ACUBE output. The top ten events for the SCDF are described here.

As can be seen from Table 5.4-2, the RNP SCDF is dominated (43.71%) by the pounding-induced cracking failure of the Class III Turbine Building (TB3); this is a loss of structural integrity due to the Turbine Building Mezzanine floor pounding into the RAB. Failure of TB3 leads to a LOOP and failure of AFW C, as well as a high chance of failure of the Class I Turbine Building (TB1), which houses the piping required to provide flow to the steam generators and piping from the CST. The second highest contributor (12.57%) is the liquefaction-induced settlement failure of the DFOST piping leading to failure of the EDGs. Contributor #3 (11.81%) is the failure of the TB gantry crane which has a high probability of interacting with and failing the CST. Contributor #4 (7.01%) is the liquefaction-induced settlement failure of the SDAFW pump piping. [redacted]. Contributor #6 (2.59%) is the liquefaction-induced settlement failure of underground cable trays running from the Intake Structure to the RAB, which contain cables for the SW system and is assumed to fail SW. Contributors #7 and #8 (1.87%) are two separate relay chatter groups which cause a loss of the EDGs. Contributor #9 (1.70%) is the liquefaction-induced settlement failure of North Header SW piping at the Intake Structure, which leads to the loss of one of two SW headers. Contributor #10 (1.54%) is failure of SSCs associated with offsite power.

Table 5.4-1: Summary of Top SCDF Cutsets

#	Cutset Prob	Event Prob	Event	Description	Cutset Description
1	8.32E-06	1.98E-05	%G07	Seismic Initiating Event (0.4g to <0.5g)	Bin %G07 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3) due to pounding-induced cracking, which leads to a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. The TB Gantry Crane fails and interacts with and fails the CST failing the normal supply to the SDAFW; failure of TB3 prevents alignment of the Condenser Water Box to the SDAFW pump suction. Loss of all injection/recirculation and SSHR leads to core damage.
		9.13E-01	SF-TB_CRANE-C-G07	SEISMIC FRAGILITY FOR %G07: Turbine Building Gantry Crane	
		9.52E-01	SF-TB-CLASS-3-POUND-C-G07	SEISMIC FRAGILITY FOR %G07: Turbine Building Class 3 - Pounding-induced cracking	
		7.20E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G07	DFOST Liquefaction-Induced Settlement	
		7.50E-01	TBCRANE-CST	TB Crane Failure Interacts with CST Causing Failure	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
2	6.08E-06	1.98E-05	%G07	Seismic Initiating Event (0.4g to <0.5g)	Bin %G07 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3) due to pounding-induced cracking, which leads to a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. TB3 interacts with and fails TB1, which fails the SDAFW pump. Loss of all injection/recirculation and SSHR leads to core damage.
		9.52E-01	SF-TB-CLASS-3-POUND-C-G07	SEISMIC FRAGILITY FOR %G07: Turbine Building Class 3 - Pounding-induced cracking	
		7.20E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G07	DFOST Liquefaction-Induced Settlement	
		5.00E-01	TB-CLASS3-1	50% chance TB Class 3 interacts with TB Class 1	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
3	5.77E-06	3.26E-05	%G05	Seismic Initiating Event (0.3g to <0.35g)	Bin %G05 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3) due to pounding-induced cracking, which leads to a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. TB3 interacts with and fails TB1, which fails the SDAFW pump. Loss of all injection/recirculation and SSHR leads to core damage.
		6.98E-01	SF-TB-CLASS-3-POUND-C-G05	SEISMIC FRAGILITY FOR %G05: Turbine Building Class 3 - Pounding-induced cracking	
		5.66E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G05	DFOST Liquefaction-Induced Settlement	
		5.00E-01	TB-CLASS3-1	50% chance TB Class 3 interacts with TB Class 1	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
4	5.50E-06	3.26E-05	%G05	Seismic Initiating Event (0.3g to <0.35g)	Bin %G05 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3) due to pounding-induced cracking, which leads to a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. The TB Gantry Crane fails and interacts with and fails the CST, failing the normal supply to the SDAFW; failure of
		6.36E-01	SF-TB_CRANE-C-G05	SEISMIC FRAGILITY FOR %G05: Turbine Building Gantry Crane	

Table 5.4-1: Summary of Top SCDF Cutsets

#	Cutset Prob	Event Prob	Event	Description	Cutset Description
		6.98E-01	SF-TB-CLASS-3-POUND-C-G05	SEISMIC FRAGILITY FOR %G05: Turbine Building Class 3 - Pounding-induced cracking	TB3 prevents alignment of the Condenser Water Box to the SDAFW pump suction. Loss of all injection/recirculation and SSHR leads to core damage.
		5.66E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G05	DFOST Liquefaction-Induced Settlement	
		7.50E-01	TBCRANE-CST	TB Crane Failure Interacts with CST Causing Failure	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
5	5.48E-06	1.94E-05	%G06	Seismic Initiating Event (0.35g to <0.4g)	Bin %G06 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3) due to pounding-induced cracking, which leads to a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. The TB Gantry Crane fails and interacts with and fails the CST, failing the normal supply to the SDAFW; failure of TB3 prevents alignment of the Condenser Water Box to the SDAFW pump suction. Loss of all injection/recirculation and SSHR leads to core damage.
		7.88E-01	SF-TB_CRANE-C-G06	SEISMIC FRAGILITY FOR %G06: Turbine Building Gantry Crane	
		8.48E-01	SF-TB-CLASS-3-POUND-C-G06	SEISMIC FRAGILITY FOR %G06: Turbine Building Class 3 - Pounding-induced cracking	
		6.30E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G06	DFOST Liquefaction-Induced Settlement	
		7.50E-01	TBCRANE-CST	TB Crane Failure Interacts with CST Causing Failure	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
6	5.20E-06	1.98E-05	%G07	Seismic Initiating Event (0.4g to <0.5g)	Bin %G07 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3) due to pounding-induced cracking, which leads to a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. Liquefaction-induced settlement failures of the SDAFW pump. Loss of all injection/recirculation and SSHR leads to core damage.
		9.52E-01	SF-TB-CLASS-3-POUND-C-G07	SEISMIC FRAGILITY FOR %G07: Turbine Building Class 3 - Pounding-induced cracking	
		7.20E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G07	DFOST Liquefaction-Induced Settlement	
		4.28E-01	SF-TP-SDAFW-PMP_SETTLE_G07	SDAFW Liquefaction-Induced Settlement	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
7	5.11E-06	5.86E-05	%G04	Seismic Initiating Event (0.25g to <0.3g)	Bin %G04 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3) due to pounding-induced cracking, which leads to a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. TB3 interacts with and fails TB1, which fails the SDAFW pump. Loss of all injection/recirculation and SSHR leads to core damage.
		4.69E-01	SF-TB-CLASS-3-POUND-C-G04	SEISMIC FRAGILITY FOR %G04: Turbine Building Class 3 - Pounding-induced cracking	
		4.15E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G04	DFOST Liquefaction-Induced Settlement	

Table 5.4-1: Summary of Top SCDF Cutsets

#	Cutset Prob	Event Prob	Event	Description	Cutset Description
		5.00E-01	TB-CLASS3-1	50% chance TB Class 3 interacts with TB Class 1	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
8	4.89E-06	7.16E-06	%G09	Seismic Initiating Event (0.6g to <0.9g)	Bin %G09 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3) due to pounding-induced cracking, which leads to a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. CST seismically fails leading to failure of the normal supply to the SDAFW; failure of TB3 prevents alignment of the Condenser Water Box to the SDAFW pump suction. Loss of all injection/recirculation and SSHR leads to core damage.
		1.00E+00	SF-TB-CLASS-3-POUND-C-G09	SEISMIC FRAGILITY FOR %G09: Turbine Building Class 3 - Pounding-induced cracking	
		7.82E-01	SF-TK-COND-STRG-TNK-C-G09	SEISMIC FRAGILITY FOR %G09: Condensate Storage Tank	
		9.76E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G09	DFOST Liquefaction-Induced Settlement	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	[redacted]
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	[redacted]
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	

Table 5.4-1: Summary of Top SCDF Cutsets

#	Cutset Prob	Event Prob	Event	Description	Cutset Description
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
12	4.69E-06	7.16E-06	%G09	Seismic Initiating Event (0.6g to <0.9g)	Bin %G09 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3) due to pounding-induced cracking, which leads to a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. The TB Gantry Crane fails and interacts with and fails the CST, failing the normal supply to the SDAFW; failure of TB3 prevents alignment of the Condenser Water Box to the SDAFW pump suction. Loss of all injection/recirculation and SSHR leads to core damage.
		9.98E-01	SF-TB_CRANE-C-G09	SEISMIC FRAGILITY FOR %G09: Turbine Building Gantry Crane	
		1.00E+00	SF-TB-CLASS-3-POUND-C-G09	SEISMIC FRAGILITY FOR %G09: Turbine Building Class 3 - Pounding-induced cracking	
		9.76E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G09	DFOST Liquefaction-Induced Settlement	
		7.50E-01	TBCRANE-CST	TB Crane Failure Interacts with CST Causing Failure	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
13	4.66E-06	7.16E-06	%G09	Seismic Initiating Event (0.6g to <0.9g)	Bin %G09 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3), which leads to a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. CST seismically fails leading to failure of the normal supply to the SDAFW; failure of TB3 prevents alignment of the Condenser Water Box to the SDAFW pump suction. Loss of all injection/recirculation and SSHR leads to core damage.
		9.52E-01	SF-TB-CLASS-3-C-G09	SEISMIC FRAGILITY FOR %G09: Turbine Building Class 3	
		7.82E-01	SF-TK-COND-STRG-TNK-C-G09	SEISMIC FRAGILITY FOR %G09: Condensate Storage Tank	
		9.76E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G09	DFOST Liquefaction-Induced Settlement	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
14	4.66E-06	8.47E-06	%G08	Seismic Initiating Event (0.5g to <0.6g)	Bin %G08 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3) due to pounding-induced cracking, which leads to

Table 5.4-1: Summary of Top SCDF Cutsets

#	Cutset Prob	Event Prob	Event	Description	Cutset Description
		9.77E-01	SF-TB_CRANE-C-G08	SEISMIC FRAGILITY FOR %G08: Turbine Building Gantry Crane	a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. The TB Gantry Crane fails and interacts with and fails the CST failing the normal supply to the SDAFW; failure of TB3 prevents alignment of the Condenser Water Box to the SDAFW pump suction. Loss of all injection/recirculation and SSHR leads to core damage.
		9.91E-01	SF-TB-CLASS-3-POUND-C-G08	SEISMIC FRAGILITY FOR %G08: Turbine Building Class 3 - Pounding-induced cracking	
		8.45E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G08	DFOST Liquefaction-Induced Settlement	
		7.50E-01	TBCRANE-CST	TB Crane Failure Interacts with CST Causing Failure	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
15	4.64E-06	1.94E-05	%G06	Seismic Initiating Event (0.35g to <0.4g)	Bin %G06 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3) due to pounding-induced cracking, which leads to a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. TB3 interacts with and fails TB1, which fails the SDAFW pump. Loss of all injection/recirculation and SSHR leads to core damage.
		8.48E-01	SF-TB-CLASS-3-POUND-C-G06	SEISMIC FRAGILITY FOR %G06: Turbine Building Class 3 - Pounding-induced cracking	
		6.30E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G06	DFOST Liquefaction-Induced Settlement	
		5.00E-01	TB-CLASS3-1	50% chance TB Class 3 interacts with TB Class 1	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	

Table 5.4-1: Summary of Top SCDF Cutsets					
#	Cutset Prob	Event Prob	Event	Description	Cutset Description
		[redacted]	[redacted]	[redacted]	[redacted]
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
19	4.51E-06	7.16E-06	%G09	Seismic Initiating Event (0.6g to <0.9g)	Bin %G09 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3) due to pounding-induced cracking, which leads to a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. Liquefaction-induced settlement failures of the SDAFW pump. Loss of all injection/recirculation and SSHR leads to core damage.
		1.00E+00	SF-TB-CLASS-3-POUND-C-G09	SEISMIC FRAGILITY FOR %G09: Turbine Building Class 3 - Pounding-induced cracking	
		9.76E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G09	DFOST Liquefaction-Induced Settlement	
		7.21E-01	SF-TP-SDAFW-PMP_SETTLE_G09	SDAFW Liquefaction-Induced Settlement	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
20	4.46E-06	7.16E-06	%G09	Seismic Initiating Event (0.6g to <0.9g)	Bin %G09 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3), which leads to a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. The TB Gantry Crane fails and interacts with and fails the CST failing the normal supply to the SDAFW; failure of TB3 prevents alignment of the Condenser Water Box to the SDAFW pump suction. Loss of all injection/recirculation and SSHR leads to core damage.
		9.98E-01	SF-TB_CRANE-C-G09	SEISMIC FRAGILITY FOR %G09: Turbine Building Gantry Crane	
		9.52E-01	SF-TB-CLASS-3-C-G09	SEISMIC FRAGILITY FOR %G09: Turbine Building Class 3	
		9.76E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G09	DFOST Liquefaction-Induced Settlement	
		7.50E-01	TBCRANE-CST	TB Crane Failure Interacts with CST Causing Failure	
		8.96E-01	X-POWEROP	Plant Capacity Factor	

Table 5.4-1: Summary of Top SCDF Cutsets

#	Cutset Prob	Event Prob	Event	Description	Cutset Description
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
22	4.30E-06	7.16E-06	%G09	Seismic Initiating Event (0.6g to <0.9g)	Bin %G09 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3), which leads to a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. Liquefaction-induced settlement failures of the SDAFW pump. Loss of all injection/recirculation and SSHR leads to core damage.
		9.52E-01	SF-TB-CLASS-3-C-G09	SEISMIC FRAGILITY FOR %G09: Turbine Building Class 3	
		9.76E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G09	DFOST Liquefaction-Induced Settlement	
		7.21E-01	SF-TP-SDAFW-PMP_SETTLE_G09	SDAFW Liquefaction-Induced Settlement	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	

[redacted]

Table 5.4-2: Top 10 SCDF Importance Measures Ranked by FV - Seismic Failures							
Component	**Description**	**FV**	**Median Capacity (g)**	**β_r**	**β_u**	**Failure Mode[2]**	**Fragility Method[2]**
SF-TB-CLASS-3-POUND	Turbine Building Class 3 - Pounding-induced cracking	43.71%	0.28	0.13	0.25	RAB pounding induced cracking and splitting of the mezzanine floor slab resulting in loss of structural integrity.	SOV
SF-TK-DG-FOSTRG-TNK_SETTLE	DFOST Liquefaction-Induced Settlement	12.57%	N/A[1]	N/A	N/A	Failure of EDG-B pipe at RAB penetration. Probability of failure at PGA: PGA(g) - Probability 0.265g - 0.388 0.325g - 0.569 0.582g - 0.887 0.717g - 0.966	SOV
SF-TB_CRANE	Turbine Building Gantry Crane	11.81%	0.29	0.21	0.24	Failure of A-frame anchor bolts	SOV
SF-TP-SDAFW-PMP_SETTLE	SDAFW Liquefaction-Induced Settlement	7.01%	N/A[1]	N/A	N/A	Failure of AFW discharge piping where the 4 in. diameter pipe meets the 6x4 reducing elbow. Probability of failure at PGA: PGA(g) - Probability 0.265g - 0.168 0.325g - 0.263 0.582g - 0.609 0.717g - 0.708	SOV
[redacted]							

Table 5.4-2: Top 10 SCDF Importance Measures Ranked by FV - Seismic Failures							
Component	**Description**	**FV**	**Median Capacity (g)**	**β_r**	**β_u**	**Failure Mode[2]**	**Fragility Method[2]**
SF-TR-UG-INTAKE_SETTLE	Underground Cable Trays at Intake Liquefaction-Induced Settlement	2.59%	N/A[1]	N/A	N/A	Liquefaction-induced settlement failure at Intake Structure. Probability of failure at PGA: PGA(g) - Probability 0.265g - 0.110 0.325g - 0.180 0.582g - 0.450 0.717g - 0.570	Rep.
SF-RC-20	Relay Chatter - DG-A,B-AUX-PNL_Barksdale Controls_D2T-M18SS	1.87%	0.95	0.27	0.9	Relay malfunction due to earthquake shaking	SOV
SF-RC-21	Relay Chatter - DG-A.B-ENG-PNL_Barksdale Controls_D2T-M80SS	1.87%	0.95	0.27	0.9	Relay malfunction due to earthquake shaking	SOV
SF-PIP-UG-N_INTAKE3_SETTLE	North Header Intake Mech 3 Liquefaction-Induced Settlement	1.70%	N/A[1]	N/A	N/A	North Header - Mechanism 3. Probability of failure at PGA: PGA(g) - Probability 0.265g - 0.223 0.325g - 0.390 0.582g - 0.722 0.717g - 0.824	SOV
SF-SLOSP	Seismic-Induced Loss of Offsite Power	1.54%	0.3	0.3	0.45	SPRAIG values for offsite power.	Rep.

Note 1: Liquefaction-Induced Settlement failures do not utilize median capacity or β-values. The probability of failure at a given PGA is provided in the Failure Mode column.
Note 2: Fragility Mode and Fragility Method per [82].

The most significant non-seismic SSC failures (e.g., random failures of modeled components during the SPRA mission time) are listed in Table 5.4-3. A summary of the SCDF results for each seismic hazard interval is presented in Table 5.4-4.

Table5.4-3: Top 10 SCDF Importance Measures Ranked by FV - Non-Seismic Failures

Component	Description and Failure Mode	FV
FPT1XSABFR	TURBINE-DRIVEN PUMP FAILS TO RUN	0.73%
OPER-61	Failure to align and start pre-staged pumps for SG makeup - Condenser Inlet Waterbox (FLEX)	0.68%
FPMCIW-LFS	CONDENSER WATERBOX INLET MOTOR PUMP CIW-L FAILS TO START (FLEX)	0.57%
FPT1XSABFS	TURBINE-DRIVEN PUMP FAILS TO START	0.43%
OPER-64	Failure to align and start portable pumps to lake for long-term water source - SG makeup (FLEX)	0.42%
XOPERC-1	Dependent HEP for OPER-35,OPER-68,OPER-18B-S1,OPER-64,OPER-01S	0.33%
OPER-14	OPERATOR FAILS TO TRANSFER POWER TO DEEPWELL PUMP DIESEL	0.33%
FTMSDPTRXM	AFW TD PUMP TRAIN C UNAVAILABLE	0.30%
FPMPLAKEFS	PORTABLE LAKE PUMP FAILS TO START (FLEX)	0.26%
OPER-13S	OPERATOR FAILS TO ALIGN AFW PUMP C	0.18%

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

Table 5.4-4 Contribution to SCDF by Acceleration Interval

Hazard Interval Description	SCDF	% of Total SCDF	Cumulative CDF
Hazard Curve: HAZARD - PGA Range: 0.1g to 0.15g	5.17E-07	1%	5.17E-07
Hazard Curve: HAZARD - PGA Range: 0.15g to 0.2g	2.13E-06	2%	2.64E-06
Hazard Curve: HAZARD - PGA Range: 0.2g to 0.25g	9.14E-06	10%	1.18E-05
Hazard Curve: HAZARD - PGA Range: 0.25g to 0.3g	1.66E-05	18%	2.84E-05
Hazard Curve: HAZARD - PGA Range: 0.3g to 0.35g	1.77E-05	19%	4.61E-05
Hazard Curve: HAZARD - PGA Range: 0.35g to 0.4g	1.41E-05	15%	6.02E-05
Hazard Curve: HAZARD - PGA Range: 0.4g to 0.5g	1.69E-05	18%	7.72E-05
Hazard Curve: HAZARD - PGA Range: 0.5g to 0.6g	7.54E-06	8%	8.46E-05
Hazard Curve: HAZARD - PGA Range: 0.6g to 0.9g	6.42E-06	7%	9.10E-05
Hazard Curve: HAZARD - PGA Range: > 0.9g	1.68E-06	2%	9.27E-05

5.5 SLERF Results

The seismic PRA performed for Robinson Nuclear Power Plant shows that the point estimate mean seismic LERF is 2.02E-05/reactor-year. A discussion of the mean SLERF with uncertainty distribution reflecting the uncertainties in the hazard, fragilities, and model data is presented in Section 5.6. Important contributors are discussed in the following paragraphs.

The top SLERF cutsets are documented in the SPRA quantification report [13]. These are briefly summarized in Table 5.5-1.

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

The top ten events for the SLERF are described here. As can be seen from Table 5.5-2, the RNP SLERF is dominated (29.59%) by the pounding-induced cracking failure of the Class III Turbine Building (TB3); this is a loss of structural integrity due to the Turbine Building Mezzanine floor pounding into the RAB. Failure of TB3 leads to a LOOP and failure of AFW C, as well as a high chance of failure of the Class I Turbine Building (TB1), which houses the piping required to provide flow to the steam generators and piping from the CST. The second highest contributor (11.85%) is the failure of the Reactor Containment Building, which is assumed to lead directly to LERF. Contributor #3 (8.03%) is the liquefaction-induced settlement failure of the DFOST piping leading to failure of the EDGs. Contributor #4 (7.67%) is the failure of the TB gantry crane which has a high probability of interacting with and failing the CST. Contributor #5 (5.02%) is the liquefaction-induced lateral spreading of Displacement Category 2, which includes the Reactor Auxiliary Building; and thus, is assumed to lead directly to LERF. Contributor #6 [redacted] Contributor #7 (4.19%) is the liquefaction-induced settlement failure of the SDAFW pump piping. Contributor #8 (2.74%) is the failure of the Reactor Auxiliary Building, which is assumed to lead directly to LERF. Contributor #9 (2.45%) are two separate relay chatter groups which cause a loss of the EDGs. Contributor #9 (2.45%) is the liquefaction-induced settlement failure of underground cable trays running from the Intake Structure to the RAB, which contain cables for the SW system and is assumed to fail SW. Contributor #10 (2.07%) is the liquefaction-induced failure of Deepwell D piping.

Table 5.5-1: Summary of Top SLERF Cutsets

#	Cutset Prob	Event Prob	Event	Description	Cutset Description
1	2.12E-06	7.16E-06	%G09	Seismic Initiating Event (0.6g to <0.9g)	Bin %G09 seismic event causes a plant trip. There is a seismic failure of the RCB, which is assumed to lead directly to core damage and LERF.
		3.31E-01	SF-RCB-C-G09	SEISMIC FRAGILITY FOR %G09: REACTOR CONTAINMENT BUILDING (RCB)	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
2	1.14E-06	1.88E-06	%G10	Seismic Initiating Event (>0.9g)	Bin %G10 seismic event causes a plant trip. There is a seismic failure of the RCB, which is assumed to lead directly to core damage and LERF.
		6.77E-01	SF-RCB-C-G10	SEISMIC FRAGILITY FOR %G10: REACTOR CONTAINMENT BUILDING (RCB)	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
3	8.22E-07	1.98E-05	%G07	Seismic Initiating Event (0.4g to <0.5g)	Bin %G07 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3) due to pounding-induced cracking, which leads to a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. The TB Gantry Crane fails and interacts with and fails the CST failing the normal supply to the SDAFW; failure of TB3 prevents alignment of the Condenser Water Box to the SDAFW pump suction. Loss of all injection/recirculation and SSHR leads to core damage. Loss of power leads to loss of containment coolers and containment spray injection, which leads to LERF.
		9.13E-01	SF-TB_CRANE-C-G07	SEISMIC FRAGILITY FOR %G07: Turbine Building Gantry Crane	
		9.52E-01	SF-TB-CLASS-3-POUND-C-G07	SEISMIC FRAGILITY FOR %G07: Turbine Building Class 3 - Pounding-induced cracking	
		7.20E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G07	DFOST Liquefaction-Induced Settlement	
		7.50E-01	TBCRANE-CST	TB Crane Failure Interacts with CST causing Failure	
		9.88E-02	XFL_PDS3P	Plant Damage State 3P	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
4	7.49E-07	7.16E-06	%G09	Seismic Initiating Event (0.6g to <0.9g)	Bin %G09 seismic event causes a plant trip. There is a seismic failure of the RAB, which is assumed to lead directly to core damage and LERF.
		1.17E-01	SF-RAB-C-G09	SEISMIC FRAGILITY FOR %G09: Reactor Auxiliary Building	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
5	7.09E-07	8.47E-06	%G08	Seismic Initiating Event (0.5g to <0.6g)	Bin %G08 seismic event causes a plant trip. There is a seismic failure of the RCB, which is assumed to lead directly to core damage and LERF.

Table 5.5-1: Summary of Top SLERF Cutsets					
#	Cutset Prob	Event Prob	Event	Description	Cutset Description
		9.34E-02	SF-RCB-C-G08	SEISMIC FRAGILITY FOR %G08: REACTOR CONTAINMENT BUILDING (RCB)	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
6	6.11E-07	1.88E-06	%G10	Seismic Initiating Event (>0.9g)	Bin %G10 seismic event causes a plant trip. There is a seismic failure of the RAB, which is assumed to lead directly to core damage and LERF.
		3.64E-01	SF-RAB-C-G10	SEISMIC FRAGILITY FOR %G10: Reactor Auxiliary Building	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
7	6.01E-07	1.98E-05	%G07	Seismic Initiating Event (0.4g to <0.5g)	Bin %G07 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3) due to pounding-induced cracking, which leads to a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. TB3 interacts with and fails TB1, which fails the SDAFW pump. Loss of all injection/recirculation and SSHR leads to core damage. Loss of power leads to loss of containment coolers and containment spray injection, which leads to LERF.
		9.52E-01	SF-TB-CLASS-3-POUND-C-G07	SEISMIC FRAGILITY FOR %G07: Turbine Building Class 3 - Pounding-induced cracking	
		7.20E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G07	DFOST Liquefaction-Induced Settlement	
		5.00E-01	TB-CLASS3-1	50% chance TB Class 3 interacts with TB Class 1	
		9.88E-02	XFL_PDS3P	Plant Damage State 3P	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
8	5.70E-07	3.26E-05	%G05	Seismic Initiating Event (0.3g to <0.35g)	Bin %G05 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3) due to pounding-induced cracking, which leads to a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. TB3 interacts with and fails TB1, which fails the SDAFW pump. Loss of all injection/recirculation and SSHR leads to core damage. Loss of power leads to loss of containment coolers and containment spray injection, which leads to LERF.
		6.98E-01	SF-TB-CLASS-3-POUND-C-G05	SEISMIC FRAGILITY FOR %G05: Turbine Building Class 3 - Pounding-induced cracking	
		5.66E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G05	DFOST Liquefaction-Induced Settlement	
		5.00E-01	TB-CLASS3-1	50% chance TB Class 3 interacts with TB Class 1	
		9.88E-02	XFL_PDS3P	Plant Damage State 3P	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
9	5.44E-07	3.26E-05	%G05	Seismic Initiating Event (0.3g to <0.35g)	Bin %G05 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3) due to pounding-induced cracking, which leads

Table 5.5-1: Summary of Top SLERF Cutsets

#	Cutset Prob	Event Prob	Event	Description	Cutset Description
		6.36E-01	SF-TB_CRANE-C-G05	SEISMIC FRAGILITY FOR %G05: Turbine Building Gantry Crane	to a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. The TB Gantry Crane fails and interacts with and fails the CST, failing the normal supply to the SDAFW; failure of TB3 prevents alignment of the Condenser Water Box to the SDAFW pump suction. Loss of all injection/recirculation and SSHR leads to core damage. Loss of power leads to loss of containment coolers and containment spray injection, which leads to LERF.
		6.98E-01	SF-TB-CLASS-3-POUND-C-G05	SEISMIC FRAGILITY FOR %G05: Turbine Building Class 3 - Pounding-induced cracking	
		5.66E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G05	DFOST Liquefaction-Induced Settlement	
		7.50E-01	TBCRANE-CST	TB Crane Failure Interacts with CST causing Failure	
		9.88E-02	XFL_PDS3P	Plant Damage State 3P	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
10	5.42E-07	1.94E-05	%G06	Seismic Initiating Event (0.35g to <0.4g)	Bin %G06 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3) due to pounding-induced cracking, which leads to a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. The TB Gantry Crane fails and interacts with and fails the CST, failing the normal supply to the SDAFW; failure of TB3 prevents alignment of the Condenser Water Box to the SDAFW pump suction. Loss of all injection/recirculation and SSHR leads to core damage. Loss of power leads to loss of containment coolers and containment spray injection, which leads to LERF.
		7.88E-01	SF-TB_CRANE-C-G06	SEISMIC FRAGILITY FOR %G06: Turbine Building Gantry Crane	
		8.48E-01	SF-TB-CLASS-3-POUND-C-G06	SEISMIC FRAGILITY FOR %G06: Turbine Building Class 3 - Pounding-induced cracking	
		6.30E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G06	DFOST Liquefaction-Induced Settlement	
		7.50E-01	TBCRANE-CST	TB Crane Failure Interacts with CST causing Failure	
		9.88E-02	XFL_PDS3P	Plant Damage State 3P	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
11	5.14E-07	1.98E-05	%G07	Seismic Initiating Event (0.4g to <0.5g)	Bin %G07 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3) due to pounding-induced cracking, which leads to a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. Liquefaction-induced settlement failures of the SDAFW pump. Loss of all injection/recirculation and SSHR leads to core damage. Loss of power leads to loss of containment coolers and containment spray injection, which leads to LERF.
		9.52E-01	SF-TB-CLASS-3-POUND-C-G07	SEISMIC FRAGILITY FOR %G07: Turbine Building Class 3 - Pounding-induced cracking	
		7.20E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G07	DFOST Liquefaction-Induced Settlement	
		4.28E-01	SF-TP-SDAFW-PMP_SETTLE_G07	SDAFW Liquefaction-Induced Settlement	
		9.88E-02	XFL_PDS3P	Plant Damage State 3P	
		8.96E-01	X-POWEROP	Plant Capacity Factor	

Table 5.5-1: Summary of Top SLERF Cutsets

#	Cutset Prob	Event Prob	Event	Description	Cutset Description
12	5.05E-07	5.86E-05	%G04	Seismic Initiating Event (0.25g to <0.3g)	Bin %G04 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3) due to pounding-induced cracking, which leads to a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. TB3 interacts with and fails TB1, which fails the SDAFW pump. Loss of all injection/recirculation and SSHR leads to core damage. Loss of power leads to loss of containment coolers and containment spray injection, which leads to LERF.
		4.69E-01	SF-TB-CLASS-3-POUND-C-G04	SEISMIC FRAGILITY FOR %G04: Turbine Building Class 3 - Pounding-induced cracking	
		4.15E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G04	DFOST Liquefaction-Induced Settlement	
		5.00E-01	TB-CLASS3-1	50% chance TB Class 3 interacts with TB Class 1	
		9.88E-02	XFL_PDS3P	Plant Damage State 3P	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
13	4.83E-07	7.16E-06	%G09	Seismic Initiating Event (0.6g to <0.9g)	Bin %G09 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3) due to pounding-induced cracking, which leads to a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. CST seismically fails, leading to failure of the normal supply to the SDAFW; failure of TB3 prevents alignment of the Condenser Water Box to the SDAFW pump suction. Loss of all injection/recirculation and SSHR leads to core damage. Loss of power leads to loss of containment coolers and containment spray injection, which leads to LERF.
		1.00E+00	SF-TB-CLASS-3-POUND-C-G09	SEISMIC FRAGILITY FOR %G09: Turbine Building Class 3 - Pounding-induced cracking	
		7.82E-01	SF-TK-COND-STRG-TNK-C-G09	SEISMIC FRAGILITY FOR %G09: Condensate Storage Tank	
		9.76E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G09	DFOST Liquefaction-Induced Settlement	
		9.88E-02	XFL_PDS3P	Plant Damage State 3P	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
14	4.77E-07	1.98E-05	%G07	Seismic Initiating Event (0.4g to <0.5g)	Bin %G07 seismic event causes a plant trip. There is a seismic failure of the RCB, which is assumed to lead directly to core damage and LERF.
		2.69E-02	SF-RCB-C-G07	SEISMIC FRAGILITY FOR %G07: REACTOR CONTAINMENT BUILDING (RCB)	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	

Table 5.5-1: Summary of Top SLERF Cutsets

#	Cutset Prob	Event Prob	Event	Description	Cutset Description
		[redacted]	[redacted]	[redacted]	[redacted]
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	

Table 5.5-1: Summary of Top SLERF Cutsets					
#	Cutset Prob	Event Prob	Event	Description	Cutset Description
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
20	4.63E-07	7.16E-06	%G09	Seismic Initiating Event (0.6g to <0.9g)	Bin %G09 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3) due to pounding-induced cracking, which leads

Table 5.5-1: Summary of Top SLERF Cutsets

#	Cutset Prob	Event Prob	Event	Description	Cutset Description
		9.98E-01	SF-TB_CRANE-C-G09	SEISMIC FRAGILITY FOR %G09: Turbine Building Gantry Crane	to a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. The TB Gantry Crane fails and interacts with and fails the CST failing the normal supply to the SDAFW; failure of TB3 prevents alignment of the Condenser Water Box to the SDAFW pump suction. Loss of all injection/recirculation and SSHR leads to core damage. Loss of power leads to loss of containment coolers and containment spray injection, which leads to LERF.
		1.00E+00	SF-TB-CLASS-3-POUND-C-G09	SEISMIC FRAGILITY FOR %G09: Turbine Building Class 3 - Pounding-induced cracking	
		9.76E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G09	DFOST Liquefaction-Induced Settlement	
		7.50E-01	TBCRANE-CST	TB Crane Failure Interacts with CST causing Failure	
		9.88E-02	XFL_PDS3P	Plant Damage State 3P	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
22	4.60E-07	7.16E-06	%G09	Seismic Initiating Event (0.6g to <0.9g)	Bin %G09 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3), which leads to a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. CST seismically fails, leading to failure of the normal supply to the SDAFW; failure of TB3 prevents alignment of the Condenser Water Box to the SDAFW pump suction. Loss of all injection/recirculation and SSHR leads to core damage. Loss of power leads to loss of containment coolers and containment spray injection, which leads to LERF.
		9.52E-01	SF-TB-CLASS-3-C-G09	SEISMIC FRAGILITY FOR %G09: Turbine Building Class 3	
		7.82E-01	SF-TK-COND-STRG-TNK-C-G09	SEISMIC FRAGILITY FOR %G09: Condensate Storage Tank	
		9.76E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G09	DFOST Liquefaction-Induced Settlement	
		9.88E-02	XFL_PDS3P	Plant Damage State 3P	

Table 5.5-1: Summary of Top SLERF Cutsets					
#	Cutset Prob	Event Prob	Event	Description	Cutset Description
		8.96E-01	X-POWEROP	Plant Capacity Factor	
23	4.60E-07	8.47E-06	%G08	Seismic Initiating Event (0.5g to <0.6g)	Bin %G08 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3) due to pounding-induced cracking, which leads to a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. The TB Gantry Crane fails and interacts with and fails the CST failing the normal supply to the SDAFW; failure of TB3 prevents alignment of the Condenser Water Box to the SDAFW pump suction. Loss of all injection/recirculation and SSHR leads to core damage. Loss of power leads to loss of containment coolers and containment spray injection, which leads to LERF.
		9.77E-01	SF-TB_CRANE-C-G08	SEISMIC FRAGILITY FOR %G08: Turbine Building Gantry Crane	
		9.91E-01	SF-TB-CLASS-3-POUND-C-G08	SEISMIC FRAGILITY FOR %G08: Turbine Building Class 3 - Pounding-induced cracking	
		8.45E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G08	DFOST Liquefaction-Induced Settlement	
		7.50E-01	TBCRANE-CST	TB Crane Failure Interacts with CST causing Failure	
		9.88E-02	XFL_PDS3P	Plant Damage State 3P	
		8.96E-01	X-POWEROP	Plant Capacity Factor	
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
		[redacted]	[redacted]	[redacted]	
25	4.59E-07	1.94E-05	%G06	Seismic Initiating Event (0.35g to <0.4g)	Bin %G06 seismic event causes a plant trip. There is a seismic failure of the Class 3 Turbine Building (TB3) due to pounding-induced cracking, which leads to a LOOP and failure of AFW C. Liquefaction-induced settlement failure of the DFOST fails EDGs; thus, loss of all AC power. TB3 interacts with and fails TB1, which fails the SDAFW pump. Loss of all injection/recirculation and SSHR leads to core damage. Loss of power leads to loss of containment coolers and containment spray injection, which leads to LERF.
		8.48E-01	SF-TB-CLASS-3-POUND-C-G06	SEISMIC FRAGILITY FOR %G06: Turbine Building Class 3 - Pounding-induced cracking	
		6.30E-01	SF-TK-DG-FOSTRG-TNK_SETTLE_G06	DFOST Liquefaction-Induced Settlement	

Table 5.5-1: Summary of Top SLERF Cutsets					
#	Cutset Prob	Event Prob	Event	Description	Cutset Description
		5.00E-01	TB-CLASS3-1	50% chance TB Class 3 interacts with TB Class 1	
		9.88E-02	XFL_PDS3P	Plant Damage State 3P	
		8.96E-01	X-POWEROP	Plant Capacity Factor	

Table 5.5-2: Top 10 SLERF Importance Measures Ranked by FV - Seismic Failures							
Component	**Description**	**FV**	**Median Capacity (g)**	**β_r**	**β_u**	**Failure Mode[2]**	**Fragility Method[2]**
SF-TB-CLASS-3-POUND	Turbine Building Class 3 - Pounding-induced cracking	29.59%	0.28	0.13	0.25	RAB pounding induced cracking and splitting of the mezzanine floor slab resulting in loss of structural integrity.	SOV
SF-RCB	REACTOR CONTAINMENT BUILDING (RCB)	11.85%	0.85	0.18	0.28	Nonlinear rotation of the socketed like connection at the underside of the basemat due to lateral displacement of the structure	SOV
SF-TK-DG-FOSTRG-TNK_SETTLE	DFOST Liquefaction-Induced Settlement	8.03%	N/A[1]	N/A	N/A	Failure of EDG-B pipe at RAB penetration. Probability of failure at PGA: PGA(g) - Probability 0.265g - 0.388 0.325g - 0.569 0.582g - 0.887 0.717g - 0.966	SOV
SF-TB_CRANE	Turbine Building Gantry Crane	7.67%	0.29	0.21	0.24	Failure of A-frame anchor bolts	SOV
SEISMIC_SPREAD_DC2	Liquefaction-Induced Lateral Spreading Distance Category 2	5.02%	N/A[1]	N/A	N/A	Liquefaction-Induced Lateral Spreading. Probability of failure at PGA: PGA(g) - Probability 0.265g - 0.000 0.325g - 0.004 0.582g - 0.043 0.717g - 0.061	Rep.

Table 5.5-2: Top 10 SLERF Importance Measures Ranked by FV - Seismic Failures							
Component	**Description**	**FV**	**Median Capacity (g)**	**β_r**	**β_u**	**Failure Mode[2]**	**Fragility Method[2]**
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	SOV
SF-TP-SDAFW-PMP_SETTLE	SDAFW Liquefaction-Induced Settlement	4.19%	N/A[1]	N/A	N/A	Failure of AFW discharge piping where the 4 in. diameter pipe meets the 6x4 reducing elbow. Probability of failure at PGA: PGA(g) - Probability 0.265g - 0.168 0.325g - 0.263 0.582g - 0.609 0.717g - 0.708	SOV
SF-RAB	Reactor Auxiliary Building	2.74%	1.12	0.24	0.26	Flexural failure of piles.	Rep.
SF-TR-UG-INTAKE_SETTLE	Underground Cable Trays at Intake Liquefaction-Induced Settlement	2.45%	N/A[1]	N/A	N/A	Liquefaction-induced settlement failure at Intake Structure. Probability of failure at PGA: PGA(g) - Probability 0.265g - 0.110 0.325g - 0.180 0.582g - 0.450 0.717g - 0.570	Rep.

Table 5.5-2: Top 10 SLERF Importance Measures Ranked by FV - Seismic Failures

Component	Description	FV	Median Capacity (g)	β_r	β_u	Failure Mode[2]	Fragility Method[2]
SF-WP-DPW-PMP-D_SETTLE	Deepwell Pump D Liquefaction-Induced Settlement	2.07%	N/A[1]	N/A	N/A	Bending failure of a bolted flange connection due to liquefaction-induced settlement. Probability of failure at PGA: PGA(g) - Probability 0.265g - 0.546 0.325g - 0.732 0.582g - 0.952 0.717g - 0.968	SOV

Note 1: Liquefaction-Induced Settlement failures do not utilize median capacity or β-values. The probability of failure at a given PGA is provided in the Failure Mode column.

Note 2: Fragility Mode and Fragility Method per [82].

The most significant non-seismic SSC SLERF contributors (e.g., random failures of modeled components during the SPRA mission time) are listed in Table 5.5-3.

Table 5.5-3: Top 10 SLERF Importance Measures Ranked by FV - Non-Seismic Failures

Component	Description and Failure Mode	FV
GINRDOORSL	PERSONNEL HATCH INNER DOOR GASKETS FAILS	1.09%
FPT1XSABFR	TURBINE-DRIVEN PUMP FAILS TO RUN	0.32%
GDOORSEALS	FAILURE OF PERSONNEL HATCH DOOR SEALS	0.32%
GELPENFO	ELECTRICAL PENETRATIONS FAILS OPEN	0.32%
OPER-61	Failure to align and start pre-staged pumps for SG makeup - Condenser Inlet Waterbox (FLEX)	0.30%
GOPER-PRE3	PRE-INITIATOR IMPORTANCE SCOPING EVENT FOR CI - P-44/45 BYPASS LEFT OPEN	0.29%
FPT1XSABFS	TURBINE-DRIVEN PUMP FAILS TO START	0.18%
OPER-64	Failure to align and start portable pumps to lake for long-term water source - SG makeup (FLEX)	0.16%
OPER-18B-CST	Failure to supply AFW with SW	0.15%
GOPER-PRE4	PRE-INITIATOR IMPORTANCE SCOPING EVENT FOR CI- PERSN HATCHES LEFT OPEN	0.14%

5.6 SPRA Quantification Uncertainty Analysis

A parametric uncertainty propagation for the SPRA base seismic CDF and LERF was performed.

Probability distribution types and associated variance parameters (e.g., error factors (EFs) in the case of lognormal distributions) are assigned to each of the basic events in the PRA. This is a basic step in the PRA development process and much of this distribution information already exists in the PRA database used for the SPRA as the SPRA database is built upon the internal events PRA database. Distribution information had to be added for modeling elements for SPRA that do not already exist in the internal events-based PRA database; these include seismic hazard interval initiators, seismic fragility basic events and seismic-adjusted HEPs.

The Monte Carlo sampling process was selected for the parametric analysis, with 20,000 samples and a "/C" value of 37,000 cutsets for SCDF and 29,195 for SLERF. 100% Binary Decision Diagram (BDD) was not possible due to insufficient computer memory; therefore, there is some percent of over counting in the uncertainty calculations and that is reasonable in the context of parametric uncertainty analysis since the primary purpose of understanding the spread of the distribution in the final point estimate result. The results are provided in Table 5.6-1, and Figures 5.6-1 through 5.6-2, each of which shows the curves of cumulative probability and probability density function.

Table 5.6-1 Parametric Uncertainty Analysis Results

	SCDF /reactor-year			SLERF /reactor-year		
	5%	50%	95%	5%	50%	95%
MC	5.81E-06	5.50E-05	5.07E-04	7.71E-07	8.74E-06	9.91E-05

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. For the SPRA, the important model uncertainties are addressed through the sensitivity studies described in Section 5.7 to determine the potential impact on SCDF or SLERF.

In terms of completeness uncertainty, the SPRA scope and level of detail is evaluated through the SPRA Peer Review to support the technical adequacy needed for risk-informed decision making.

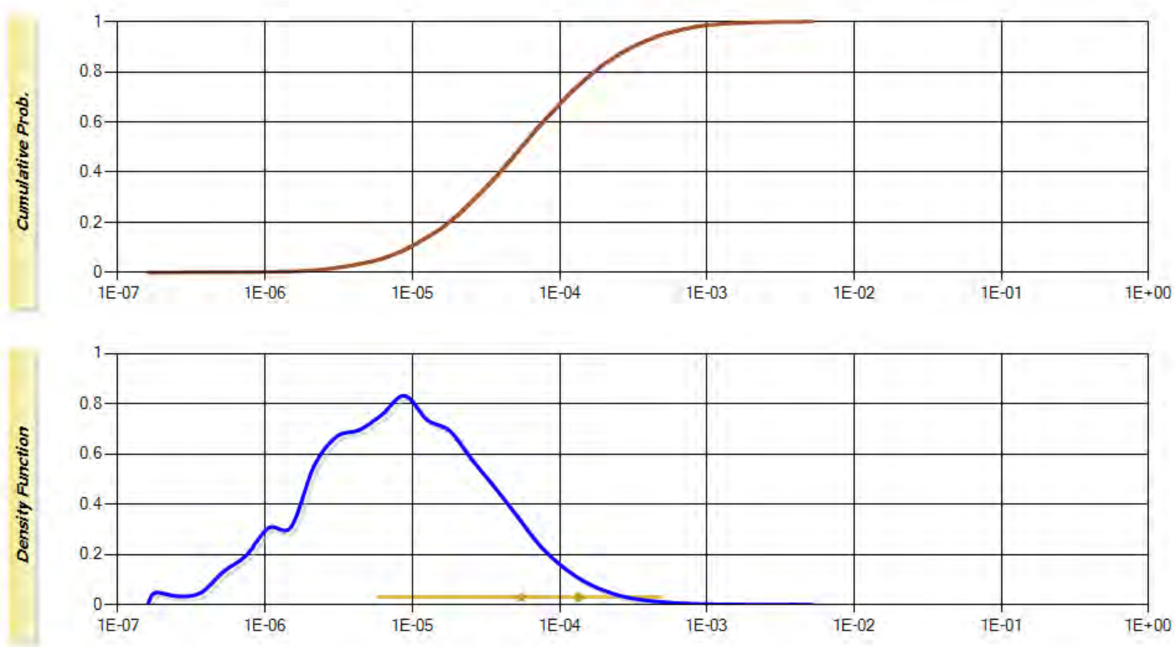

Figure 5.6-1 – RNP SPRA SCDF Parametric Uncertainty (Monte Carlo, 20K Samples)

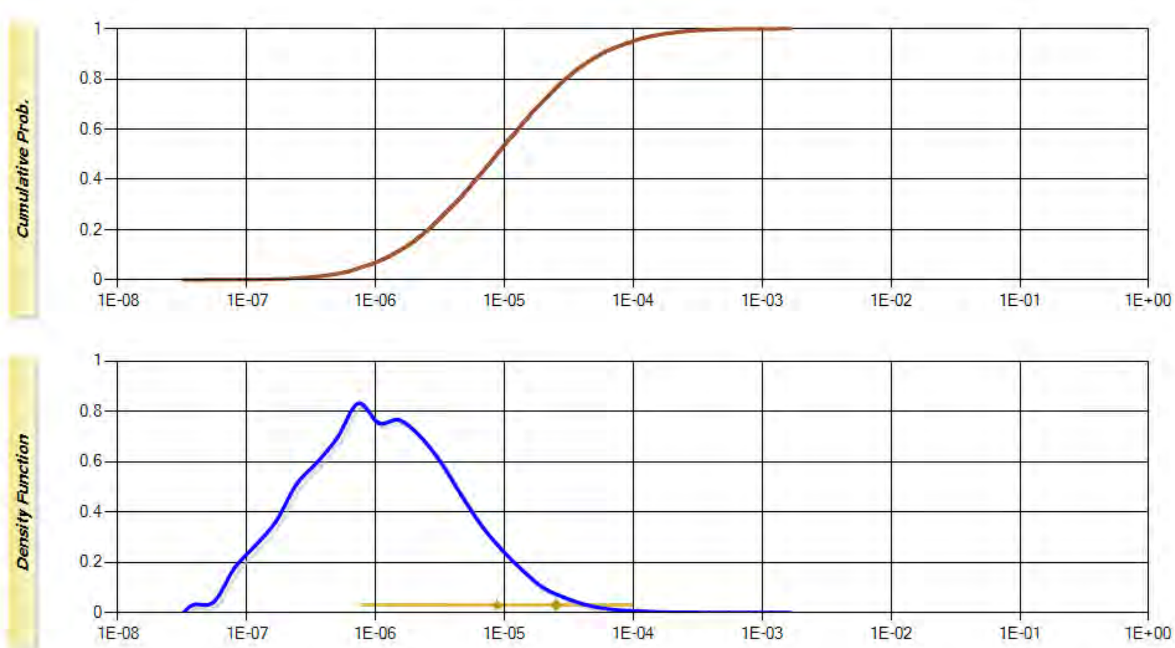

Figure 5.6-2 – RNP SPRA LERF Parametric Uncertainty (Monte Carlo, 20K Samples)

5.7 SPRA Quantification Sensitivity Analysis

Various sources of model uncertainties were reviewed and examined to identify sources that may have a significant impact on the SCDF and SLERF. A detailed description of each sensitivity is provided in the station calculation documenting the sensitivity notebook [70]. Table 5.7-1 shows a summary of the SPRA sensitivity analysis.

Table 5.7-1: Summary of RNP SPRA Sensitivity Cases

Sensitivity Case #	Description	CDF	Delta CDF	% Delta CDF	LERF	Delta LERF	% Delta LERF
Base Case	Base Case	9.31E-05[2]	--	--	2.02E-05	--	--
IE-1a	Use the Upper Bound percentile hazard curve (95th percentile)	3.58E-04	2.65E-04	285%	7.67E-05	5.65E-05	280%
IE-1b	Use the Lower Bound percentile hazard curve (5th percentile)	4.35E-06	-8.88E-05	-95%	8.14E-07	-1.94E-05	-96%
IE-1c	Use 12 hazard intervals	9.31E-05	-2.48E-08	-0.03%	2.00E-05	-1.57E-07	-0.8%
SY-1a	Half median capacity of OSP	9.48E-05	1.63E-06	1.7%	2.05E-05	2.58E-07	1.3%
SY-1b	Screen fragility groups with HCLFP ≥ 0.75g	9.30E-05	-8.93E-08	-0.1%	2.02E-05	-1.05E-08	-0.1%
SY-1c	Increase Demand of Liquefaction-Induced Settlement Failures	1.03E-04	9.91E-06	11%	2.19E-05	1.67E-06	8.3%
SY-1d	Increase Capacity of DFOST Piping to Resist Liquefaction-Induced Settlement Failure	8.98E-05	-3.32E-06	-3.6%	1.98E-05	-4.34E-07	-2.1%
SY-1e	Decrease Capacity of Deepwell Pumps A/B/C to Resist Liquefaction-Induced Settlement Failure	9.44E-05	1.29E-06	1.4%	2.03E-05	1.34E-07	0.7%
SY-1f	Increase Capacity of Deepwell Pump D to Resist Liquefaction-Induced Settlement Failure	9.31E-05	-2.23E-08	-0.02%	2.01E-05	-7.55E-08	-0.4%
SY-1g	Increase Capacity of SDAFW Pump to Resist Liquefaction-Induced Settlement Failure	8.97E-05	-3.39E-06	-3.6%	1.98E-05	-4.31E-07	-2.1%
SY-1h	Decrease Capacity of SW Piping to Resist Liquefaction-Induced Settlement Failure	9.37E-05	5.86E-07	0.6%	2.03E-05	1.24E-07	0.6%
SY-1i	Increase Capacity of TBCrane	8.86E-05	-4.54E-06	-4.9%	1.94E-05	-7.57E-07	-3.7%
SY-1j	Decrease Capacity of RCB	1.34E-04	4.12E-05	44%	6.14E-05	4.12E-05	204%

Table 5.7-1: Summary of RNP SPRA Sensitivity Cases

Sensitivity Case #	Description	CDF	Delta CDF	% Delta CDF	LERF	Delta LERF	% Delta LERF
SY-2a	Decrease TB3-TB1 interaction probability	8.62E-05	-6.89E-06	-7.4%	1.93E-05	-9.09E-07	-4.5%
SY-2b	Increase TB3-TB1 interaction probability	1.00E-04	6.88E-06	7.4%	2.11E-05	9.35E-07	4.6%
SY-2c	Decrease TBCrane-CST interaction probability	8.95E-05	-3.67E-06	-3.9%	1.97E-05	-5.16E-07	-2.6%
SY-2d	Set TBCrane-CST interaction probability to zero	8.20E-05	-1.11E-05	-12%	1.86E-05	-1.55E-06	-7.7%
SY-2e	Assume TB3 fails SDAFW	1.06E-04	1.31E-05	14%	2.21E-05	1.88E-06	9.3%
SY-2f	Assume Lateral spreading to DC2 does not lead directly to LERF	N/A[1]	N/A	N/A	1.92E-05	-1.01E-06	-5.0%
SY-3a	Assume no relay chatter scenarios	8.84E-05	-4.76E-06	-5.1%	1.96E-05	-5.78E-07	-2.9%
SY-5a	Uncorrelate seismic fragility groups	9.25E-05	-6.68E-07	-0.7%	1.66E-05	-3.56E-06	-18%
SY-5b	Correlate liquefaction-induced settlement failures	1.20E-04	2.71E-05	29%	2.42E-05	3.96E-06	20%
HR-2a	Credit Portable FLEX SG Makeup	9.16E-05	-1.51E-06	-1.6%	1.98E-05	-3.53E-07	-1.7%
HR-2b	All HEPs and JHEPS set to 95% percentile	9.50E-05	1.92E-06	2.1%	2.03E-05	9.97E-08	0.5%
HR-2c	All HEPs and JHEPS set to 5% percentile	9.19E-05	-1.25E-06	-1.3%	2.01E-05	-1.03E-07	-0.5%

Note 1: Seismic CDF is not quantified as this only affects the LERF model.

Note 2: This value was based on the combined cutsets while 9.27E-05 was per the summation of the hazard bin results.

5.8 SPRA Logic Model and Quantification Technical Adequacy

The Robinson Nuclear Power Plant SPRA risk quantification and results interpretation methodology were subjected to an independent peer review against the pertinent requirements in the ASME/ANS PRA Standard [37].

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

6.0 Conclusions

A seismic PRA has been performed for Robinson Nuclear Power Plant in accordance with the guidance in the PRA Standard [5 and 37] and the SPID [2]. The Robinson SPRA shows that the point estimate SCDF is 9.27×10^{-5}/reactor-year and the SLERF is 2.02×10^{-5}/reactor-year. The SPRA as described in this submittal reflects the as-built/as-operated Robinson Nuclear Power Plant as of the freeze date – June 2015 [77]. An assessment is included in Appendix A of the impact of the results of plant changes not included in the model since the model freeze date.

The insights from this study reveal the SCDF and SLERF are dominated by the seismic failure of the Turbine Building Class 3 caused by building pounding between the RAB and the mezzanine floor portion of the Turbine Building Class 3. To mitigate this unique seismic vulnerability, Robinson will implement a plant modification to provide additional protection from seismic hazards. The modification involves changes to the existing FLEX strategy to provide AFW flow to the SGs (See Table 6-1). The results of the corresponding sensitivity analysis show the SCDF and SLERF reductions to be approximately 40 percent and 30 percent, respectively. The following action(s) will be performed as a result of the SPRA.

Table 6-1 Planned Actions			
Action	**System Description**	**Action Description**	**Completion Date**
1	Auxiliary Feedwater provided by a modified FLEX strategy	This modified FLEX strategy involves SG makeup using intermediate pressure AFW pumps with available water sources at the site (e.g., Lake Robinson or alternate water sources, such as existing or new tanks).	By 12/31/2022

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8.0 Acronyms

AFE	Annual Frequency of Exceedance
AFW	Auxiliary Feedwater
ANS	American Nuclear Society
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram (also ATWT, Anticipated Transient Without Trip)
BDD	Binary Decision Diagram
BOP	Balance of Plant
CDFM	Conservative Deterministic Failure Model
CEUS	Central and Eastern United States
CMS	Conditional Mean Spectra
CSR	Cyclic Stress Ratio
CRR	Cyclic Resistance Ratio
DRS	Design Response Spectrum
ECCS	Emergency Core Cooling System
EPRI	Electric Power Research Institute
ESEP	Expedited Seismic Evaluation Program
FEM	Finite Element Model
FIRS	Foundation Input Response Spectra
FLEX	Diverse and Flexible Coping Strategies
FV	Fussel-Vesely
GIP	Generic Implementation Procedure
GMC	Ground Motion Characterization
GMI	Ground Motion Incoherence
GMRS	Ground Motion Response Spectra
HBRSEP	H.B. Robinson Steam Electric Plant, Unit No. 2
HEP	Human Error Probability
HFE	Human Failure Event
IPEEE	Individual Plant Examination for External Events
ISRS	In-Structure Response Spectra
HF	High Frequency
LF	Low Frequency
LMSM	Lumped Mass Stick Model
LOCA	Loss of Coolant Accident
MAFE	Mean Annual Frequency of Exceedance
MDAFW	Motor Driven Auxiliary Feedwater
NEI	Nuclear Energy Institute
NGVD	National Geodetic Vertical Datum
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NTTF	Near Term Task Force
PGA	Peak Ground Acceleration
PRT	Peer Review Team
PSHA	Probabilistic Seismic Hazard Analysis

RLME	Repeated Large Magnitude Earthquake
RNP	Robinson Nuclear Power Plant
RPS	Reactor Protection System
SBO	Station Blackout
SCDF	Seismic Core Damage Frequency
SCOR	Soil Column Outcrop Response
SDAFW	Steam Driven Auxiliary Feedwater
SEL	Seismic Equipment List
SEWS	Screening Evaluation Worksheets
SFP	Spent Fuel Pool
SFR	Seismic Fragility Element Within ASME/ANS PRA Standard
SG	Steam Generator
SHA	Seismic Hazard Analysis Element Within ASME/ANS PRA Standard
SHS	Seismic Hazard Submittal
SLERF	Seismic Large Early Release Frequency
SMA	Seismic Margin Assessment
SOV	Separation of Variables
SPID	Screening, Prioritization and Implementation Details
SPR	Seismic PRA Modeling Element Within ASME/ANS PRA Standard
SPRA	Seismic Probabilistic Risk Assessment
SQUG	Seismic Qualification Utility Group
SRSS	Square Root of the Sum of the Squares
SRT	Seismic Review Team
SSC	Structure, System or Component
SSEL	Safe Shutdown Equipment List
SSHAC	Senior Seismic Hazard Analysis Committee
SSI	Soil Structure Interaction
TSCR	Truncated Soil Column Response
UHRS	Uniform Hazard Response Spectra
UHS	Ultimate Heat Sink
USI	Unresolved Safety Issue

Appendix A

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

This Appendix has two purposes:

1. Provide a summary of the SPRA peer review
2. Provide the bases for why the SPRA is technically adequate for the 50.54(f) response.

The Robinson PRA was subjected to an independent peer review against the pertinent requirements of ASME / ANS RA-S Case 1 [37] (Code Case), which is an accepted alternate approach to Part 5 (Seismic) of Addenda B of the PRA Standard [5].

The information presented here establishes that the Robinson 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 Regulatory Guide 1.200 R2 [12] and the requirements in Section 1-6 of the ASME/ANS PRA Standard [5 and 37], and presents the significant results of the peer review.

A.1. Overview of Peer Review

The peer review assessment [7], and subsequent disposition of peer review findings, is summarized here. The scope of the review encompassed the set of technical elements and supporting requirements (SR) for the SHA (seismic hazard), SFR (seismic fragilities), and SPR (seismic PRA modeling) 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 Robinson SPRA peer review was conducted during the week of November 11, 2018 at the Duke offices in Charlotte North Carolina. As part of the peer review, a walk-down of portions of Robinson Nuclear Plant was performed on November 12, 2018 by selected members of the peer review team.

A Focused Scope Peer Review, FSPR, of LERF and Dam Failure Fragility was also performed remotely September 25-29, 2019. The scope of the review was the SRs related to the revised LERF hydrogen analysis as well as the refined Robinson Dam fragility.

A.2. Summary of the Peer Review Process

The peer review was performed against the requirements of ASME / ANS RA-S Case 1 [37], which is an accepted alternate approach to Part 5 (Seismic) of Addenda B of the PRA Standard [5]. The team utilized the peer review process defined in NEI 12-13 [6]. The review was conducted over a 30-day period, including 3 weeks of offsite review prior to a five day on-site portion of the review.

The SPRA peer review process defined in [6] involves an examination by each reviewer of their assigned PRA technical elements against the requirements in the Code Case to ensure the robustness of the model relative to all of the requirements.

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

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

As part of the review team's assessment of capability categories, F&Os are prepared. There are three types of F&Os defined in [6]: Findings, which identify issues that must be addressed in order for an SR (or multiple SRs) to meet Capability Category II; Suggestions, which identify issues that the reviewers have noted as potentially important but not requiring resolution to meet the SRs; and Best Practices, which reflect the reviewers' opinion that a particular aspect of the review exceeds normal industry practice. The focus in this Appendix is on Findings and their disposition relative to this submittal as well as Findings from the Focused Peer Review conducted in September 2019.

A.3. Peer Review Team Qualifications

The review was conducted by: Kenneth Kiper of Westinghouse, Mr. Jeffrey Kimball of Rizzo International, Inc.; Dr. Arash Zandieh of Lettis Consultants International, Inc.; Dr. Ram Srinivasan, independent consultant; Joe Vasquez of Dominion Energy Company; Benny Ratnagaran of Southern Nuclear Company; Dr. Andrea Maioli of Westinghouse; and Nathan Barber of Pacific Gas & Electric Company and Chris Peckat of American Electric Power participated as a working observer.

The team was assembled by the peer review team lead. The lead and reviewer qualifications are summarized here below and have been reviewed by Duke Energy and have been confirmed to be consistent with requirements in the ASME/ANS PRA Standard and the guidelines of NEI-12-13.

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

Mr. Jeff Kimball was the lead for the review of the Seismic Hazard Analysis (SHA) technical element. Mr. Kimball has over 38 years of experience in site characterization; ground motion modeling including site response and probabilistic seismic hazard analysis (PSHA). Mr. Kimball has served as SHA reviewer for a number of recent SPRAs and serves on the Participatory Peer Review Panel for the NGA-East Project. Mr. Kimball was assisted by Dr. Arash Zandieh. Dr. Zandieh has 8 years of experience in seismic hazard analysis, earthquake engineering, engineering seismology, geotechnical and structural engineering, and statistical analysis. Dr. Zandieh has participated in a number of SPRA peer reviews.

Dr. Ram Srinivasan led the Seismic Fragility Analysis (SFR) review. Dr. Srinivasan has over 45 years of experience in the nuclear industry, principally in the design, analysis (static and dynamic, including seismic), and construction of nuclear power plant structures. He is actively involved in the Post-Fukushima Seismic Assessments (NRC NTTF 2.1 and 2.3) and is a member of the NEI Seismic Task Force and the ASME/ANS JCNRM, Part 5 Working Group (Seismic and other External Hazards PRA). He has participated on several previous SPRA peer reviews, either as reviewer or utility consultant. Dr. Srinivasan was assisted by Mr. Joe Vasquez and Mr. Benny Ratnagaran. Mr. Vasquez has 17 years of nuclear engineering experience covering all areas within the Engineering Mechanics including pipe stress analysis, pipe and equipment support analysis, pressure vessel design and analysis, seismic qualification of mechanical and electrical equipment, seismic margins assessment and fragility analyses, and fracture mechanics. He has participated on several previous SPRA peer reviews, either as reviewer or utility defender. Mr. Ratnagaran has 7 years of experience in developing seismic PRAs, seismic response analysis, and structural fragility analyses. He has also participated on several previous SPRA peer reviews, either as reviewer or utility defender.

Dr. Andrea Maioli was the lead for the review of the Seismic System Response Analysis (SPR) technical element. Dr. Maioli has over 15 years of experience in the nuclear safety area generally and seismic PRA specifically. He has served as lead engineer for a number of seismic PRA and seismic margin studies for existing and new nuclear power plants. He has participated in and led a number of SPRA peer reviews. Dr. Maioli was assisted by Mr. Nathan Barber. Mr. Barber is a nuclear engineer and mechanical engineer with over 16 years' experience working in the nuclear power industry. He is the technical lead for the Seismic PRA update at Diablo Canyon. He has participated in a number of peer reviews, including internal events PRAs and SPRAs. Mr. Chris Peckat from American Electric Power served as working observer for the SPR technical element. Any observations and findings that Mr. Peckat generated were given to the peer review team for their review and ownership. As such Mr. Peckat assisted with the review but was not a formal member of the peer review team.

This peer review report was compiled by the peer review team lead. A draft copy of the peer review report was sent to Duke Energy as well as the other peer review members on Dec 10, 2018.

The Focused Scope Peer Reviewer's qualifications are given below:

Mr. LaBarge is a Principal Engineer in the Risk Applications and Methods group Westinghouse and has approximately 14 years of experience with PRA models in the nuclear industry. Mr. LaBarge has experience developing Level 1 and Level 2 PRA models and methods for a variety of applications. Mr. LaBarge has experience working closely with utilities in order to create PRA models that are consistent with the as-built as-operated plant. Mr. LaBarge is one of the Westinghouse experts in the area of Level 1 and Level 2 PRA model development, MAAP analysis, severe accident analysis and SAMG development. He is also currently the program committee chair for the ANS NISD vice chair of the JCNRM Level 2 PRA Standard Writing Group and a member of the JCNRM Subcommittee on Standards Development. Mr. LaBarge has experience participating in peer reviews representing both the utilities and as a peer reviewer.

Dr. Glenn Rix is a Senior Principal in Kennesaw Georgia with expertise in seismic hazard evaluation, geotechnical earthquake engineering, and performance based and risk based analysis. He had a distinguished career as a faculty member of the school of Civil and Environmental Engineering at Georgia Institute of Technology for 24 years prior to his consulting career. Dr. Rix has participated as a member of several peer reviews as well as a defender on numerous peer reviews.

A.4. Summary of the Peer Review Conclusions

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

2018 Full Scope Peer Review

Seismic Hazard (SHA)

The seismic hazard at RNP was evaluated using a site-specific probabilistic seismic hazard analysis (PSHA). The SPRA Standard requires the inputs to the site-specific PSHA to be based on current geological, seismological, and geophysical data; local site topography; and surficial geologic and geotechnical site properties. The RNP PSHA fully met this requirement. The RNP seismic hazard analysis and PSHA used the seismic source characterization (SSC) and ground motion model (GMM) based on SSHAC studies that have compiled comprehensive earth science datasets. The SSC model was developed as a SSHAC Study Level 3 (CEUS-SSC model; EPRI, 2012) and the GMM was developed as a SSHAC Study Level 2 update of a previous SSHAC Study Level 3 (EPRI, 2013). Both studies involved teams of experts and participatory peer review panels. The Technical Integration teams who completed these models considered the full range of earthquake data (geological, seismological, and geophysical) to develop the SSC model and GMM. The RNP PSHA included an update to the CEUS-SSC earthquake catalog and an assessment of recent seismicity and literature to determine in any changes or updates to the SSC model, seismicity rates, and GMM were needed. It was concluded that no updates to the SSC and GMM were needed.

The SPRA Standard requires the effects of local site response along with the uncertainties in characterizing the local site response analysis to be identified and included. The PSHA includes site characterization efforts to gather new geologic and geotechnical data to aid in the assessment of site response and liquefaction. Site-specific site response analyses were performed to include the effects of local site response. Uncertainties in site response inputs were included in the analyses. Therefore, RNP PSHA met the requirement to include effects of local site response. However, insufficient basis for establishing the base case site profiles for the Plant Area, representing the epistemic uncertainty, was provided. Therefore, there was a finding that should be addressed to fully satisfy the intent of the Standard.

The Standard requires that a screening analysis be performed to assess whether in addition to the vibratory ground motion other seismic hazards need to be included in the SPRA. For RNP, screening analysis for secondary seismic hazards were performed and documented, including: fault displacement, landslide, soil liquefaction, liquefaction-induced settlement, soil settlement related to non-liquefying seismic events, potential for earthquake-induced flooding, fracking induced earthquakes, seismic seiches and Tsunami. Two secondary seismic hazards were screened in: soil liquefaction and liquefaction-induced settlement. For those hazards, probabilistic distributions for liquefaction-induced settlement and lateral spreading displacements at locations of important structures, systems and components for different hazard levels were evaluated.

The SPRA Standard requires documentation of the hazard evaluation to be consistent with the applicable supporting requirements. The documentation of the RNP PSHA is a collection of reports that describes the geotechnical studies, site profile development, PSHA methodology, rock hazard results, site response analyses and results, soil hazard results, and the assessment of secondary seismic hazards including liquefaction. The SPRA standard requires the hazard evaluation to be documented in a manner that facilitates PRA applications, upgrades, and peer review. Moreover, the process used to perform the hazard evaluation and the evaluation results is also required to be documented. Overall, the documentation for the PSHA is complete and meets the intent of the Standard. However, there are a few elements that require additional discussions and justifications. Therefore, there were several findings in the documentation that should be addressed to fully satisfy the intent of the Standard.

Seismic Fragility (SFR)

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

The seismic response analyses of the RNP Structures that feed into the fragility evaluations are based on input response spectra corresponding to the GMRS generated and reported in SHA documents. It is seen that over 50% plant seismic CDF risk contribution would occur at earthquake input levels at or below the GMRS. Thus, the selection of the GMRS as the Reference Earthquake for the RNP SPRA is appropriate. Five sets of time histories of ground motion corresponding to the GMRS were generated. Each set included two horizontal components and one vertical component.

RNP developed new finite element models (FEM) for seismic response analysis. In a few cases, existing lumped mass spring models (LMSM) were enhanced to conform to the current practice. The peer review team concurs that the structural models are generally realistic.

RNP performed median-centered response analysis for various structures and considered the appropriate variabilities. Soil-structure interaction (SSI) was considered for the Reactor Containment Building (RCB), and Reactor Auxiliary Building (RAB). SSI effects were deemed to be not significant for the lighter Turbine Building (TB). The SSI analysis included the pile-soil spring elements.

RNP performed probabilistic response analysis in developing the fragility of the Turbine Building (Class III). Thirty simulations were used following Latin Hypercube sampling to ensure stability of the analysis. The analyses were performed corresponding to three earthquake levels (GMRS, 70% GMRS, and 85% GMRS).

Seismic walkdowns performed for the Robinson SPRA were generally found to be comprehensive and complete. Walkdown documentation is voluminous and meets expectation. While a few issues were identified based on peer review team (PRT) walkdown where seismic interactions may not have been noted on walkdown forms and/or bases for seismic review team (SRT) walkdown judgments were not clearly evident, the preponderance of evidence suggests thorough walkdowns were performed and documented by experienced personnel. Per response from RNP, the SPRA model has been reviewed for plant changes up to October 2017. Many ex-control room operator actions were characterized as having multiple pathways therefore investigations were focused only on the equipment that needs to be manipulated and the immediately adjacent areas. Seismic induced fire and flood sources were assessed and documented. With a few exceptions, walkdowns performed adequately identified credible seismic interactions and consequences of potential interactions identified were adequately addressed within fragility documentation developed.

Fragilities were calculated for all the relevant failure modes identified for SSCs (in SFR-E2) that significantly contribute to the seismic CDF or seismic LERF. In addition to the typical functional, structural, and anchorage fragility modes, soil liquefaction and building interactions were identified to be significant for RNP SSCs. As noted in the SHA review, detailed probabilistic assessment of the soil liquefaction effects was performed for affected SSCs. Detailed analysis of the TB and RAB interaction was performed though the peer review team determined to be conservative. RNP fragility evaluation notebook describes the methods used to calculate the seismic fragilities of SSCs that are in the PRA model. The Separation of Variables (SoV) method was used for most of the top risk contributors. For non-risk contributors, representative fragilities using EPRI Hybrid approach were calculated. For a few SSCs, that do not contribute significantly to the SCDF or SLERF, judgement based lower bound fragilities were used. Considering that the fragility of the top risk contributor (TB pounding the RAB) to CDF and LERF was conservatively calculated and representative fragilities were used for a few risk significant components, the peer review team assessed the fragility of top risk contributors to be not realistic.

The peer review team was able to perform the peer review using the documentation received from the project team. Only a few minor items would require correction.

In summary, the fragility analysis generally meets the applicable requirements of the ASME/ANS RA-S Standard CODE CASE 1. However, the peer review team believes that further refinement of the fragility of the top contributor is likely to decrease the SPRA CDF and LERF estimates.

Peer Review Team Interpretation of Supporting Requirement SFR-E3

During a previous SPRA peer review, the peer review team identified an issue with the wording of SR SFR-E3 and concluded that, as published, the wording for CC-I does not match with the intent of the SR due to two typos. Following a dedicated discussion with the authors of the SFR section in the code and with the JCNRM leadership, the typos were confirmed and the team performed the review against a modified version of the SFR-E3 which, for CC-I, reads as follows:

ESTIMATE seismic fragilities for the failure modes of interest identified in SFR-E2 using plant-specific data and ENSURE that they are realistic ***conservative****. JUSTIFY (e.g., through the calculation of seismic CDF and LERF per HLR-SPR-E) the use of generic fragility data (e.g., fragility test data, generic seismic qualification test data, and earthquake experience data) or conservative assumptions for the SSCs as being appropriate for the plant and not significant to the overall results.*

This modified wording has been formally approved by the JCNRM. Note that the wording and understanding of SFR-E3 CC-II remain unchanged.

Seismic Plant Response (SPR)

The Robinson seismic plant response (SPR) model integrates the site-specific hazard, fragilities and system-analysis and accident sequence aspects. The starting point for the analysis was the existing internal events PRA model. Limited modifications were made to the underlying model, including the addition of FLEX strategies. These modifications were added to the model in a fashion consistent with the requirements in Part 2 of the Standard.

The RNP SPRA used standard EPRI tools (i.e., CAFTA, FRANX, ACUBE) to incorporate the seismic induced failures within the internal events PRA logic. These tools retained all the underlying random failures and operator actions and then were used to quantify the seismic induced CDF and LERF.

A detailed Seismic Equipment List was generated and the associated fragilities were included in the model consistently with the observations from the walkdowns and correlation considerations made in the fragility analysis. Because of the unique nature of the seismic hazard at the RNP site, the RNP SPRA includes two types of fragilities:

(a) Lognormal fragilities were used to model functional, anchorage and special interaction failures. These fragilities were managed through the FRANX code mapping to the underlying internal events logic. Full correlation was assumed for the modeling of the lognormal fragilities.

(b) Non-lognormal fragilities were used to model soil failure and lateral spreading. These fragilities were manually added to the internal events logic. For these non-lognormal fragilities, capacities were assumed to be uncorrelated and the fragility groupings were based on the lateral spreading profiles.

The RNP SPRA explicitly models a reasonably complete set of seismic induced initiators, derived from the internal events model. However, the peer review team noted the absence of documentation of a systematic process to derive internal events and external events challenges to the plant and to identify similar seismic-induced versions of such challenges. Since documentation is critical in supporting and reproducing any screening process, the initial technical supporting requirements SPR-A1 and A2, which are being considered not met and related findings were written regarding the need for a documented systematic assessment. Absence of a review of operating experience at the RNP plant associated with seismic events also resulted in SPR-A3 being judged not met.

Seismic performance shaping factors were systematically considered in adapting the human reliability analysis performed for internal events to seismic-induced sequences. The RNP SPRA team went beyond the recently published EPRI method in the level of details applied to the analysis, using a more refined breakdown of each action. The review team noted that operator actions explicitly developed for the SPRA (e.g., operator action associated with FLEX) were not modeled with the same approach and should have similar considerations for different seismic hazard levels.

The RNP SPRA is quantified in a manner that allows an adequate estimation of the risk profile and identification of lead risk contributors in terms of accident sequences, individual components, fragility groups and operator actions. The quantification process is challenged by well-known and understood limitations in the tools used (e.g., truncation and stability challenges due to challenges to the rare event approximation assumption). While such challenges may be slightly overestimating the seismic risk, they are not expected to significantly change the risk insights that can be drawn by the SPRA.

Some limitations were observed in the characterization and documentation of uncertainties associated with key assumptions in the overall SPRA; associated recommendations were given on this topic and supporting requirement SPR-E7 was judged not met.

Finally, because the underlying internal events LERF model meets only capability category I, the seismic LERF model is also judged to be CC-I in SPR-E6.

2019 Focused Scope Peer Review

Robinson LERF PRA Model

The IE-PRA focused peer review included nine SRs from Part 2 of the PRA Standard [5] and 2 SRs from ASME/ANS RA-S CASE [37]. The peer review assessment results show that all applicable SRs were judged to be Met at CC-II or above.

[redacted]

[redacted]

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

Table A-1 presents a summary of the SRs graded as not met or not Capability Category II, and the disposition for each. Table A-2 presents summary of the Finding F&Os that have not been closed through an NRC accepted process, and the disposition for each (included at the end of this Appendix due to size).

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

SR	Assessed Capability Category	Associated Finding F&Os	Disposition to Achieve Met or Capability Category II
SHA			
N/A	N/A	N/A	N/A
SFR			

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

SR	Assessed Capability Category	Associated Finding F&Os	Disposition to Achieve Met or Capability Category II
C-SFR-E3	CCI Not Met for the FSPR	28-2 28-3 29-2 30-2 2-1 (FSPR)	Findings 28-2 and 29-2 were written concerning the use of generic conservative fragilities for SLOCA and SSLOCA and the abbreviated resolution follows: Following the original walkdowns of the items on the seismic equipment list (SEL), Duke conducted supplemental walkdowns of piping and tubing whose failure could lead to SSLOCA and SLOCA and concluded that all reviewed SSLOCA/SLOCA piping/tubing items would have High seismic capacity. Based on this, the plant specific SSLOCA and SLOCA fragilities were generated for use in the Robinson SPRA. Finding 28-3 was written concerning a possible improvement in the Turbine Building Class 3 pounding fragility via a refined estimate of building pounding forces at the impact interface; the summarized resolution: The energy dissipation at the impact point does not offer significant protection to the vulnerable diaphragm. While model refinement can increase the precision of the fragility, the resulting slightly modified fragility will not change the risk conclusions. Finding 30-2 was written concerning the issue of a possible correlation between the Turbine Building Class 3 pounding and shaking fragilities; a summary of the resolution: While realistically these two failure modes should be at least partially correlated, any partial correlation curves are bounded by these two cases; and therefore, partial correlation possibilities are not an important consideration. [redacted] In summary, all the Findings identified have been appropriately resolved and meet Capability Category II of the Standard.

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

SR	Assessed Capability Category	Associated Finding F&Os	Disposition to Achieve Met or Capability Category II
SPR			
C-SPR-A1	Not Met	25-4	Finding 25-4 documents the lack of a systematic disposition of internal initiating events. The disposition has been completed and the documentation updated such that this SR is met. The resolution to this Finding meets Capability Category II of the Standard.
C-SPR-A2	Not Met	24-4 25-2 25-3 25-10	Finding 24-4 documents a lack of a complete identification of seismically induced consequential events. The review has subsequently been completed, documented and no additional hazards identified. Finding 25-2, 25-3, and 25-10 are addressing potential issues with seismically induced fires. All 3 potential issues have been addressed. The resolutions to these Findings meet Capability Category II of the Standard.
C-SPR-A3	Not Met	25-1	Finding 25-1 identifies the lack of documentation for any site specific events or review of industry events that could be applicable to the SPRA. A review has been documented and no changes were required. The resolution to this Finding meets Capability Category II of the Standard.
C-SPR-E7	Not Met	24-2 24-8 24-20	Finding 24-2 was written to identify the lack of re-binning hazard intervals for optimization. Re-binning was performed and documented. Finding 24-8 addresses uncertainty surrounding the Turbine Building failing the SDAFW pump. Documentation of this uncertainty has been updated. Finding 24-20 was written to identify the lack of a systematic review of key assumptions. This review has been complete and is documented. The resolutions to these Findings meet Capability Category II of the Standard.

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

SR	Assessed Capability Category	Associated Finding F&Os	Disposition to Achieve Met or Capability Category II
C-SPR-E6	CCI	24-7 24-21 24- 22 24-23	Finding 24-21 relates to a lack of LERF timing change consideration. The binning of CDF to LERF sequences were re-visited and no LERF model changes were required. Findings 24-7 and 24-22 are concerned with documentation updates only. Based on this, the resolutions of these Findings meet Capability Category II of the Standard. Finding 24-23 relates to the use of the underlying internal events LERF model Met at CC-I. Although only Met at CC-I, the LERF methodology used by the site is an acceptable means of calculating LERF for use in risk-informed applications, including changes to its licensing basis as well as its response to NTTF 2.1 Thus, this Finding is resolved for the purposes of NTTF 2.1 seismic.

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

The set of supporting requirements from the ASME/ANS PRA Standard [5] 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 respond to implement the 50.54(f) letter. The conclusions of the peer review discussed above and summarized in this submittal demonstrates that the RNP SPRA model meets the expectations for PRA scope and technical adequacy as presented in RG 1.200, Revision 2 [12] as clarified in the SPID [2].

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

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

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

The RNP SPRA reflects the as-built and as-operated plant as of the freeze date for the SPRA, June 2015. There are no permanent plant changes that have not been reflected in the SPRA model, except for those discussed further in Section A.9.

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

The PWR Owners Group performed peer reviews of the RNP internal events PRA and internal flooding PRA that form the basis for the SPRA to determine compliance with ANS/ASME PRA Standard RA-Sa-2009 [5] along with the NRC clarifications provided in Regulatory Guide 1.200, Revision 2 [12]. The full scope internal events peer review was performed in October 2009 and the focused scope internal flooding peer review was performed in June 2015. These reviews documented findings for all supporting requirements (SRs) which failed to meet at least Capability Category II. All the internal events and internal flooding PRA peer review findings that may affect the SPRA model have been addressed.

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

- Table A-1 provides a summary of the disposition of SRs judged by the peer review to be not met, or not meeting Capability Category II.
- Table A-2 provides a summary of the disposition of the open SPRA peer review findings (included at the end of this Appendix due to size).
- Table A-3 provides an assessment of the expected impact on the results of the RNP SPRA of those SRs and peer review Findings that have not been fully addressed.

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

SR #	F&O #	Summary of Issue Not Fully Resolved	Impact on SPRA Results
[redacted]	[redacted]	[redacted]	[redacted]
SPR-A1	25-4	Systematic disposition of internal events	Documentation update only and no impact on the model (See Table A-2 for more details).
SPR-A2	24-4 25-2 25-3 25-10	Finding 24-4 documents a lack of a complete identification of seismically induced consequential events. The review has subsequently been completed, documented and no additional hazards identified. Finding 25-2, 25-3, and 25-10 are addressing potential issues with seismically induced fires. All 3 potential issues have been addressed this SR is now met.	Documentation update only and no impact on the model (See Table A-2 for more details).
SPR-A3	25-1	Finding 25-1 identifies the lack of documentation for any site specific events or review of industry events that could be applicable to the SPRA. A review has been documented and no changes were required; this SR is now met.	Documentation update only and no impact on the model (See Table A-2 for more details).
SPR-E7	24-2 24-8 24-20	Finding 24-2 was written to identify the lack of re-binning hazard intervals for optimization. Re-binning was performed and documented. Finding 24-8 addresses uncertainty surrounding the Turbine Building failing the SDAFW pump. Documentation of this uncertainty has been updated.	Per FO 24-2, the model was revised to place more bins at the lower accelerations and to condense the higher accelerations into fewer bins. Documentation update only for FOs 24-8 and 24-20 and no impact on the model.

Table A-3 Summary of Impact of Not Met SRs and Open Peer Review Findings			
SR #	F&O #	Summary of Issue Not Fully Resolved	Impact on SPRA Results
		Finding 24-20 was written to identify the lack of a systematic review of key assumptions. This review has been complete and is documented. This SR is met.	

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

The PRA Standard 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 [14] and EPRI 1016737 [15] 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 RNP SPRA model quantification (see Section 5 of this submittal).
- Modeling uncertainties are considered in both the base internal events PRA and the SPRA. Assumptions are made during the PRA development as a way to address a particular modeling uncertainty because there is not a single definitive approach. Plant-specific assumptions made for each of the Robinson Nuclear Power Plant SPRA technical elements are noted in the SPRA documentation that was subject to peer review, and a summary of important modeling assumptions is included in Section 5.
- Completeness uncertainty addresses scope and level of detail. Uncertainties associated with scope and level of detail are documented in the PRA but are only considered for their impact on a specific application. No specific issues of PRA completeness were identified in the SPRA peer review.

A summary of potentially important sources of uncertainty in the Robinson Nuclear Power Plant SPRA is listed in Table A-4.

Table A-4 Summary of Potentially Important Sources of Uncertainty		
PRA Element	**Summary of Treatment of Sources of Uncertainty per Peer Review**	**Potential Impact on SPRA Results**
Seismic Hazard	Site-specific site response analyses were performed to include the effects of local site response. Uncertainties in site response inputs were included in the analyses. Therefore, RNP PSHA met the requirement to include effects of local site response. The focused peer review team commented that a more thorough discussion of uncertainty in estimating fragility of the Robinson dam should be discussed.	The seismic hazard reasonably reflects sources of uncertainty. Regarding the peer review team’s comment on the Robinson Dam fragility, Duke provided the explanation that was acceptable to the peer review team during the peer review week and the dam fragility calculation was updated to include a more through discussion of uncertainty. Thus, no changes were made to the model.

Table A-4 Summary of Potentially Important Sources of Uncertainty		
PRA Element	**Summary of Treatment of Sources of Uncertainty per Peer Review**	**Potential Impact on SPRA Results**
Seismic Fragilities	No specific peer review team comments on sources of uncertainty in fragilities.	Many sensitivity studies described in Section 5.7 of this report evaluate the impact of changes to fragilities on the SPRA results as one means of assessing the impact of fragilities uncertainties on the SPRA results. No changes to the model were recommended based on these results.
Seismic PRA Model	The plant model assigns a 50 % probability of SDAFW pump failure due to failure of the TB Class III, assuming that the TB Class III collapses toward the TB Class I causing damage to the SDAFW pump. The review team commented that one important assumption (i.e., SDAFW pump surviving the failure of the Class III TB failure) is not fully addressed for the associated uncertainty.	A follow-up walkdown was performed to determine if the SDAFW Pump can be damaged if the TB Class III collapses away from the TB Class I. The walkdown results showed that there was a sufficient spatial separation from the closest point of the TB Class III to the SDAFW Pump skid and the turbine speed governor, steam piping and pump discharge line are well shielded from the impact of the TB Class III failure. In addition, there were no soft targets whose failure could prevent pump function. Based on this, the use of 50 % probability of SDAFW pump failure due to failure of the TB Class III is justified. Thus, no changes were made to the model.

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

The RNP SPRA reflects the plant as of the cutoff date for the SPRA, which was June 2015. Table A-5 lists significant plant changes subsequent to this date and provides a qualitative assessment of the likely impact of those changes on the SPRA results and insights.

Table A-5 Summary of Significant Plant Changes Since SPRA Cutoff Date

Description of Plant Change	Impact on SPRA Results
Transmission Upgrade Project installed new 115/230 kV SUTs, 4 kV Switchgear and associated equipment for Bldg. 469 etc.	As all of these are the changes to the existing switchyard that provides offsite power from the grid, the industry generic seismic fragility associated with a Loss of Offsite Power (LOOP) that has been used in the Robinson SPRA still applies to them. Thus, no beneficial impact on the Robinson SPRA.

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
SR C-SPR-D3	19-1	Section 6.3 of the station calculation for the Robinson Seismic Probabilistic Risk Assessment Human Reliability Analysis Notebook [65] addresses the addition of operator actions related to use of FLEX equipment. FLEX actions were evaluated with detailed HRA approach, except that a bin context adjustment was not used. Instead, FLEX actions were considered to apply across all seismic bins. The rationale for not using seismic bins is that the SSCs that drive the bin adjustments become irrelevant for FLEX. However, this fails to account for the increasingly challenging plant context created by failures at higher	SPR-D3 requires that the HRA account for relevant seismic-related effects on operator actions. The timing provided in the FLEX validation may be appropriate for low seismic hazards but should be adjusted to account for challenges from increasing seismic hazard levels.	For FLEX actions with a short time window (e.g., less than 2 hrs), consider quantifying separate actions based on different seismic hazard levels.	It was initially assumed that the FLEX actions were independent of the SSCs that drive the bin adjustments. However, it was judged that this approach may be non-conservative for FLEX actions at higher bin levels in which an extended time window (i.e., greater than 2 hours) was not available. Therefore, FLEX actions were reviewed to determine which actions may require additional bin-specific modeling. First, a review of the time window for the FLEX actions was performed. Those FLEX actions with a Tsw ≥ 2 hours were excluded from further consideration, specifically OPER-64, -67, -68 and -69. Only OPER-61 meets the criteria with a time window of less than 2 hours since it has a Tsw of 61 minutes. To address the finding, seismic HRA bin-specific variations of OPER-61 were developed, based on the bin adjustments detailed in Table 8.4 of the station calculation for the Robinson Seismic Probabilistic Risk Assessment Human Reliability Analysis Notebook [65]

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
		seismic accelerations. It may be possible to argue that actions such as OPER-69 with long time windows are independent of seismic hazard level. However, for actions such as OPER-61 (SG makeup) with a relatively short time window, variability of operator reliability by hazard level should be considered.			The HRA Calculator now contains detailed evaluations for OPER-61-S1, S2, S3 and S5. The resulting HEPs are included in the Table 10.1 of the station calculation for the Robinson Seismic Probabilistic Risk Assessment Human Reliability Analysis Notebook [65]. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard with NRC Clarification. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SPR-F1	19-2	The station calculation for the Robinson Seismic Probabilistic Risk Assessment Human Reliability Analysis Notebook [65] and the station calculation for the	1.Table 8.4 in the station calculation for the Robinson Seismic Probabilistic Risk Assessment Human Reliability Analysis Notebook [65] documents the PSF Adjustments for Seismic HFEs for the six HRA bins. However, Bin 4 is defined for only	Address the documentation issues.	1. To address the finding, Bins 4 and 6 have been retained in Table 8.4 for completeness but have been shaded to reflect that they are screened from the analysis. HFEs for these bin values have been deleted from the HRA Calculator and the Table 10.1 of the station

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
		Robinson Seismic Probabilistic Risk Assessment Quantification Notebook [13] include some specific issues that do not facilitate peer reviews and future uses and upgrades.	one acceleration value (0.75g) and this bin is apparently combined with Bin 5 (defined for 0.75g or higher). In addition, Bin 6 is defined as 0.40g or higher, with overlaps with Bins 1 to 5. The Note 1 explains that Bin 6 is screened due to the building failures at this same hazard level. It appears that Bin 4 and Bin 6 should be removed from the HRA bin definitions. 2. Action OPER-99 is in the station calculation for the Robinson Seismic Probabilistic Risk Assessment Quantification Notebook [13] but not the station calculation for the Seismic Equipment List [64] or the station calculation for the Robinson Seismic Probabilistic Risk Assessment Human Reliability Analysis Notebook [65] Based on discussions with the Duke team, the PR team understands that OPER-99 was a placeholder human failure event that is not needed and should be deleted from the Quantification Notebook [13]. 3. Based on discussions with the Duke team, the PR team		calculation for the Robinson Seismic Probabilistic Risk Assessment Human Reliability Analysis Notebook [65] 2. The HFE OPER-99 has been deleted from the station calculation for the Robinson Seismic Probabilistic Risk Assessment Quantification Notebook [13]. 3. The station calculation for the Containment Analysis [76] was reviewed to identify the following HFEs that are embedded in the Level 2 quantification: OPER-ILI, OPER-ISOL, OPER-IV, and OP-H2REC. OPER-ISOL is already set to 1.0 and was screened from seismic evaluation. The other events had Tsw's less than 4 hours, so seismic HRA bin values were calculated for them. The HRA Calculator now contains quantifications of -S1, S2, S3 and S5 values for OPER-ILI, OPER-IV and OP-H2REC. The resulting HEPs are included in the station calculation for the Robinson Seismic Probabilistic Risk Assessment Human Reliability Analysis Notebook [65] Table 10.1.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			understands that the level 2 model includes some embedded operator actions that are not included in the station calculation for the Robinson Seismic Probabilistic Risk Assessment Human Reliability Analysis Notebook [65]. These are longer term actions compared to the seismic event and, thus, the impact from the seismic event is not expected to be significant. However, this screening analysis has not been documented.		Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SPR-D3	19-3	Several operator actions that were reviewed against the PSF Adjustments for Seismic HFEs defined in Tables 8.3 and 8.4 of the station calculation for the Robinson Seismic Probabilistic Risk Assessment Human Reliability Analysis Notebook [65]. Some inconsistent were identified with the application of these rules.	The rules for PSF Adjustments for Seismic HFEs should be completely defined and used, with any exceptions clearly documented.	For all operator actions, revisit the application of PSF Adjustment rules to verify they have been applied consistently. If additional rules are needed for exceptions to the current rules, document those rules and provide a basis to support them.	The entire list of seismic HFEs was reviewed in HRA Calculator against the Table 8.4 of the station calculation for the Robinson Seismic Probabilistic Risk Assessment Human Reliability Analysis Notebook [65] detailed seismic HFE adjustment rules. During this review, the rules were either applied as-is or diversions from the rules were specifically noted on the first (BE Data) screen of the HFE. An internal review was also conducted to ensure correctness and concurrence with the seismic HRA bin rule application and documentation.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
		OPER-03 (implement feed and bleed) was modeled using the five seismic HRA bins. Several minor deviations were identified from the PSF adjustment rules. The S0 version (OPER-03-S0) uses moderate Stress, while S1, S2 and S3 use high Stress, inconsistent with the rules. For the S1 version, the Tdelay is the same as S0, but should be 2 min longer. For the S4 and S5 versions, the cause decision tree Pca use branch d rather then e. OPER-28 (provide alternate cooling to CCW) was modeled using the five seismic HRA bins for ex-MCR actions. Several minor deviations were identified from the PSF adjustment rules. The S0 version (OPER-28-S0) uses moderate			Some instances were also identified during the seismic HFE review where the rules themselves were refined; these changes were made to Table 8.4 in the station calculation for the Robinson Seismic Probabilistic Risk Assessment Human Reliability Analysis Notebook [65]. Updated HEPs resulting from this review are shown in Table 10.1. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard with NRC Clarification. This finding is considered resolved for the purposes of NTTF 2.1.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
		Stress, while S1, S2 and S3 use high Stress, inconsistent with the rules. For the S0 version, the Tcog is 15 min, but Tcog values for the S1 to S5 versions seem to be based on an S0 value of 16 min (then modified according to Table 8.3 of the station calculation for the Robinson Seismic Probabilistic Risk Assessment Human Reliability Analysis Notebook [65] OPER-14 (start deepwell pumps for AFW source) was modeled using the five seismic HRA bins (e.g., OPER-14-S1). However, this action has a large time margin (330 min) and Table 8.4 of the station calculation for the Robinson Seismic Probabilistic Risk Assessment			

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
		Human Reliability Analysis Notebook [65] includes the rule: If Time Margin is > 4 hours, no seismic bin values are calculated. Based on the long time margin, it is not clear why seismic-impacted HEPs were calculated. OPER-49 (manually initiate SI) was modeled using the five seismic HRA bins. However, For this action, only minor modifications were made to the HEPs (i.e., adding 2 minutes to Tdelay). It is not clear why the standard PSF adjustments have not been used. Perhaps because this is a memorized action with very limited time window (10 mins), unique rules have been used. If so, they have not been documented with a basis.			

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
SR C-SPR-D3	19-4	This SR requires the assessment of accessibility as impacted by seismic hazard level. The PSF Adjustments in Table 8.3 of the station calculation for the Robinson Seismic Probabilistic Risk Assessment Human Reliability Analysis Notebook [65] provide an increase in Texe to account for generic delays in access. However, no detailed assessments of operator pathways for ex-MCR actions were documented. Table C-1 in the station calculation for the Seismic Equipment List [64] provides a description of the location that ex-MCR actions are performed.	Assessment of accessibility for ex-MCR actions is critical to determine feasibility of these actions at different seismic hazard levels.	Revise Section C in the station calculation for the Seismic Equipment List [64] to include paths taken for each local operator action, including FLEX actions. In addition, provide a conclusion for each local action with regard to feasibility (based accessibility) and whether the timing-adjustment factors provided in the station calculation for the Robinson Seismic Probabilistic Risk Assessment Human Reliability Analysis Notebook [65] are sufficient to account for delays in operator transit.	This SR requires the assessment of accessibility as impacted by seismic hazard level. The PSF adjustments in Table 8.3 of the station calculation for the Robinson Seismic Probabilistic Risk Assessment Human Reliability Analysis Notebook [65] provide an increase in Texe to account for generic delays in access. However, no detailed assessments of operator pathways for ex-MCR actions were documented. Table C-1 of the station calculation for the Seismic Equipment List [64] provides a description of the location that ex-MCR actions are performed. However, this table does not provide a description of the pathways to the action location. Also, it does not provide a conclusion regarding whether the action is feasible based on accessibility and whether the timing-adjustment factors provided in the station calculation for the Robinson Seismic Probabilistic Risk Assessment Human Reliability Analysis Notebook [65] are

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
		However, this table does not provide a description of the pathways to the action location. Also, it does not provide a conclusion regarding whether the action is feasible based on accessibility and whether the timing-adjustment factors provided in the station calculation for the Robinson Seismic Probabilistic Risk Assessment Human Reliability Analysis Notebook [65] are sufficient to account for delays in operator transit. In addition, Table C-1 of the station calculation for the Seismic Equipment List [64] does not address the locations and pathways for performing FLEX actions (e.g., OPER-61).			sufficient to account for delays in operator transit. In addition, Table C-1of the station calculation for the Seismic Equipment List [64] does not address the locations and pathways for performing FLEX actions (e.g., OPER-61). Resolution: Telephone interviews were conducted with Operations personnel from RNP. Operator action pathways were discussed, and documentation has been updated in the station calculation for the Robinson Seismic Probabilistic Risk Assessment Human Reliability Analysis Notebook [65] to describe these pathways in detail. Further, a conclusion regarding the feasibility of these actions has been added to the aforementioned notebook. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard with NRC Clarification.

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
					This finding is considered resolved for the purposes of NTTF 2.1.
(SR C-SPR-B1/C-SPR-F2	24-1	Documentation of the review of internal events assumptions applicability to the SPRA is missing.	The second documented assumption in Section 4.0 of the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77] says: "The Internal Events PRA is used as the technical basis for both CDF and LERF. All assumptions and success criteria in the Internal Events PRA are retained in the SPRA for the portions of the sequence models that apply [This assumption provides continuity between the Internal Events PRA and the SPRA. Any future changes to the Internal Events PRA success criteria would be addressed as part of the maintenance and update process of the integrated PRA." Based on the answers to Peer Review questions, this assumption was meant to imply that a review of the assumptions associated with the internal events model was performed, and that all	Generate adequate documentation to confirm applicability of the underlying internal events for seismic.	A review of the internal events model and supporting documentation was performed in order to ensure applicability of the internal events modeling assumptions to the SPRA, as discussed in Section 4.0 of the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77]. However, the peer reviewer deemed this documentation inadequate per the Basis of Significance. One approach to resolve this F&O is to disposition every assumption pertaining to the internal events model and any supporting calculations. This approach was taken and a list of the assumptions with dispositions has been added to Section 5.3.8 of the station calculation for the H.B. Robinson Seismic Probabilistic Risk

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
			assumptions related to the portions of the internal events model that apply to the SPRA were retained. The only documented evidence of the review of the assumptions is the quoted assumption in the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77]. The review was verified/internally reviewed as part of model development and during cutset reviews of the model results. This high level statement does not provide adequate documentation of the applicability of the internal events model to SPRA. The review of the assumptions associated with the underlying model is a critical step and absence of its documentation does not allow for appropriate review or verification even from the internal reviewer to confirm that the appropriate modeling changes have been made (see part b of the SR). Note also that an assumption that, for example, resulted in not modeling specific components,		Assessment Model Notebook [77]. This documentation is considered more than adequate. No changes to the SPRA model were required based on the reviewed assumptions. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard with NRC Clarification. This finding is considered resolved for the purposes of NTTF 2.1.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			would be difficult to capture in a cutset review. This F&O is linked to SPR-B1 because any assumption that is not considered valid needs to be explained (or the model appropriately modified) to fully meet SPR-B1. Absent appropriate documentation it is not possible to fully confirm this, although there is no evidence of anything inappropriately modeled.		
SR C-SPR-E3/C-SPR-E7	24-2	There is no optimization of the hazard bins used quantification.	Section C.6 of the station calculation for the Robinson Seismic Probabilistic Risk Assessment Quantification Notebook [13]. says: "Ten (10) hazard intervals provide a very reasonable compromise between quantification precision, model maintenance, and quantification processing challenges. If more hazard intervals were built into the model, the overall calculated SCDF and SLERF may reduce by a few percent, depending upon the	Investigate the effects of a more refined binning of the hazard curve at lower g levels. If the current binning is retained confirm that this is not overestimating significantly CDF at lower g levels.	The effects of refining the hazard bins were investigated and discussion has been added to Section C.6 of the station calculation for the Robinson Seismic Probabilistic Risk Assessment Quantification Notebook [13] It was determined that increasing the number of total bins has little impact on the CDF/LERF and importance measures results, while leading to much longer quantification and importance measure calculation times. Therefore, ten hazard bins are still used for quantification.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			interval slicing and number of intervals". In response to a Peer Review question, it was concluded that a sensitivity to show stability of the SPRA results with respect to the number and size of hazard intervals was not performed. A sensitivity (Case IE-1c) to show the potential change in SCDF and SLERF that would result from adding more intervals was performed and documented in the station calculation for the Robinson Seismic Probabilistic Risk Assessment Uncertainty and Sensitivity [70]. From figure 4-3 of the station calculation for the Robinson Seismic Probabilistic Risk Assessment Quantification Notebook [13]. it is evident that the CCDP of the plant goes to 1.0 (except for the plant availability factor) after the 3rd hazard interval. After that, the CDF contribution is essentially the hazard curve. There are therefore only 3 intervals that meaningfully describe the plant response to the seismic event. This is not consistent with common practice, which is to generate a more refined binning of the hazard		Similarly, it was determined that rebinning the hazard curve so that the lower acceleration levels are separated into more bins has little impact on the CDF/LERF and importance measures results; however, the model was revised to place more bins at the lower accelerations and to condense the higher accelerations into fewer bins. This rebinning was performed in order to provide additional intervals where CCDP/CLERP is less than 1.0 and to be more consistent with common practice. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard with NRC Clarification. This finding is considered resolved for the purposes of NTTF 2.1.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			in the interval that is more meaningful for the quantification. This can also overestimate the current risk profile. This F&O is written against back-referenced SRs QU-B1, B2 and B3 because this limitation (B1) is not appropriately addressed and the truncation (B2 and B3), which in seismic PRA is to be intended as a more generic stability of the results, including bin numbers and size, is not fully investigated. The F&O is also applicable to SPR-E7 for back-referenced SR QU-E4 because this uncertainty in the quantification process is not assessed (note that adding a single hazard bin at the end is meaningless if CCDP is already 1.0).		
SR C-SPR-A2	24-4	Incomplete identification of seismically-induced consequential events.	Section 5.3.7.2 of the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77] discusses Seismic Hazards Other Than Vibratory Ground Motion but there is no apparent discussion of the potential for seismic-induced equivalent of other external	Perform a systematic review of other external hazards (beyond the secondary hazards that have been addressed) that may have a seismic-equivalent and address any events that may not be screened out.	Section 5.3 of the SPRAIG [11]. was used as guidance to perform a systematic review of other external hazards that may have a seismic-equivalent. Each identified external hazard was then dispositioned for inclusion in the SPRA. Section 5.3.7.3 was added to the

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			initiators (e.g., seismic failure of nearby major gas lines). Based on Duke's answer to a Peer Review question, the IPEEE analysis is available for Robinson. The information was reviewed to support the SPRA. Formal review was performed as part of the station calculation for the Seismic Equipment List [64], but not specifically to screen out the potential for other seismic-induced external initiating events. A screening analysis of other seismic hazards was performed specifically for the SPRA. Section 11 of Attachment 1 to the station calculation for the Final Seismic Analysis Report [16] provides the screening of other seismic hazards including, fault displacement, landslides, soil liquefaction, soil settlement, earthquake-induced flooding, fracking-induced earthquakes, and seismic seiches (this addresses hazards identified in SHA-I2). All other seismic hazards were screened from the SPRA except soil liquefaction and soil settlement.	Section 5.3 of the SPRAIG [11] can be used as guidance to perform this task	station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77] in order to document this review. For each identified hazard, a determination was made whether to include the hazard in the SPRA and a disposition is provided. As stated in the Basis for Significance, multiple hazards were included in the SPRA. No additional hazards were modeled in the SPRA as a result of this review. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard with NRC Clarification. This finding is considered resolved for the purposes of NTTF 2.1.

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
			Section 5.3.7.2 of the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77] discusses only lateral spreading and liquefaction-induced settlement failures since these were not screened for RNP. The station calculation for the Robinson Nuclear Power Plant Seismic Induced Flood and Fire Assessment [69] analyzes seismic-induced internal fires and floods as a result of seismically failed SSCs.		
SR C-SPR-B3/C-SPR-F2	24-6	Incomplete documentation of the mapping of seismic failures to existing basic events.	The station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77] Appendix B is a report from the FRANX fragility to Component table. This report of the mapping reflects the mapping used in the model but per se does not document the mapping (i.e., it does not provide reason for the mapping). While it is recognized that a large portion of the mapping is straightforward (i.e., a fragility group for a single component or two unequivocally identical	Ensure that there is adequate documentation of the mapping of the seismic failures (i.e., fragility groups) to basic events included (or appropriately added) in the model when the mapping is not straightforward.	Appendix B of the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77] documents the FRANX Fragility_To_Comp table, which provides the BEs mapped to each fragility group. The mapping is straightforward (e.g., the RWST fragility group is mapped to the failure of the RWST BE in the model). Some items were identified in the Basis for Significance that made it appear the mapping was not as straightforward; these items are dispositioned below.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			components is mapped only to the basic events for those components) some other mapping is less straightforward and needs to be explained. This is particularly evident for 2 over 1 failures mapping. An example of this is the mapping of the fragility group for the charging pump to the %S1 initiator. Based on Duke answer to the peer review question, '"the mapping of the charging pump fragility group to initiating event %S1 (the fragility group is not mapped to %S2) was an original conservatism that should have been removed following the addition of the very small LOCA (VSLOCA) logic in the model. Originally it was assumed that a loss of all charging pumps would result in VSLOCA conditions requiring long-term RCS makeup. Due to the high capacity of the charging pumps, this conservatism has negligible impact on the results." Another example is the SF-PA-HAGAN fragility group, that is modeled both to the DUMMY_CD and to other specific basic events,		Additionally, all group to BE mappings were reviewed and no other issues were identified. 1. Fragility group SF-PM-CHG-PMP is no longer mapped to initiator %S1. This was an original conservatism in the model that should have been removed following the addition of the VSLOCA logic in the model. 2. Fragility group SF-PA-HAGAN is now only mapped to Dummy_CD. This group was originally mapped to a specific BE in the model but was then updated to be mapped to Dummy_CD; however, the mapping to BE SF_PA_HAGAN was inadvertantly not removed from FRANX. Additionally, BE SF_PA_HAGAN has been removed from the model as it is no longer used. Hagan Racks have been added to the list in Section 5.3.6 of the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77] 3. Fragility group SF-MV-RCP-CLNG-MOV is no longer mapped to

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
			without explicitly showing up in section 5.3.6 of the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77] nor anywhere else in the notebook where a description of the reason of this double mapping is provided. Absence of this documentation makes the internal review and verification of the appropriate modeling (covered in SPR-B3) challenging.		Dummy_CD. This group was updated to be mapped to specific BEs instead of Dummy_CD; however, the mapping to Dummy_CD was inadvertantly not removed from FRANX. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard with NRC Clarification. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SPR-E6/C-SPR-F2	24-7	Documentation of the modeling associated with DUMMY_PDS is missing.	One direct to CD accident sequence was added to the logic (via mapping to the DUMMY_CD, thus the relevance of this F&O to SPR-B8), and not all the contributions are appropriately transferred to LERF. Duke answer to the peer review question explained which subset of fragilities (among those mapped to DUMMY_CD) is also mapped to	Document the rationale for the mapping to DUMMY_PDS with a rationale for the selection of equipment. Justify not appropriately transferring CDF sequences to LERF (e.g., a sensitivity can be provided where either DUMMY_PDS is	Discussion of the mapping to basic event DUMMY_PDS has been added as the second bullet in Section 5.3.6. of the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77] a subset of those fragility groups mapped straight to core damage are also mapped straight to large early release by mapping the

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
			DUMMY_PDS). There is no documentation of the rationale for this mapping beyond the mere reporting of the mapping. This modeling is not straightforward and needs to be documented and clarified.	removed from the logic or all the fragilities mapped to DUMMY_CD are also mapped to DUMMY_PDS).	groups to BE DUMMY_PDS. While the mapping of these fragility groups directly to core damage is a known conservatism with little impact on the results, allowing the failure of these SSCs to go to LERF is overly conservative. Therefore, the DUMMY_PDS basic event is introduced to limit which fragility groups impact LERF. No model changes were made in resolving this finding. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard with NRC Clarification. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SPR-E7	24-8	One important assumption (i.e., SDAFW pump surviving the failure of the Class	The catastrophic failure of the Class III TB is modeled with a 50% chance of impacting the Class I TB (apparently based on the prevalent	The survivability of the SDAFW following any failure of the Class III TB is a potential uncertainty	The plant model assigns a 50% probability of SDAFW pump failure due to failure of the Class III Turbine Building. In this scenario,

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
		III TB failure) is not addressed for the associated uncertainty.	direction of the collapse). When the Class III TB falls towards the Class I TB, everything in the TB is assumed to be failed (including the SDAFW pump). When the Class III TB falls away from the Class I TB, then the CST is impacted but the SDAFW pump is considered unaffected. It is noted that the SDAFW is a couple of feet away from the invisible line that separates the Class I and Class III portion of the TB. During the peer review walkdown the team observed significant equipment of relatively large dimension crossing the boundary between the two TB sections. The peer review team also observed elements on the SDAFW skid that could be judged as soft target (in disagreement with the original SEWS). It is at least unsure that a collapse of the Class III TB would leave the SDAWF unaffected. In answering to the peer review question, Duke pointed to a sensitivity performed on the split fraction (current 50-50 for the base case) which only partially	that should be addressed. A conservative way would be to fail specifically SDAFW every time the TB Class III fails. A dedicated sensitivity could be performed to address this eventuality. Additional justification should probably be provided to support the current base case in term of the specific soft targets observed during the peer review walkdown.	the Class III Turbine Building collapses towards the Class I Turbine Building causing damage to the SDAFW pump located in the latter. A follow-up walkdown was performed to determine if the SDAFW pump can be damaged if the Class III Turbine Building collapses in some other direction; e.g., away from the Class I Turbine Building. The walkdown team accessed the SDAFW pump skid and reviewed and measured the distances, envisioning how a collapse might occur and its effect on the pump. The distance measured from the closest point of the Class III Turbine Building to the SDAFW pump skid was a minimum of eight feet. Equipment mounted on the pump exposed to the Class III Turbine Building includes the pump suction piping, various associated process gauges and transmitters, some lube oil piping, and a control valve and tubing associated with temperature control. The turbine speed governor, steam piping, and pump discharge line are located on the opposite side of the skid and were judged to be shielded from elements of the collapsed Class III

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			addresses the concern of damaging the SDAFW even if the Class III TB does not interact significantly with the Class I TB (i.e., damage to the SDAWF would be due to ancillary failures and interaction). Duke also pointed to a December 2017 memo from the Fragility Vendor. Such memo, again, supports the 50% chance of interaction, but does not answer the original question as page 2 of the memo says: "The document requests that the Frag, “Show that the SDAFW pump and all support SSCs will survive seismic failure of the Class I Turbine Building, including any above ground suction, steam inlet, turbine cooling, and pump discharge piping, as well as DC power cables, given seismic failure of the Class III Turbine Building and subsequent interaction with the Class I Turbine Building.” Based on the station calculation for the Turbine Building Class III fragility) for the Class 3 Turbine Building, we judge the Steam Driven Auxiliary Feedwater		Turbine Building that might be present. Piping in the vicinity of the pump passes across the boundary between the Class I and Class III Turbine Buildings. The larger piping includes a 20-in. diameter feedwater line and a 16-in. diameter line from the heater drain tank. This piping is separated from the SDAFW pump by distances of several feet. In addition, one of the Class I Turbine Building columns located between the pump and the feedwater line shields the former from impact by the latter. The walkdown team judges that the available spatial separation and shielding is sufficient to preclude damage to the SDAFW Pump even If the Class III Turbine Building were to fail in some direction other than towards the Class I Turbine Building. During the walkdown, the AFW system engineer was interviewed to better understand how damage to the various components might affect pump operability. He indicated that the pump/turbine assembly was modified so that it is

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			pump unlikely to survive failure of the Class 1 Turbine Building.		now self-cooling, and the control valve and tubing that provided cooling previously have been abandoned in place. Therefore damage to these items would not affect the pump's operability. The gauges and transmitters on the suction pipe could very well be damaged and, given a breach in the line some water would be lost, but the overall volume of process fluid passing through the pumps would be minimally affected. This is consistent with the resolution to F&O 29-5, Item 3 that there are no soft targets whose failure could prevent pump function. Based on the above, the walkdown team judged that the probability of failure of the SDAFW pump given failure of the Class III Turbine Building is 50%. The SEWS for the SDAFW pump has been updated for the above and is in the station calculation for the Seismic Capacity Walkdown Report [53]. Additional information may be obtained from this SEWS. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
					PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard with NRC Clarification. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SPR-B3/C-SPR-F2	24-13	Discrepancies between the model notebook log change and the CAFTA model.	Some inconsistencies have been observed between the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77] Table A-1, that discusses the model log changes, and the actual final model. An example is gate CLASS-3-TB, which is documented in entries #34, 35, 36 and 37 of Table A-1 of the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77] , from which it seems justified under the following gates in the model: FMMSEGAMFN, FMMSEGBMFN, FMMSEGCMFN, FFLEX002. Gate CLASS-3-TB is apparently present in the model in multiple other places (e.g., under gates	Resolve the inconsistencies between the model and the change log. It is observed that issues have been observed on the first two randomly selected entries of Table A-1 of the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77] . An extend of condition is recommended. Consider adding enough details in Table A-1 of the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77] (either in	Inconsistencies were identified between the SPRA model and the change log provided as Table A-1 in the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77]. Furthermore, it was suggested that additional detail be provided for model changes, as needed. The specific items provided in the Basis of Significance pertaining to gate CLASS-3-TB and the lateral spreading failure of the RWST, have been corrected in Table A-1 of the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77] . An extent of condition review was performed, and Table A-1 of the station calculation for the H.B. Robinson

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
			FDGAFW-CFN-TB3, ATK%%PACFF-TB3, QAVV1-3CFF-TB3, etc.…). In response to the peer review question, the RNP S-PRA indicated that the other gates should also be added in the change log. Another example is entry #31 of Table A-1 the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77] that does not match with the model (the model shows that gate SEISMIC_SPREAD_DC2-3 is under some additional/different gates not reported in the notebook (e.g., RWST-1). In response to the peer review question, the RNP S-PRA confirmed that the model seems to be correct and the issue remains in the documentation.	the notes or in the description of the change) to describe why the change is done and not only that is done, and if applicable point to a section in the notebook where the rationale is further discussed. This would make easier to review and confirm than the modeling and the log are complete and comprehensive.	Seismic Probabilistic Risk Assessment Model Notebook [77] was corrected as necessary to match the SPRA model. Additionally, the comments for some modeling changes were enhanced in Table A-1 of the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77] . Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard with NRC Clarification. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SPR-E3	24-15	Limited quantification results provided.	To meet back-reference SR QU-A2, the quantification needs to be performed at the sequence level as well as the top level. It is recognized that in a seismic PRA, when multiple initiators are in	Present the quantification results in a more refined fashion (e.g., as applicable at least classes of plant initiators (e.g., LOCAs, LOOPs,	The station calculation for the Robinson Seismic Probabilistic Risk Assessment Quantification Notebook [13] has been revised to provide the risk contributions from the top accident sequences for

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
			theory possible, a quantification on a sequence by sequence basis may not be practical. Nevertheless, the station calculation for the Robinson Seismic Probabilistic Risk Assessment Quantification Notebook [13] has been revised to provide the risk contributions from the top accident sequences for CDF and LERF. The accident sequence contributions are now provided in Section 4.1.3. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard with NRC Clarification. does not discusses this limitation nor presents any additional break down of the results beyond the break-down based on g levels.	others)) to meet the requirement/intent of QU-A2.	CDF and LERF. The accident sequence contributions are now provided in Section 4.1.3 of the station calculation for the Robinson Seismic Probabilistic Risk Assessment Quantification Notebook [13]. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
SR C-SPR-E3	24-16	Base case LERF is not converged per Standard requirements.	The suggested rationale for the 5% difference between the base case LERF and the best achievable truncation is based on a speculation and not supported by any additional considerations. It is also noted that the top cutsets are normally the ones that have much more of an impact in CDF/LERF reduction when post-processed via ACUBE, so some of the LERF increase may be real and not only related to ACUBE limitations.	Use the deepest truncation scheme as base case LERF or provide additional justification that LERF is converged.	To demonstrate convergence the quantification was modified. The details can be found in the station calculation for the Robinson Seismic Probabilistic Risk Assessment Quantification Notebook [13]. The results show that the percent change between the baseline and lower truncation limit cases is approximately 6.0%. If the CLERP is assumed to be the ACUBE code lower bound limit for each of the hazard intervals for which ACUBE could only process a subset of the cutsets, then the percent change would be approximately 4.6%, which is less than 5%, indicating the results would be close to the 5% convergence value if more cutsets could be evaluated using the ACUBE code. Using the truncation values from the lower truncation case requires a much longer calculation time and does not meaningfully impact the risk results, the importance measures, or the top contributors. Therefore, the model is considered to show convergence at the chosen truncation levels and is documented in the station

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					calculation for the Robinson Seismic Probabilistic Risk Assessment Quantification Notebook [13]. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SPR-E3	24-17	No discussion on circular logic check/correction.	QU-B5 requires to address circular logic. While CAFTA will automatically stop the quantification if circular logic is detected, there is no documentation of whether any specific model change was performed to resolve any circular logic issue.	Confirm that no model changes were made to address circular logic issues.	No specific model changes were made in order to address circular logic, as circular logic was not introduced due to the incorporating of the SPRA logic. This statement has been added to Section 5.3.3 of the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77]: "No circular logic was introduced by the addition of the SPRA logic;

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					therefore, no model changes were made to address circular logic." Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SPR-E3	24-18	No evidence of mutually exclusive logic check for seismic relevance.	Back-referenced SR QU-B7 and QU-D3 requires to address mutually exclusive logic and flags potentially impacting the results. It is understood that any mutually exclusive logic retained from the internal events model have been carried through the in seismic PRA. There is no documented evidence of any review performed to confirm that:1. The mutually exclusive logic is appropriate also in case of seismic, where correlation can challenge the original logic;2. No fragility group (especially those that are mapped to multiple basic	Confirm that the mutually exclusive logic present in the S-PRA is applicable and that no fragility groups are prevented from propagation.	The RNP mutually exclusive logic was reviewed for applicability to the SPRA and was determined to be applicable. Furthermore, the mutually exclusive logic does not hinder failure propagation of the fragility groups. This statement has been added to Section 5.3.3 of the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77]: "The mutually exclusive logic contained in the Internal Events model has been reviewed and is

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
			events) are prevented from propagating through the logic from the existing mutually exclusive logic. Based on discussion with the RNP S-PRA team, no mutual exclusive logic has been added explicitly for the S-PRA.		applicable to the SPRA. The mutually exclusive logic does not prevent the propagation of seismic failures for any fragility groups. No mutually exclusive logic was added specifically for the SPRA." Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SPR-E3	24-19	There is no evidence of a review of non-significant cutsets consistent with current expectation from the NEI guidance [6] (i.e., a minimal number of randomly selected cutsets for each decade).	QU-D5 explicitly requires to review non-significant cutsets. Review of non-significant cutsets is mentioned in the minutes of interim cutsets review but there is no discussion on how many and how deep in the model. NEI guidance suggested the minimal number of randomly	Provide evidence of a sufficient review of non-significant cutsets to assess model inconsistencies or simplifications. Good practice would be to also document the non-significant cutsets that are reviewed,	A formal cutset review was performed which included a review of cutsets from each truncation decade for both CDF and LERF; thus, a review of non-significant cutsets was performed. All cutsets were determined to be reasonable. This cutset review is documented as Section H.3 of the station calculation for the Robinson Seismic Probabilistic Risk Assessment Quantification

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
			selected cutsets for each decade to be addressed.	although it is understood not to be a requirement.	Notebook [13] and is discussed in Section 4.1.4.2 of the station calculation for the Robinson Seismic Probabilistic Risk Assessment Quantification Notebook [13]. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard.
SR C-SPR-E7/C-SPR-F3	24-20	No systematic review of specific assumptions for their impact on the risk profile and the associated insights.	Provide a systematic review and uncertainty estimate of the assumptions supporting the S-PRA. This should include: 1. Assumptions from the SHA assessment (because the SSHAC process already converts epistemic uncertainties in the fractiles used for the quantification, it is not expected that all the assumptions are considered, but only those that can be addressed with a potential sensitivity);	Systematically process all the assumptions supporting the S-PRA through the process discussed in QU-E4. It is suggested that what Duke judges are the key assumptions supporting the different aspect of the RNP S-PRA (see points 1 through 6) are initially collected in the subject summary reports, and that the uncertainty report addresses the	A systematic review of key assumptions supporting the SPRA was performed. All aspects of the SPRA were reviewed including the seismic hazard assessment, the seismic fragility assessment, the seismic logic model development, and the seismic HRA. The uncertainty associated with all identified key assumptions is characterized; sensitivity analyses were performed to evaluate the uncertainty as applicable. This review and uncertainty evaluation is documented in Section 3.1 and associated sensitivity analyses are

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
			2. Assumptions from the SFR evaluation. F&O 24-8 is an example of this. 3. Assumption from the underlying internal events model (at least the assumption that were recognized to have model uncertainties in the IE uncertainty assessment should be re-visited in the S-PRA. 4. Assumption from the S-PRA logic model development; 5. Assumptions from the seismic HRA 6. Assumption from the quantification process (the hazard binning addresses this in some extent). F&O 24-2 is an example of this.	uncertainty associated with them in a systematic way. Consistent with the expectations form back-referenced SR QU-E4, the uncertainty assessment for each assumption may have multiple elements, and there is no expectation or requirement that a dedicated sensitivity analysis is performed for each assumption. It is suggested that, for each sensitivity that is indeed performed for this purpose, not only the overall CDF/LERF results are reported, but any indication of changes in insights (e.g., unexpected fluctuation of importance measures, etc…) It is also suggested that the range of CDF/LERF calculated based on sensitivities are	detailed in Section 4 of the station calculation for the Robinson Seismic Probabilistic Risk Assessment Uncertainty and Sensitivity [70] Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard with NRC Clarification. This finding is considered resolved for the purposes of NTTF 2.1.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
				overlapped with the parametric uncertainty range to assess how CDF/LERF may change in complex. This would be valuable in a risk-aggregation perspective.	
SR C-SPR-E6	24-21	No timing change considerations for LERF.	There is no evidence that the RNP S-PRA team addressed the need to revisit the binning of CDF to LERF sequences.	Justify that the binning of CDF sequences into LERF PDS is still applicable for higher g levels where evaluation time can be challenged (i.e., the definition of Early should be confirmed applicable). Re-bin any non-LERF sequences into LERF if the timing can be impacted.	Although the “Late” time is not explicitly defined with a single evacuation time value, the MAAP results show the non-LERF/late release categories have a range of 9 hours to 25 hours between vessel failure and containment failure. Other non-LERF may be early or late categories but have small release magnitude and thus are not LERF. Thus, all late release categories are also medium or small in magnitude, except for one release category which is large in magnitude but has ~20 hours between vessel failure and containment failure, which is sufficient for evacuation, so no late release categories need to be re-binned to LERF for the RNP SPRA. Further details on this F&O can be found in the station calculation for

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					the RNP SPRA Peer Review Resolution [71] Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SPR-E6	24-22	No LERF breakdown per PDS	LE-F1 CC-II requires a break-down of the LERF results based on PDS.	Provide a LERF result breakdown based on PDS.	The station calculation for the Robinson Seismic Probabilistic Risk Assessment Quantification Notebook [13] has been revised to provide the risk contributions from the top plant damage states for LERF. The plant damage state contributions are now provided in Section 4.1.3 of the station calculation for the Robinson Seismic Probabilistic Risk Assessment Quantification Notebook [13]. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
					The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SPR-E6	24-23	Underlying internal events LERF SRs met at CC-I.	The seismic PRA uses the internal events LERF as basis. A number of back-referenced SRs are only met at CC-I in internal events LERF and no seismic specific changes are made.	Resolve findings in the underlying LERF study.	Performing a LERF analysis per NUREG/CR-6595 [91] is an acceptable methodology as referenced in Reg. Guide 1.174 Rev 3 [92]. The advantage of this approach is that it allows LERF to be calculated quickly, though approximately, without the need for performing a detailed Level-2 PRA and the NRC has explicitly accepted this approach as being sufficient for the determination of LERF. The Robinson LERF analysis employs a methodology fully endorsed by the NRC as an acceptable means of calculating LERF for use in risk-informed applications, including changes to its licensing basis as well as its response to NTTF 2.1 seismic. This does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
					[5 and 37] and is not considered an upgrade. The response to this finding meets the requirements of NTTF 2.1 seismic and Capability Category I of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SPR-A3	25-1	SPR-A3 requires a review of plant specific response to any past seismic event. The RNP SPRA documentation does not include a discussion of any applicable site specific events (or relevant industry events). Although mention is given to the SPID [2]and SPRAIG [11] in the self assessment, no discussion was found in the SPRA notebooks.	The self assessment states that Industry documents such as the SPRAIG [11] and SPID [2] were reviewed to ensure that the list of initiating events included in the Robinson SPRA accounts for industry experience. Beyond reference to the industry documents in the self assessment, no specific review of other SPRAs was included. No documentation of the findings of the industry document review with respect to this requirement were found.	Perform a review of plant specific events (if applicable) and relevant seismic risk evaluations from other plants.	No plant-specific seismic events have occurred. Review of relevant seismic risk evaluations from other plants was performed. A review of the Surry and Oconee SPRAs was performed in order to ensure all initiating events are accounted for in the RNP SPRA. These evaluations considered events, such as, seismic-induced flooding, seismic-induced fires, and seismic-induced failures of structures and dams, which are similarly evaluated in the RNP SPRA. No new initiating events or accident sequences were included in the RNP SPRA based on the review of the Surry and Oconee SPRAs. Discussion on this review has been added to Section 5.2.1 of the plant station calculation for the H.B. Robinson Seismic Probabilistic

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
					Risk Assessment Model Notebook [77] Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard.
SR C-SPR-A2	25-2	Based on the information in the FRANX database, component DS-BUS has a representative HCLPF of 0.10, well below the screening HCLPF of 0.75. The potential for and consequences of seismically induced bus arcing fire was not assessed for this component.	SPRA-A2 requires that a systematic process be performed to identify initiators caused by secondary hazards. Not assessing the potential for fire due to arc flash from an electrical bus failure may result in omission of a valid initiating event.	Review the potential for a seismically induced bus duct fire associated with DS-BUS and identify the consequences of such a fire if it can't be screened based on frequency alone.	Because of the uncertainty in the capacity of the anchorage for the DS-BUS and the potential for interaction between adjacent panels, the DS-BUS was ranked Low (L) regardless of seismic deficiencies identified during the seismic capacity walkdown [53]. This resulted in a judgment-based lower bound fragility with a HCLPF capacity of 0.1 g PGA. This fragility information was first used for defining the DS DG System Fragility Group, SF-DG-DS-SPRT (DSDG Supporting Equipment). Then, this fragility

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					group was mapped to various DSDG-related items [77]. The fire compartment where the DS Bus is located is FC250 and the fire ignition source ID assigned to the DS Bus is 1448 [81]. The conditional core damage probability associated with this scenario is 4.39E-04, setting severity factor and non-suppression probability all conservatively to one due to seismic damage anticipated in the area [81]. This CCDP is used in the bounding seismic CDF analysis for the purpose of screening the DS Bus duct as a seismic-induced fire ignition source. The resulting total seismic CDF is 5.07E-8, which is less than the threshold value of 5E-07 for screening. Based on this result, the seismically induced bus duct fire associated with DS-BUS is screened out of the RNP SPRA. This is documented in the station calculation for the Robinson Nuclear Power Plant Seismic Induced Flood and Fire Assessment [69].

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
					Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard with NRC Clarification. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SPR-A2	25-3	SPRA-A2 requires that a systematic process be performed to identify initiators caused by secondary hazards. The assessment used to screen out the Lube Oil (LO) storage tank fire scenario is not properly characterized.	The SIFF Notebook statement that the class III turbine building will collapse before any LO tank fire can occur is misleading. There will be two scenarios or cutsets: 1. One in which the LO tank fails and TBIII does not fail with a subsequent LO tank fire. 2. A separate scenario will occur when the LO tank does not fail and then the TBIII fails. In all likelihood, the probability of LO tank seismic failure combined with a low ignition probability (due to high flash point, potential suppression etc.), will have an	The scenario with the LO tank failure and ignition may be insignificant compared to TBIII failure, but the scenario would occur independent of TBIII failure and should be screened in this context	The Lube Oil (LO) Storage Tanks are located in the southwest portion of the Class III portion of the Turbine Building. The finding points out Sections 1.2.1 and 4 of the station calculation for the Robinson Nuclear Power Plant Seismic Induced Flood and Fire Assessment [69] as an inappropriate basis for screening the LO tanks as seismic-induced fire ignition source. The LO storage tanks have the low probability of fire damage due to the high lube oil flash point (432° F) and the availability of fire detection and suppression system to limit the spread of the fire. The high flash point would make it difficult for a

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
			insignificant contribution when compared to the TBIII failure.		fire to occur even if a seismic event was to rupture the LO storage tanks and a damaging hot gas layer is unlikely to form due to the Turbine Building being large and open to the outdoors. Based on this consideration, the two quoted sentences have been deleted to clear up the inconsistency noted in the Finding in the station calculation for the Robinson Nuclear Power Plant Seismic Induced Flood and Fire Assessment [69]. This is a documentation issue only and this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard with NRC Clarification. This finding is considered resolved for the purposes of NTTF 2.1.

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
SR C-SPR-A1	25-4	There is no description in the modeling notebook of a systematic disposition of internal initiating events that were included in the SPRA.	SPRA-A1 requires that a systematic process be performed to identify initiators caused by the seismic event. No evidence of a systematic process to identify seismic imitators was identified. Table A-4 of the station calculation for the Seismic Equipment List [64] lists the internal events initiators that were screened out as non-applicable and Table A-22 in the station calculation for the Seismic Equipment List [64] documents the disposition of all basic events and initiating events in the model for the purposes of SEL development however, they are reviewed only for the purposes of SEL development and not for identification of initiating events. Based on Duke's response to a Peer Review question, there are several other initiators modeled in the internal events PRA that are not listed in Section 5.3.1 of the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77]	Document a systematic review of each internal initiating event and identify whether or not that event should be included as an initiating event in the seismic PRA. During the peer review, the Robinson SPRA team described a process that would address this issue. This process involved a comprehensive review of internal and flooding initiating events and a subsequent disposition for each with regard to system impacts or mapping to seismic initiating events.	A systematic review of all internal events and internal flooding initiating events were reviewed for inclusion in the SPRA. Table 5-1 was added to Section 5.3.1 of the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77], which provides a list of the initiating events and whether to include each initiating or not in the SPRA, both initial and final assessments. A final disposition is provided for each initiating event, which either states why an initiating event is not included or how it is utilized in the SPRA. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
			A review of all initiators in the internal events model was performed and dispositioned for the SPRA; however, a table of each initiator with its disposition is not formally documented. Appropriate documentation of the rationale for dispositioning seismic-equivalent of internal events initiators needs to be provided to support reproducibility of the analysis and future maintenance of the SPRA.		
SR C-SPR-B2/C-SPR-F2	25-6	The description in Section 7.0 of the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77] regarding disposition of open F&O AS-A5-1 does not include discussion of the seismic modeling assumption for steam line break.	The response to a Peer Review question indicates that impact of TB failure on steam lines was considered in the disposition to the open F&O (AS-A5-1). This discussion is important to understanding how this F&O does not impact the results of the SPRA.	Include the discussion in the response to the peer review question regarding assumed MSIV closure given steam line failure due to turbine building seismic failure.	Internal Events F&O AS-A5-1 was dispositioned in Section 7 of the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77]; however, the discussion did not include the assessment of the seismic modeling for the steam line break. The assessment concluded that the MSIVs would remain available to provide function to isolate the main steam line given a Class 3 TB failure, including failure which results in interaction with the Class 1 TB.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
					This discussion has been added to the disposition of AS-A5-01 in Section 7 of the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77]. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard with NRC Clarification. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SPR-B8	25-8	FLEX AFW pumps in the RNP SPRA use the internal event AFW pump reliability. Use of data for safety related installed plant equipment for diesel driven mobile FLEX equipment may not be appropriate.	Section D.5.6 of the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77] discusses the decision to use nominal AFW pump reliability data for the diesel driven AFW pumps. Because of the distinctly different characteristics between the mobile FLEX AFW pumps and the installed, motor driven AFW	Recommend either use of available industry data for FLEX equipment or to adjust the Robinson plant data to account for the expected difference in reliability between installed plant equipment and mobile FLEX equipment.	Nominal AFW pump reliability data was originally used as a surrogate for the FLEX pump reliability data. The FLEX pump data has been updated based on Duke fleet-specific FLEX data. Duke PRA collected FLEX pump data from all six of its fleet sites. This data was used to determine FLEX pump failure probabilities.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			pumps, use of this data is not appropriate. (refer to DA-D2).		Section D.5.6 of the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77] has been revised with this discussion and the testing data. Sections D.3.4 and D.3.6 of the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77] were also revised. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SPR-B9	25-9	Documentation of the eyewash station seismically induced flood does not address the potential for propagation into other areas, features that	This IFSN HLR requires an assessment of flooding propagation path and identification of potential downstream impacts. Discussion of the eyewash station flooding scenario in the seismically induced	In the seismically induced fire and flooding document, provide either a reference to the relevant section of the RNP internal flooding PRA or include a	The concern about the eyewash station which is supplied by the Potable Water System and it's potential breach and subsequent consequences was conservatively evaluated. With a LOSP, the

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
		could terminate propagation or operator actions to terminate flow.	flooding PRA does not address the potential for propagation. A response from the RNP PRA modeling consultant indicates that the RNP internal flooding PRA has assessed propagation for this scenario but this information is not contained in the seismic fire and flooding documentation.	discussion of the eyewash station flooding modeling considerations (e.g. assessment of propagation).	Potable Water System ceases to be a flood source. In other words, the loss of offsite power prevents or terminates a seismic-induced flooding event. To better understand the risk impact of this scenario, a bounding CDF assessment was performed using the fragilities for the eyewash station and the LOSP. Using this seismic-induced flooding fragility, a bounding risk assessment was performed to gauge the impact of this flooding event on the overall seismic CDF. The results demonstrate the insignificance of the seismic-induced flooding event initiated by the eyewash station failure and provides the justification that additional flooding scenario for flood propagation does not need to be retained for the RNP SPRA model. Further details on this F&O can be found in the station calculation for

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					the RNP PRA Model Peer Review Resolution [71] This does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SPR-A2	25-10	Some components that should be retained for further assessment based on EPRI Report 3002012980 [93] are not retained.	The station calculation for the Robinson Nuclear Power Plant Seismic Induced Flood and Fire Assessment [69] references the EPRI Report 3002012980 seismically induced fire screening methodology [93]. Table 2-2 of this EPRI report lists ignition sources to be considered for the SPRA. Air compressors, pumps, diesel generators, and bus ducts are listed as 'Retain' in Table 2-2 but are not dispositioned in the station calculation for seismic induced flood and fire assessment [69].	Disposition the screening of all ignition sources listed as “Retain” in EPRI 3002012980 [93] Table 2-2 or include them in the SPRA. Consider the need to conduct walkdowns of equipment that should be retained per SFR-D6.	Robinson Fire PRA identified a total of 2120 fire ignition sources with different fire scenarios [81]. These sources are processed using a combination of qualitative and quantitative screening filters. Details are provided in the station calculation for the Robinson Nuclear Power Plant Seismic Induced Flood and Fire Assessment [69]. Based on the results of these screening steps, there are no potential seismic-induced fire

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
					ignition sources that need to be retained in the Robinson SPRA. This does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard with NRC Clarification. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SPR-B9	25-11	Potential flooding consequences of heat exchanger anchorage failure not assessed.	The failure mode listed in the station calculation for Seismic Fragility Evaluation Notebook [82] for heat exchangers is anchorage failure. In the RNP SPRA model, the seismic failure is mapped to a plugging failure of the heat exchanger and seismically induced flooding impacts are not assumed. These flooding impacts may be limited to loss of the system function (e.g. CCW) or could involve flooding induced failure of nearby components.	Assess whether an anchorage failure for heat exchangers would also result in a seismically induced flood.	A bounding SCDF was computed using the failure of the Robinson Dam to represent the Service Water and inducing a flood when the CCW Hx anchorage fails and taking the consequence directly to core damage. The results show that overall SCDF remains essentially the same relative the baseline SCDF. This demonstrates the insignificance of the seismic-induced flooding event initiated by the CCW Hx anchorage failure and provides the justification

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
					that the postulated flooding scenario does not need to be retained for the RNP SPRA model. Further details on this F&O can be found in the station calculation for the RNP PRA Model Peer Review Resolution [71]. This does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SPR-B4	25-12	No documented basis for the SF-TK-DG-FO-STRG-TNK fragility group.	The SF-TK-DG-FO-STRG-TNK fragility group is associated with 8 different components including entries for the tank itself along with others for the DFO XFER pumps, piping etc. In this case, per response from the Robinson SPRA team, failure of the tank is assumed to impact the other components in the group. This rationale for grouping is different than what is described in section 5.3.5 of the	Document the rationale behind grouping in the SF-TK-DG-FO-STRG-TNK fragility group.	As discussed in the Basis for Significance, failure of this group is associated with multiple components due to interactions. Discussion on interaction items using the DFOST fragility as an example has been added to the bullet list in Section 5.3.5 of the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
			station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77].		[77]. This F&O is related to F&O 24-6. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SHA-E2	26-1	The site response analysis is based on three site shear wave velocity profiles which are discussed in Section 6.2.3 of the station calculation for the Geotechnical Analysis Report [83] and Section 5.1.1 of the station calculation for Seismic Plant Hazard Analysis [16], the two reports do not document a sufficient basis for establishing the three	Section 6.2.3 of the station calculation for the Geotechnical Analysis Report [83] and Section 5.1.1 of the station calculation for Seismic Plant Hazard Analysis [16] The provide the information and basis for establishing the three site profiles used to perform the site response analysis. New shear wave velocity data was gathered at the site to aid in defining the site profile. The station calculation for the Geotechnical Analysis Report [83] notes that it is “normal practice” to	Revise Section 6.2.3 of the station calculation for the Geotechnical Analysis Report [83] and Section 5.1.1 of the station calculation for Seismic Plant Hazard Analysis [16] to improve the explanation and basis for establishing the three base case shear wave velocity profiles and the weights assigned to each one.	**Basis for Establishing Three Shear Wave Velocity Profiles** As discussed in Section 5.2.1 of the station calculation for Seismic Plant Hazard Analysis [16] a site stratigraphic framework was identified that included an upper layer with sand and clay/silt layers of variable consistency, a clay layer with typically hard consistency, and a lower layer of interbedded sand and clay layers of variable thickness and variable consistencies. Rock was identified

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
		base case profiles and the weights assigned to each one.	develop three site profiles with a reference to EPRI (2012, the SPID) [2], however the station calculation for the Geotechnical Analysis Report [83] does not sufficiently explain the role of the new data and why it leads to the three profiles selected. For example, is the new data reflective of profile epistemic uncertainty or lateral variability, particularly for profile A and B? The station calculation for Seismic Plant Hazard Analysis [16] indicates that the relative weights assigned to each profile are primarily based on the available data; however the PSHA does not sufficiently explain how the position of the data was considered in establishing the profile weights.		by a deep boring as being approximately 400 feet below the ground surface. Nine Spectral Analysis of Surface Wave (SASW) lines were performed at locations on all sides of the Plant Area, and Suspension PS seismic velocity logging to measure the shear wave velocity (V_s) was performed in Borings B-1A (upper 120 feet) and B-1B (full depth, to rock). All the V_s data showed a distinct increase in V_s in the hard clay layer, and the V_s in material above the hard clay was generally similar at all points. The significant difference was the indicated presence of a low V_s layer under the hard clay (velocity inversion) in 4 of the data sets, but not in others. Thus, a single site profile with variation represented by upper and lower bound values to represent uncertainty was not reasonable. Three separate Vs profiles were developed, one to represent areas without the velocity inversion (Profile C) and two to

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
					represent areas with the velocity inversion (Profiles A and B). The separate Profiles A and B were based on the observation of differences in the velocities observed between the P-S suspension log data from Boring B-1B (basis for Profile A) compared to those obtained for SASW Arrays 6, 7, and 8 (basis for Profile B). Figure GAR-28A compares the shear wave velocity profiles interpreted for the three SASW arrays used to develop B to the Boring B-1B P-S suspension log shear wave velocity profile of Profile A. All four profiles show the same general characteristics of the hard clay layer underlain by a layer of lower velocity. However, the variability in velocity among the SASW profiles is small and none of the three captures the strength of the velocity inversion seen in the P-S suspension log data. The assessment was made that the P-S suspension log provides a more detailed picture of the variation of velocity with depth than

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
					does the deeper portion of the SASW lines. This is indicated by the decrease in detail in the interpreted SASW profiles with increasing depth. Discussions with technical reviewers also indicated a preference for the use of P-S suspension log data compared to SASW to define a detailed velocity profile. On this basis it was judged appropriate to capture epistemic uncertainty in the strength of the velocity inversion beneath the hard clay with two profiles, one based on the detailed P-S suspension log data that show a strong inversion and one based on the SASW data that show a weaker inversion. **Basis for Selecting Weights for the Three Site Velocity Profiles** Section 5.1.1 of the station calculation for the Final Seismic Analysis Report [16] notes that the three velocity profiles were developed from geophysical investigations conducted around the periphery of the Category 1 structures (plant protected area). Information on V_s within the plant

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					protected area was limited to downhole velocity testing in three borings in the area of the ISFSI. Those borings did not extend through the hard clay layer. Previous borings in 1965 by Dames and Moore generally stopped within the hard clay layer and did not include testing for V_s. Because of the absence of definitive Vs data in the plant protected area, the presence or absence of a velocity inversion in the plant protected area could not be determined. Equal weight was assigned to the condition of a velocity inversion existing or not existing. Review of potentially useful data within the plant protected area has identified four pieces of information relevant to the velocity inversion presence or absence that had not been previously evaluated. These are the records from drilling and installation of four deep wells for plant water supply. Three deep wells were installed within the protected area in 1968 (Deep Wells

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					A, B and C). A fourth well (Deep Well D), was installed in 2004. Records for the deep well installation from plant files provided by Duke contain, among other information, the Formation Logs of the wells. The Formation Log lists the well driller's description of different soil layers penetrated, and, in most cases, notes about drilling difficulty if greater or less than "normal" drilling. Normal well drilling practice is to observe wash water coloration and to examine sediments captured by inserting a strainer into the wash water return. In addition to the Formation Logs, natural gamma and resistivity logs and rates of drilling penetration were available for Deep Well D. The stratigraphic and drilling information on the Formation Logs for Deep Wells A, B, C and D has been transcribed to the boring log format used in reporting results of the geotechnical borings. The transcribed formation logs, along with the upper 250± of the Boring B-1B log are summarized on the

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
					station calculation for the Geotechnical Analysis Report [83]. Boring B-1B formed the basis for developing Profile A and identified the velocity inversion below Layer 2 of the Profile. Boring B-1B has a natural gamma log as does Deep Well D. Natural gamma logs provide information on the variations in soil type and are useful in stratification. In the station calculation for the Geotechnical Analysis Report [83], those two gamma logs were compared and a close similarity in the indications of clay and sand was noted. The purpose of that comparison was to include the Deep Well D information on depth to rock in the overall analysis. A further comparison is made showing the gamma log plotted on a portion of the stratigraphic boring log for boring B-1B. The gamma logs in Boring B-1B and Deep Well D are further compared for their behavior in the zone where Profile A (based on Boring B-1B) indicated the

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
					presence of the velocity inversion. The Vs log shows similar increases and decreases to the gamma logs. The stiff clay layer (Layer 2) is consistent with the depth range of higher gamma values shown on the gamma logs. The velocity inversion (Layer 3A) also begins in reasonable proximity to the start of the lower gamma readings, indicating a change in soil from a more clayey to a more sandy composition. The Formation Log for Deep Well D notes that below this change in soil type, loose sandy soils, rapid drilling advance and losses of circulation were encountered for approximately 100 feet. Those notes indicate presence of weak soils, consistent with a low V_s. From these observations, the conditions in Deep Well D are consistent with those in Boring B-1B and the possibility of the velocity inversion being present at Deep Well D is interpreted as likely. Similarly, the Formation Logs for Deep Wells A, B and C consistently indicate the presence

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
					of the stiff/hard clay zone described as Layer 2 in the velocity profiles. There is also notation in the logs for Deep Wells A and C that soils below the hard clay exhibit soft drilling over the remainder of the well bore holes; however, the log for Deep Well B does not clearly indicate such conditions. Based on the match between Deep Well D and Boring B-1B information, the formation logs for Deep Wells A, C and D show indications that the velocity inversion could be present in these wells. Because three of the four deep wells, which are on all sides of the reactor containment building, show potential for a velocity inversion, the use of an equal weighting for Profiles A and B which include a velocity inversion and Profile C which does not include a velocity inversion is reasonable. Section 5.2.1 of the station calculation for the Geotechnical Analysis Report [83] is edited to add information on the Deep Wells A, B and C. Section 6.2.3 of the

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
					Geotechnical Analysis Report [83] is edited to provide more information on the potential for having (or not having) a velocity inversion within the area of the Category 1 structures. Section 5.1.1 of the Final Seismic Analysis Report [16] is edited to further explain the selection of equal weighting for Profile C relative to Profiles A and B. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SHA-J1	26-3	The approach used to perform the SRA is different than that described in the EPRI Guidance Document (SPID) [2] Sections 4.2.3, 5.2 and 5.3 of the	The SRA is based on using seed time histories which are loosely matched to the target CMS for either high frequency or low frequency. Insufficient detail is provided to understand what is meant by loosely matched and	Revise Sections 4.2.3, 5.2 and 5.3 of the Final Seismic Analysis Report [16] to enhance the justification for performing the SRA using single time	The response to F&O 26-3 is based on work developed in the station calculation for the Final Seismic Analysis Report [16]. The calculation describes two sensitivity analyses that demonstrate that the approach of using a single time

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
		Seismic Analysis Report (PSHA) [16] describe how time histories are selected and conditioned to the target conditional mean spectra (CMS). While 30 time histories are selected for use, these are used one-by-one with each of the randomized soil profiles to perform the SRA. Insufficient detail is provided to understand whether the conditioning process or the use of one time history per randomized soil profile impacts the assessment of site response amplification factors.	more importantly whether this impacts the resulting SRA amplification factors at any given input loading level. The text states that this was done to account for the natural variability in the frequency content of ground motions; however since each SRA run (one randomized profile, one randomized G/Gmax and damping) uses only one loosely matched time history, the approach taken would appear to introduce ground motion amplitude (or strain) variability into the SRA process beyond what is intended by the EPRI guidance (SPID) [2].	histories per randomized soil profile, and that loosely conditioning the time history to the CMS is appropriate. The response to the peer review questions related to the adequacy of the time history approach for SRA adequately addressed this issue; appropriate text from these responses forms the input for the revision to Section 5.3.1of the station calculation for Seismic Plant Hazard Analysis [16]	history selected from the set of 30 time histories for each site response analysis with one of the 60 randomized dynamic properties profiles produces both consistent mean results and consistent variability results. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SHA-J1	26-4	In Section 5.3.1 of the station calculation for Seismic Plant Hazard Analysis [16]: "The computed effective shear strains in the soil layers were examined	Figures 5-52 to 5-57 of the station calculation for Seismic Plant Hazard Analysis [16] show the shear strain computed in the SRA. There are shear strain values exceeding 1% especially for profiles A and B and at higher loading levels. Using the equivalent	Revise Section 5.3.1 of the station calculation for Seismic Plant Hazard Analysis [16] to justify the level of shear strain that is considered acceptable when using the EL	Station calculation for the Final Seismic Analysis Report [16] was prepared to review recent work regarding when results of equivalent linear (EL) 1-D site response may be biased compared to non-linear (NL) analyses. The results of the calculation show that

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
		and found to be generally less than 1 percent at the higher loading levels. Therefore, the use of the equivalent linear approach is considered appropriate. It should be noted that at the highest loading levels, the imposition of a minimum site amplification of 0.5 means that the computed site amplification values at high strains are often not used in developing the soil hazard." The level of shear strain that is considered acceptable when using equivalent linear approach should be justified. Using the results beyond the acceptable strain level to develop the median and standard deviation site amplification needs to be justified as well.	linear (EL) approach out of the range applicability at any point in the soil profile can impact the reliability of the site amplifications and therefore the seismic hazard calculated for the control point.	approach. The response to the peer review question related to shear strain limits adequately addressed this issue; appropriate text from this response forms the input for the revision to Section 5.3.1.	the EL site response analyses for the RNP provide an acceptable representation of site amplification for the purposes of risk quantification at the RNP. Information from the calculation is incorporated into the station calculation for the Final Seismic Analysis Report [16]. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
SR C-SHA-J2	26-5	Figure 5-29 of the station calculation for Seismic Plant Hazard Analysis [16] shows weighted mean amplification functions for the full column site profile for elevation 226 ft at different hazard levels. These functions represent the combinations of mean amplifications for the six sets of dynamic properties. It will be useful to provide figures showing the mean and standard deviation site amplification factor for each of the six alternative characterizations of the RNP site profile (3 base case velocity profiles and 2 alternative dynamic material properties) for selected ground motion levels.	To better understand and examine the effect of different sets of dynamic properties used in the site response analysis on the seismic hazard, site amplification functions for each of the alternative characterizations could be illustrated.	Revise Section 5.5.1 of the station calculation for Seismic Plant Hazard Analysis [16] to provide figures showing the mean site amplification factor for each of the six alternative characterizations of the RNP site profile (3 base case velocity profiles and 2 alternative dynamic material properties) for selected ground motion levels. The response to the peer review question related to site profiles adequately addressed this issue; appropriate text from this response forms the input for the revision to Section 5.5.1.	Mean amplification functions for the six site response analysis cases were computed and plotted in the station calculation for the Final Seismic Analysis Report [16]. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
SR C-SHA-J2	26-6	Section 5.8.2 of the station calculation for Seismic Plant Hazard Analysis [16] indicates that the vertical to horizontal (V/H) response spectral ratios described in Section 5.6 and used for GMRS and FIRS at elevations 226 ft and 216 ft, respectively were also used to develop a vertical FIRS for the base of reactor piles. However, review of tables 5-19 and 5-20 which provide the GMRS and FIRS at elevations 226 ft and 216 ft, respectively and table 5-38 which provides the FIRS at the base of reactor piles suggests that the V/H ratios developed in Section 5.6 were not used for the base of reactor piles.	The base of reactor piles located in the hard clay layer which is stiffer than the soil condition at elevations 226 ft and 216 ft. Therefore, the V/H ratios developed and use for soil condition may not be applicable to the stiffer clay. The horizontal and vertical FIRS provided in table 5-38 of the Seismic Analysis Report (PSHA) [16] indicate that different V/H ratios were used for the base of reactor piles. The use of V/H ratios other than those developed in Section 5.6 should be discussed in Section 5.8.2 of the station calculation for Seismic Plant Hazard Analysis [16]	Revise Section 5.8.2 of the station calculation for Seismic Plant Hazard Analysis [16] to indicate that different V/H ratios were applied for the vertical SCOR FIRS at the base of the piles. The response to the peer review question provides appropriate text for revising Section 5.8.2.	The text of the third paragraph of Section 5.8.2 of the station calculation for Seismic Plant Hazard Analysis [16] is modified as follows: "The soil hazard curves are then used to compute a horizontal SCOR FIRS for the base of the reactor building piles, which is smoothed and extended to cover the frequency range of 0.1 to 100 Hz using the approach described above in Section 5.5. Because the piles are founded in the hard clay, which is stiffer than the average soil, the vertical to horizontal (V/H) response spectral ratios described in Section 5.6 may be too low at low frequency. Therefore, the V/H ratios for the base of the piles are calculated using both the envelope spectral ratios from Figure 5-34 and the ratios developed by McGuire et al. (2001) for CEUS hard rock conditions of PGA in the 0.2 to 0.5 g range interpolated to the 3 frequency values at which horizontal FIRS are computed. These values and the envelope from Figure 5-34 are then enveloped and used to develop a vertical FIRS for the elevation

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
					159.2 ft. control point. The SCOR FIRS are tabulated in Table 5-38 and illustrated on Figure 5-44." Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SFR-B3	28-1	The damping value (2% critical) used in deriving the Rayleigh damping coefficients for use in the nonlinear analysis of the Class III TB in the station calculation for the Seismic Fragility of Pounding between the Class III Turbine Bldg. and the Reactor Aux. Bldg. by Nonlinear Analysis [68] may be too conservative.	A damping value of 2% critical is used in deriving the Rayleigh damping coefficients for use in the nonlinear analysis of the station calculation for the Seismic Fragility of Pounding between the Class III Turbine Bldg. and the Reactor Aux. Bldg. by Nonlinear Analysis [68] While this value may be appropriate for light steel braced structures, it may not be realistic for a structure such as the Class III TB. The peer review team is aware that damping values corresponding to Response Level 1 as defined in ASCE 4 [59] are to be used in performing nonlinear dynamic	Perform a sensitivity analysis to determine if use of a more realistic damping (e.g., 4%) would significantly lower the plant risk (CDF and LERF). Alternatively, justify the use of 2% damping as being realistic for this response analysis.	The Fragility vendor performed a sensitivity analysis using the proposed 4% damping to see if it would significantly lower the plant risk (CDF and LERF). The changes to the fragility were less than 5% as a median capacity and were relatively minor. The impact on CDF and LERF would be non consequential. Further details on this F&O can be found in the station calculation for the RNP PRA Model Peer Review Resolution [71].

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
			analysis. However, for the case of the TB Class III, additional factors need to be considered in selecting an appropriate damping value for the structure. Though the steel framing was designed as friction-type connections, it is not clear they would remain as such during the GMRS. The forces in the steel members need to be evaluated to determine the connections would slip resulting in a bearing-type connection. For the latter, 4% damping would be appropriate for Response Level 1. In addition, consideration should be given to the deformations of the non-structural elements and cracking of the mezzanine slabs caused by the pounding between TB Class III and RAB.		Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
SR C-SFR-E3	28-2	A refined fragility evaluation was not performed for SSLOCA though this item is one of the top risk contributors to CDF and LERF.	SSLOCA has been assigned a representative fragility of 0.1g HCLPF corresponding to the SSE level. Given that SSLOCA appears as one of the top contributors to SCDF as reported in the station calculation for the Robinson Seismic Probabilistic Risk Assessment Quantification Notebook [13] (FV of 5.05%) and SLERF (FV of 9.1%), a refined fragility would be required for this component. In response to a question, RNP stated that a separate risk reduction project has been initiated for the site, and as part of that, walkdowns were performed for SSLOCA during the October 2018 refueling outage. That work should result in an improved HCLPF for SSLOCA, and along with potential modifications.	Update the fragility evaluation for SSLOCA incorporating the results of the on-going risk reduction program at RNP.	Following the original walkdowns of the items on the seismic equipment list (SEL), Duke conducted supplemental walkdowns of piping and tubing whose failure could lead to Small Small Loss of Coolant Accident (SSLOCA) in order to generate a more plant-specific fragility for SSLOCA. These walkdowns are summarized in the station calculation for the Seismic Capacity Walkdown Report [53]. All of the piping/tubing was judged to have High seismic capacity. This capacity ranking translates into the same fragility information recommended for the RNP safety-related piping determined to have high seismic capacities by the EPRI SPRAIG [11]and considered to be appropriate as a more refined fragility for the SSLOCA-related piping and tubing in the station calculation for the Robinson Representative Fragilities Overview [61]. This does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					The response to this finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SFR-E3	28-3	The fragility of the Class III TB was evaluated for the pounding between TB and RAB in the station calculation for the Seismic Fragility of Pounding between the Class III Turbine Bldg. and the Reactor Aux. Bldg. by Nonlinear Analysis [68] Per Section 6 (Methodology), the pounding effects were determined based on imposing the relative displacement between the RAB and Class III TB. Such an approach could potentially	Though the Class III TB Fragility (Pounding effects) appears as the top CDF/LERF contributor (FV > 40%) per the station calculation for the Robinson Seismic Probabilistic Risk Assessment Quantification Notebook [13], no sensitivity analysis could be found in the notebook. The fragility evaluation of the Class III TB needs to be based on realistic evaluation of the pounding effects of the Reactor Auxiliary Building and TB Class III. It is seen, however, that the impact between the TB Pedestal and the TB Structure was modeled using impact elements and energy dissipators.	Perform a more realistic evaluation of the TB III and RAB pounding effects. For example, the impact forces on TB III resulting from the RAB-TB III pounding could be determined from a simplified response analysis model of the RAB and TB III and incorporation of impact elements and energy dissipators. The impact forces could then be applied to a static model of the TB III to evaluate the fragility.	Class III TB and RAB pounding interactions were assessed to have a relatively low seismic fragility for the TB Class III. The effects of this pounding affected several SSCs and resulted in being a significant risk contributor for the Robinson SPRA. The peer reviewers noted that there was a potential this fragility might be conservatively biased and further noted that refinements may be possible based on their past experiences with other SPRAs. However, the specifics of the impact mechanism for the Robinson Class III TB and RAB are different from other more typical building impacts observed in other SPRAs and Duke has solid technical reasons for concluding

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
		overestimate the impact forces (and thus underestimate the capacity of the structure) as it does not take credit for the energy dissipation of the impacting surfaces. The industry practice for evaluation of pounding between adjacent structures during a seismic event is based on modeling the impact surfaces with appropriate stiffness (Hertz Contact Law) and non-linear damping properties.			that more detailed analyses along the lines suggested by the peer review team would result in minimal changes to the fragility and the resulting risk. The energy dissipation at the impact point does not offer significant protection to the vulnerable diaphragm. While model refinement can increase the precision of the fragility, the resulting slightly modified fragility will not change the risk conclusions. Further details on this F&O can be found in the station calculation for the RNP PRA Model Peer Review Resolution [71] Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
SR C-SFR-F2	28-4	Several areas were identified throughout the fragility analysis documentation which required corrections.	The following items require correction to the documents as indicated: 1. Section 7.0 of the station calculation for Seismic Fragility Evaluation Notebook [82] should be expanded to describe all three failure modes, since they are included in the logic model, and to highlight that the failure mode for liquefaction-induced settlement controls over the other failure modes. 2. Section 5.1 (5th bullet) of the station calculation for Seismic Fragility Evaluation Notebook [82] should be corrected to reflect that the vertical ISRS have peaks up to 20 Hz. 3. Revise the station calculation for the Robinson Nuclear Power Plant Relay Contact Chatter Analysis [72] to provide additional justification for why the vibration isolators at the EDG control panels are ineffective. Also, revise the document to update references to SQURTS reports rather than just the test data summary sheets.	The various items listed in the 'Basis' column require corrections as noted.	1. Descriptions of the other two dam failure modes have been added to Section 7 of the station calculation for Seismic Fragility Evaluation Notebook [82] Emphasis on the controlling failure mode being liquefaction-induced failure has been added . 2. Section 5.1 of the station calculation for the Seismic Fragility Evaluation Notebook [82] has been updated to indicate that the Turbine Building Class III vertical ISRS have peaks at frequencies up to 20 Hz. 3. Section 8.1.1.2 and section 11, references 2 to 4 of the station calculation for the Robinson Nuclear Power Plant Relay Contact Chatter Analysis [72] has been updated to document the vibration isolators and updated SQURTs reports. 4. Section 4.9.1.1 of the station calculation for the Robinson Representative Fragilities Overview [61] has been updated

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
			4. Revise the station calculation for the Robinson Representative Fragilities Overview [61] to reflect justification for the use of 0.4 composite variability for relays 5. Update the station calculation for the Seismic Capacity Walkdown Report [53] for EDG room exhaust fans and coolant expansion tanks as discussed in response to the peer review questions. 6. Update walkdown report for SI-870B to supplement walkdown judgment that interaction between the valve and the handrail of the platform is not a credible interaction concern –Also, confirm that the platform is not a SI concern for any other equipment in the BIT room. 7. Walkdown SEWS [53] for the MCR ceiling and the fragility analysis within the station calculation for Representative Fragilities should be updated to confirm the open item with respect to anchorage of light fixtures. 8. Track completion of WO 13316743 for the CST in order to clear the unverified assumption in		to document the basis for variabilities included in the relay representative fragilities. 5. SEWS for Fans HVE-17 and HVE-18 and Tanks DG-A-EXP-TK and DG-B-EXP-TK have been updated to document the basis for adequate anchorage and to re-rank the components as High in the station calculation for the Seismic Capacity Walkdown Report [53]. 6. A supplemental review of BIT room was performed and the SEWs forms were updated to enhance the documentation for the observations noted in the F&O. The station calculation for the Seismic Capacity Walkdown Report [53] was updated accordingly. 7. The SEWs for the MCR ceiling in the station calculation for the Seismic Capacity Walkdown Report [53] was updated accordingly as was the corresponding fragility

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
			the station calculation for the Seismic Fragility Evaluation of the CST [85]. 9. The basis for screening out the displacement-based failure mode between the RAB and RCB should be documented. 10. The basis for screening out FLEX haul path should be documented. 11. The TB ISRS is conservatively biased and additional documentation should be provided to justify its impact. 12. The GMRS is used as the input motion for SSCs housed within the intake structure and for the vertical direction, FIRS is used as the vertical input motion. Since this embedded structure is expected to move with the soil, additional clarification should be provided on why the soil amplification in the vertical direction would not affect the vertical ISRS. 13. Preliminary studies were conducted to determine the effect of pile foundation modeling on the		contained in the station calculation for the Robinson Representative Fragilities Overview [61]. 8. Work Order 13316743 for the CST was completed and closed on 4 June 2019. The station calculation for the Seismic Fragility Evaluation of the CST [85] has been revised accordingly to reflect the completion of this work order. 9. The station calculation for the Response Analysis Notebook [55] has been updated to justify the screening out of failure modes associated with the impact between the RCB and RAB. 10. Section 5.7 of the station calculation of the Representative Fragilities [61] has been added to document screening of the FLEX haul paths.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
			response of the RCB and RAB, but it is not documented. Additional basis should be provided to better characterize the final SSI modeling approach adopted.		11. Section 5.1 of the station calculation for the Turbine Building Class III seismic response analysis [75] has been updated to discuss the impact of disregarding horizontal pile flexibility. As indicated, inclusion of pile flexibility could affect gantry crane response at certain frequency ranges but not at others. Disregarding pile horizontal flexibility is considered to be sufficiently accurate. 12. Section 3 of the station calculation for seismic response analysis [87] has been updated to provide justification for using the soil column outcrop vertical Foundation Input Response Spectrum as vertical input to SSCs in the Intake Structure. As noted, vertical FIRS are available at Elevation 159 ft. Vertical response of the Intake Structure occurs at very high frequencies. At these frequencies, the vertical FIRS at

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
					Elevation 159 ft exceeds the vertical GMRS at Elevation 226 ft. Significant amplification of motion between Elevation 159 ft and the Intake Structure Foundation at Elevation 172 ft is considered to be very unlikely. Use of the vertical FIRS is considered to be appropriate. 13. 5.7 Section 4.1.6 of the station calculation for the Response Analysis Notebook [55] is added to identify conclusions obtained by sensitivity studies investigating modeling of the pile-founded structures. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
SR C-SFR-A2	28-5	The assumption that equipment in different elevations of the same building as having different seismic demand, used in determining fragility group correlation, may not be appropriate for RNP.	The 2nd bullet of Section 5.3.5 of the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77] states: "Located on the same elevation in the building: In-structure response spectrum at the given elevation in the building is expected to be essentially the same for all equipment located at that elevation." Per this criterion, similar equipment with similar orientation at different elevations of the same building would be assumed to be not correlated. Given that the RNP Structures are dominated by the pile-soil-structure mode, the ISRS appear to be similar at different elevations within a given structure in the horizontal direction. In response to a peer review team question, RNP provided the results of an assessment of the top 25 contributors to CDF and LERF. The assessment concluded that none of the 25 components evaluated	Modify the correlation criteria specified in Section 5.3.5 of the station calculation for the H.B. Robinson Seismic Probabilistic Risk Assessment Model Notebook [77] to reflect the seismic response of RNP structures. Document the results of the assessment made for the top contributors would not be adversely impacted by the correlation assumption.	The model does not assume correlation between similar equipment with similar orientation on different elevations of a structure. As discussed in the finding, this may be non-conservative as the ISRS are similar at different elevations within a given structure in the horizontal direction due to the pile-soil-structure failure. However, assuming complete correlation between equipment on separate elevations would be overly conservative. The impact this assumption has on the model was analyzed by reviewing the top contributors. All SSCs in the top 25 contributors for CDF and/or LERF based on Fussel-Vesely (FV) or with an FV greater than 5.0E-03 were reviewed for potential correlations with SSCs at different elevations in multi-story buildings expected that are expected to undergo similar seismic responses on the different elevations. This correlation assumption in not applicable to SSCs such as structures, tanks, cranes, and

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
			would be impacted by the multi-elevation correlation criterion.		piping. Additionally, liquefaction-induced settlement and lateral spreading failures are treated separately, so this assumption is not applicable to these hazards. The remaining SSCs include conduit in the reactor auxiliary building (RAB), conduit in the containment building (CB), vacuum relief valves, air handlers and coolers for the AFW pump room, and Barksdale and Dwyer Instruments relays; these SSCs and their FVs were reviewed and are included in the F&O Resolution Notebook [71]. The impact of the correlation assumption on the remaining top contributors is dispositioned for RAB Conduit, CB Conduit, Vacuum Relief Valves, AFW Room Air Handlers and Coolers, Barksdale Relays and Dwyer Instruments Relays. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard with NRC

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					Clarification. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SFR-C1/C-SFR-E1	29-1	Insufficient justification provided to support aggressive capacity used for fragility evaluation of high ranked valves based on Robinson site standard document.	Valves ranked as high capacity by walkdown were assigned a fragility using a capacity extracted from a Robinson site standard document for Pipe Stress Analysis Procedure which requires the following: 'Valves must meet allowable valve accelerations of 2 g vertical and 3 g for each horizontal direction for all dynamic conditions unless other values are approved by the qualification.' The peer review team expressed some concern that not all plant valves may meet these seismic requirements. It is possible that exceptions were taken for certain valves and lower capacities were adequate for design basis (for example, valves with cast iron yokes).	Perform a review of valves ranked high to confirm that all have a capacity of at least 3g in each horizontal direction and 2g vertical.	For the safety-related valves, Duke has confirmed that the majority of them were purchased and installed to meet the generic capacities of 3.0 g horizontal and 2.0 g at the Robinson site, with the exception of FCV-6416 (SDAFW Pump Discharge Flow Control Valve). Therefore, it is appropriate to use the high generic capacities as an input to development of the safety-related valve fragility for all practical purposes. For FCV-6416, the walkdown report notes that the operator height of valve is 47 inches, exceeding the GIP guidelines [50] and the yoke is laterally supported by a structural angle frame bolted to both sides of the operator. Because of this unique support configuration, the valve fragility analysis identified yoke failure as the controlling failure mode in the station calculation for the Robinson Representative Fragilities Overview [61] and developed the

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
					corresponding HCLPF capacity of 0.67g, NOT based on the generic valve capacities. For the non-safety valves, Systems, Structures, and Components (SSCs) included in the station calculation for the Seismic Equipment List [64] have been first reviewed to identify relevant valves. This review has found that there are a total of 40 non-safety valves. Each valve is re-assessed with respect to the appropriateness of its fragility basis. The objective of this assessment is to identify any non-safety valve case where the generic high capacity only applicable to the safety-related or Seismic Category I valve was inappropriately used as a capacity input for fragility development and to make appropriate adjustments or corrections to the existing fragility parameters of the valve. These values can be found in the station documentation for the station calculation for the RNP PRA Model Peer Review Resolution [71]. This does not represent a change in methodology, scope, or capability as defined in Appendix 1-

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
					A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SFR-E1/C-SFR-E3	29-2	EPRI SPRAIG [11] values used for small and medium LOCA with minimal technical justification	In Robinson the station calculation for the Robinson Representative Fragilities Overview [61] limited basis is provided for the use of generic fragility values obtained from the EPRI SPRAIG [11] for small and medium LOCA (SLOCA and MLOCA, respectively). The document concludes that the use is representative and 'possibly conservative'. The Peer Review Team notes that based on current SPRA quantification results, SLOCA shows up as a risk contributor.	Provide technical basis to support the conclusion that generic fragilities from the EPRI SPRAIG [11] are appropriate for Robinson. Evaluate appropriateness of SPRAIG values wherever else they are credited in the RNP SPRA logic model (e.g., distributed systems).	Duke conducted supplemental walkdowns of piping and tubing whose failure could lead to SLOCA in order to generate a more plant-specific fragility for SLOCA. The scope of the SLOCA walkdown included piping and tubing with inside diameters of 0.35 in. to 1.5 in. The walkdown results showed that all of the piping/tubing was judged to have High seismic capacity. Because no seismic concerns were identified by the walkdown, the SPRAIG fragility for piping (rather than the SPRAIG fragility for SLOCA) is now considered to be appropriate. With this fragility, SLOCA should not be risk-significant so further cost to develop a plant-specific fragility is considered to be unwarranted. Section 4.4.2 of the station calculation for the Robinson

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
					Representative Fragilities Overview [61] has been updated to provide additional justification for the use of the SPRAIG fragility for medium loss of coolant accident (MLOCA). During the area walkdowns for Robinson, the walkdown teams did not note any specific issues for the piping that could cause a MLOCA which would invalidate the use of the EPRI SPRAIG value. The contribution to SCDF and LERF from MLOCA is negligible. As such, the use of the EPRI SPRAIG fragility for MLOCA is judged to be appropriate and the costs to develop a plant-specific fragility is considered to be unwarranted. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
SR C-SFR-E4/C-SFR-F1	29-3	Seismic fragility backup calculations not readily accessible for future updates and maintenance of the Robinson SPRA.	Backup calculations prepared for the station calculation for the Robinson Representative Fragilities Overview [61] are not documented. Methods are described and sample calculations are provided; however, detailed calculations and / or spreadsheets were not provided the peer review team for review. For example, anchorage calculations supporting SRT walkdown judgments, CDFM relay fragility calculations, and other calculation results for SEL equipment (e.g., valves) in the table in Appendix A.	In order to satisfy the intent of this supporting requirement, backup calculations prepared for the station calculation for the Robinson Representative Fragilities Overview [61] should be documented in order to facilitate future PRA applications, upgrades, and peer reviews.	Excel files containing representative fragility calculations have been attached to the station calculation for the Robinson Representative Fragilities Overview [61] These files are identified in Section 2.3 of the station calculation for the representative fragility report [61]. This does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this finding meets the requirements of NTTF 2.1 Seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SFR-D5/C-SFR-F1	29-4	Fire protection piping with vulnerable piping joints dispositioned by walkdown SRT with limited technical basis.	Walkdown SEWS for fire protection piping located in the second floor of the RAB correctly identified the presence of Victaulic couplings. Based on PRT walkdown, however, the SRT did not document other issues existing in the field including what appeared to be loose pipe straps and hard contact of this piping with a primary water line. Further, the SRT dispositioned the	Revise the walkdown SEWS for the fire protection piping on the second floor of the RAB (Walkdown SEWS RAB-FLOOR 2-ALL-FIRE PIPING) to indicate that the piping is dry and not a flooding concern. Similarly, update fire and flood assessment	Subsequent review determined that all fire protection piping on the second floor of the RAB is dry and therefore flooding is not a concern. The SEWS for RAB-FLOOR 2-ALL-FIRE PIPING has been revised accordingly (Attachment 2, Station

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
			presence of Victaulic couplings as 'OK' by judgment and that the line could leak a little. More technical basis should have been provided to support SRT judgment. It is noted that subsequent input from the RNP fragility team provided during the peer review indicates that the subject fire line is dry; therefore, flooding concerns are not credible.	documentation as appropriate.	calculation for the Seismic Capacity Walkdown Report [53] This does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this finding meets the requirements of NTTF 2.1 Seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SFR-D7	29-5	During Peer Review Team walkdown of the Robinson site, a few instances were identified where the walkdown SEWS either did not appear to capture all potential seismic interactions or judgment of the walkdown SRT to document disposition of potentially credible seismic interactions was not provided.	During Peer Review Team walkdown of a sample of SEL components at the Robinson site, the following seismic interactions were either not documented or adequately dispositioned as non-credible concerns on the walkdown SEWS: (1) Disposition of potential seismic interaction between MS line PORV (RV1-1) and structural steel as not a credible concern. The walkdown SEWS includes a photo showing proximity of the valve to the steel, but no discussion provided. Earthquake experience indicates	Since Peer Review Team walkdown only reviewed a limited sample of SEL components, walkdown SEWS documented within the station calculation for the Seismic Capacity Walkdown Report [53] should be reviewed on an expanded sampling basis to confirm whether there are any other instances where credible seismic interactions were	All of the specifics identified in the F&O were re-verified and the SEWs were updated accordingly. There were no issues found from the verification walkdown that would have invalidated the SPRA model or any of it's conclusions. Based on the sampling conducted by the extended condition walkdown and a cross check of the SEWS that document the original walkdowns, it is concluded that the walkdowns were conducted properly with respect to evaluating and documenting potential interactions.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
			that valves have failed due to impact with adjacent structural steel. (2) Flexibility of piping attached to CST for uplift displacements at tank failure and disposition that potential interaction of the gasifier tank is not credible (3) No basis provided on the SEWS for SRT conclusion that SDAFW-PMP contains no credible soft targets. (4) CO2 tanks not identified as a possible seismic interaction with MCC-5 on the walkdown SEWS (5) It does not appear as if walkdown SEWS for all DS DG equipment (e.g., DS DG Batteries) indicate the potential interaction with the gantry crane	either not noted or properly dispositioned. Additionally, walkdown documentation should be updated as appropriate for items identified in the basis portion of this F&O.	Further details on this F&O can be found in the station calculation for the RNP PRA Model Peer Review Resolution [71]. This does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this finding meets the requirements of NTTF 2.1 Seismic and Capability Category II of the Standard with NRC clarification. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SFR-F2	29-6	Potential differential displacement of piping in the BIT room area of the RAB was not explicitly evaluated by walkdown inspection.	The piping in the BIT room area of the RAB was observed during PRT walkdown of the Robinson plant and it appeared as if piping traversing between the containment and the RAB had somewhat limited flexibility. Based on responses from the RNP fragility	Provide and document disposition that piping in the BIT room area is bounded by the review of differential displacements of piping systems exiting containment documented within the station	An area walkdown of the BIT room was performed to determine if piping crossing between the Reactor Containment Building (RCB) and RAB has sufficient flexibility to accommodate relative building displacements. The walkdown found two instances of

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
		(This F&O originated from SR C-SFR-D7)	team, walkdown SRTs did not access the BIT room for purposes of performing area walkdown inspections for distributed systems.	calculation for the Seismic Fragility Evaluation of the RCB [94].	piping crossing between structures, several instances of tubing spanning the gap, and flexible conduit powering the motors to Valves SI-870 A and B. The containment purge line was previously reviewed in Seismic Fragility Evaluation of the RCB [94]. The walkdown confirmed that this line should have a displacement capacity greater than that of the RCB piles. The hydrogen purge line was observed to have several bends with no attachment points near the RCB-RAB interface, and should also have a displacement capacity greater than that of the RCB piles. The tubing and conduit were observed to have sufficient flexibility providing displacement capacities greater than that of the RCB piles. A new area piping SEWS, RAB-BIT ROOM-PIPING, has been developed to address the effect of relative building displacements on piping in the BIT room (Attachment 2, Station calculation of the Seismic Capacity Walkdown Report [53]). Additional details on the BIT room piping walkdown may be obtained from this SEWS.

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

SR	F&O	Description	Basis	Suggested Resolution	Disposition
					Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard with NRC Clarification
SR C-SFR-F2	29-7	Walkdown team composition and qualifications not specified for Operator Pathway Walkdowns	Seismic Review Team resumes are provided as appropriate for SEL equipment walkdowns documented within Appendix C of the station calculation for the Seismic Capacity Walkdown Report [53] ; however, team composition and qualifications for Operator Pathway walkdowns documented within Calculation SEL Notebook [64], Appendix C are not provided.	Provide team composition and qualifications for participants other than those listed in the station calculation for the Seismic Capacity Walkdown Report [53]	Résumés for Jensen Hughes personnel participating in the seismic walkdowns (John Reddington and Rick Anoba) have been added to Appendix C of the station calculation for the Seismic Capacity Walkdown Report [53] Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
SR C-SFR-B5	30-1	The technical basis for using in-layer motions (defined at depth corresponding to the bottom of piles) in the vertical direction is limited to justify the compatibility between the SSI modeling approach adopted.	In the vertical response analysis, the skin resistance offered by the sand layers is neglected compared to the tip resistance offered by the underlying clay layer and the piles are interacting with the soil only at the bottom. Since the approach does not credit any soil resistance along the length except at the bottom, using in-layer motion at depth an input to the vertical response analysis seemed not appropriate for this configuration. Based on the discussions with the SPRA team during the onsite review, it was judged that the in-layer motion at depth was the appropriate motion for this configuration. The piles are anchored into the clay layer for about 11' which in turn provides the fixity such that the pile bottom experiences the same input motion as the soil column at depth irrespective of the pile soil interaction above the clay layer.	The additional discussions with the SPRA team during the on-site review as such explained in the basis of this finding should be included in the corresponding SSI response analysis calculation.	Station calculations RAB SSI Analysis [95] and RCB SSI Analysis [96] have been updated to include justification for using the vertical in-layer motion at Elevation 159.2 ft as vertical input to the seismic response analyses of the RCB and RAB. As noted, the SSI models represent the piles as connected vertical to the soil within the clay layer. The piles are not connected to the soil vertically at any other point because very little side friction resistance during the installation process was observed and similar behavior would be expected in response to seismic vertical motion. Since the vertical force transfer is primarily or completely through the piles, the vertical foundation level is deemed to be the bottom of the pile model where the force enters the structure. The control point for vertical analyses is set at the same elevation at the bottom of pile. The bottom of the piles will move in response to the movement of the hard clay layer and the movement of the hard clay layer will be affected by the response of the soil above regardless of whether or not the piles are connected to the soil

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
					above this point in the model. Therefore an in-layer (or in-column) motion at Elevation 159.2 ft. is proper for use. Additional details may be obtained from station calculations RAB SSI Analysis [95] and RCB SSI Analysis [96]. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.
SR C-SFR-E3	30-2	The collapse of the TB-Class III building has been included in the S-PRA model as two different fragility groups (SF-TB-CLASS-3-POUND and SF-TB-CLASS-3).	The fragility group SF-TB-CLASS-3 represents the collapse due to excessive first story drifts and SF-TB-CLASS-3-POUND represents the collapse due to pounding with the adjacent RAB. The consequence of both these fragility groups is the collapse of the turbine building but their governing failure modes (excessive drift and pounding due to relative	Assess the degree of correlation between these two failure modes and calculate the combined fragility. If the failure modes are assessed to be completely correlated, then model the fragility group with the lowest fragility.	As stated in the Finding, some degree of correlation exists between the Class 3 Turbine Building shaking fragility and pounding fragility. The composite fragilities for two bounding cases in terms were compared for correlation: zero correlation (independent) and perfect correlation

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			displacement) are not completely independent of each other.		As the pounding fragility is much larger than the shaking fragility over the hazard range of interest, there is virtually no difference between the pounding fragility curve and the perfect correlation composite fragility curve. The composite fragility curve for the zero correlation case is slightly more conservative than the perfect correlation case, but the degree of conservatism is so small that they are equal for all practical purposes. While realistically these two failure modes should be at least partially correlated, any partial correlation curves are bounded by these two cases; and therefore, partial correlation possibilities are not an important consideration. This justifies the current modeling assumption of independence between the pounding and shaking fragilities in the as-built and as-operated model. This does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	**F&O**	**Description**	**Basis**	**Suggested Resolution**	**Disposition**
					and is not considered an upgrade. The response to this finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.
Focused Scope Peer Review Findings and Resolutions					
SR C-SFR-E3	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			[redacted]	[redacted]	[redacted]

Table A-2: Summary of Finding F&Os and Disposition Status					
SR	F&O	Description	Basis	Suggested Resolution	Disposition
			[redacted]		
SR C-SFR-F3	2-2	SFR-F3 requires that sources of model uncertainty and related assumptions associated with the fragility analysis be appropriately documented. There are several sources of uncertainty that should be acknowledged and discussed further.	As noted in Section 6.4.1 of the station calculation for seismic fragility evaluation of the Robinson Dam [67], "the confidence levels in Figure 13 capture the variability in the dam soil model properties and ground motion included within the LHS conducted. These are judged to be the dominant sources of uncertainties in the fragility." There are other sources of uncertainty that may affect the confidence levels, including uncertainty associated with the estimated depth of transverse cracking conditioned on crest displacement (Figure 17 of Appendix B [67]) and the estimated probability of failure conditioned on crack exposure (Figure 5). Additional discussion and evaluation of these sources is needed before concluding that the variability in soil properties and	A more thorough discussion of sources of uncertainty in estimating the fragility of Robinson Dam should be developed consistent with the verbal explanation provided by the fragility vendor during a conference call with the peer review team.	Duke agrees with the statement that the dam fragility included the aleatory uncertainty associated with the dam/foundation soil properties and the earthquake ground motions, but neglected calculating the epistemic uncertainties such as those associated with the dam soil model, the relationship between crest deformation and crack depth or in the relationship between crack exposure and probability of failure. There are several reasons that justify Duke's judgement that the aleatory uncertainties are sufficient for this fragility assessment of the Robinson Dam: 1. The aleatory uncertainty calculated from the Latin Hypercube Simulation are very large, they incorporate a large variability in the soil parameters

Table A-2: Summary of Finding F&Os and Disposition Status					
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			ground motions are the dominant sources.		and in the earthquake ground motions. 2. Epistemic uncertainly certainly exists, but the calculation of those uncertainties would entail a very large effort and a much longer schedule than is available for this Robinson SPRA. Our collective judgment of the SPRA team is that additional large effort is not warranted. Based on our experience the epistemic uncertainty will be much smaller than the calculated aleatory uncertainty for the dam. And once this epistemic uncertainty is calculated, it will be SRSSed with the larger aleatory uncertainty which will result in a composite uncertainty close to the existing aleatory uncertainty. 3. Duke performed a completely independent method of developing a seismic fragility using earthquake experience data. This independent fragility was developed at the lower end of the ground motions but the fragility overlapped the more

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					detailed FLAC analysis fragility in the 0.3 to 0.35 g part of the spectrum. These two completely independent fragility derivations resulted in nearly identical best estimate values for the fragility which provides a very strong case that the epistemic uncertainties that were not evaluated as part of the fragility derived from the FLAC analyses will not bias the best estimate results. In addition the uncertainties associated with the earthquake experience derived fragilities (these uncertainties include both aleatory and epistemic variabilities based on the use of a large number of dam performance data from a large number of large earthquakes) are smaller than the aleatory uncertainties derived from the Latin Hypercube Simulations. Which provides additional justification that these large aleatory uncertainties are sufficient for this fragility. Finally, it is Duke's judgement that even if we were to estimate some

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					additional epistemic uncertainty to add into the existing fragility, the overall results would not be expected to change appreciably. Therefore, this does not represent a change in methodology, scope, or capability as defined in Appendix 1-A of the ASME/ANS PRA Standard and is not considered an upgrade. The response to this Finding meets the requirements of NTTF 2.1 seismic and Capability Category II of the Standard. This finding is considered resolved for the purposes of NTTF 2.1.