UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 22, 2020

Mr. Ernest J. Kapopoulos, Jr.
Site Vice President
H. B. Robinson Steam Electric Plant
Duke Energy Progress, LLC
3581 West Entrance Road, RNPA01
Hartsville, SC 29550

SUBJECT: H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 – STAFF REVIEW OF SEISMIC PROBABILISTIC RISK ASSESSMENT ASSOCIATED WITH REEVALUATED SEISMIC HAZARD IMPLEMENTATION OF THE NEAR-TERM TASK FORCE RECOMMENDATION 2.1: SEISMIC (EPID NO. L-2019-JLD-0022)

Dear Mr. Kapopoulos:

The purpose of this letter is to document the staff's evaluation of the H.B. Robinson Steam Electric Plant, Unit No. 2 (Robinson), seismic probabilistic risk assessment (SPRA) which was submitted in response to Near-Term Task Force (NTTF) Recommendation 2.1 "Seismic." Based on the regulatory commitments in letter from Duke Energy Progress, LLC (Duke, the licensee) dated June 19, 2020 (ADAMS Accession No. ML20171A761) as well as the interim actions identified in the same supplement, the U.S. Nuclear Regulatory Commission (NRC) has determined that further regulatory action associated with NTTF Recommendation 2.1 "Seismic" is not warranted for Robinson.

By letter dated March 12, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12053A340), the NRC issued a request for information under Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.54(f) (hereafter referred to as the 50.54(f) letter). The request was issued as part of implementing lessons learned from the accident at the Fukushima Dai-ichi nuclear power plant. Enclosure 1 to the 50.54(f) letter requested that licensees reevaluate seismic hazards at their sites using present-day methodologies and guidance. Enclosure 1, Item (8), of the 50.54(f) letter requested that certain licensees complete a SPRA to determine if plant enhancements are warranted due to the change in the reevaluated seismic hazard compared to the site's design-basis seismic hazard.

By letter dated December 12, 2019 (ADAMS Accession Nos. ML20084P290 (public) and ML19346E204 (non-public)), supplemented by letters dated March 31, 2020 (ADAMS Accession Nos. ML20094K843 (public) and ML20092E957 (non-public)) and June 19, 2020, Duke provided its SPRA submittal in response to Enclosure 1, Item (8) of the 50.54(f) letter for Robinson. As applicable, the NRC staff assessed the licensee's implementation of the Electric Power Research Institute's Report 1025287, "Seismic Evaluation Guidance - Screening, Prioritization, and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic" (ADAMS Accession No. ML12333A170). This report was endorsed by the NRC by letter dated February 15, 2013 (ADAMS Accession No. ML12319A074). In addition, consistent with the licensee's submittal, the NRC staff utilized a reviewer checklist that is based on American Society of Mechanical Engineers (ASME)

American Nuclear Society (ANS) (RA-S Case 1 "Case for ASME/ANS Ra-Sb-2013, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (herein called the "Code Case Standard"). Use of this reviewer checklist for licensees choosing to use the Code Case Standard was described in a letter to the Nuclear Energy Institute (NEI) dated July 12, 2018 (ADAMS Accession No. ML18173A017).

The reviewer checklist for the Robinson SPRA submittal and supplements is contained in Enclosure 1 to this letter. As described below, the NRC has concluded that the Robinson SPRA submittal and supplements meet the intent of the SPID guidance and that the results and risk insights provided by the SPRA support the NRC's determination that regulatory actions associated with NTTF Recommendation 2.1 "Seismic" should be considered under NRC's backfit provisions.

BACKGROUND

The 50.54(f) letter requested, in part, that licensees reevaluate the seismic hazards at their sites using updated hazard information and current regulatory guidance and methodologies. The request for information and the subsequent NRC evaluations have been divided into two phases:

> **Phase 1**: Issue 50.54(f) letters to all operating power reactor licensees to request that they reevaluate the seismic and flooding hazards at their sites using updated seismic and flood hazard information and present-day regulatory guidance and methodologies and, if necessary, to request they perform a risk evaluation.
>
> **Phase 2**: Based upon the results of Phase 1, the NRC staff will determine whether additional regulatory actions are necessary (e.g., updating the design basis and structures, systems, and components important to safety) to provide additional protection against the updated hazards.

By letter dated March 31, 2014 (ADAMS Accession No. ML14099A204) and revised on July 17, 2015 (ADAMS Accession No. ML15201A006), Duke submitted the reevaluated seismic hazard information for Robinson. The NRC performed a staff assessment of the submittal and issued a response letter on October 19, 2015 (ADAMS Accession No. ML15280A199). The NRC's assessment concluded that Duke conducted the hazard reevaluation using present-day regulatory guidance and methodologies, appropriately characterized the site, and met the intent of the guidance for determining the reevaluated seismic hazard at Robinson.

By letter dated October 27, 2015 (ADAMS Accession No. ML15194A015), the NRC documented a determination of which licensees were to perform: (1) an SPRA; (2) limited scope evaluations; or (3) no further actions, based on, among other factors, a comparison of the reevaluated seismic hazard and the site's design-basis earthquake. As documented in that letter, Robinson was expected to complete an SPRA with an estimated completion date of March 31, 2019, which would also assess high frequency ground motion effects. By letters dated November 29, 2018 (ADAMS Accession No. ML18337A159), and October 21, 2019 (ADAMS Accession No. ML19294A028), the licensee requested to extend the SPRA submittal to October 31, 2019, and December 12, 2019, respectively. The staff responded in letters dated January 10, 2019 (ADAMS Accession No. ML19004A356), and October 28, 2019 (ADAMS Accession No. ML19296C623), respectively. In addition, Duke was expected to perform a limited-scope evaluation for the spent fuel pool (SFP). This SFP limited-scope evaluation was submitted by letter dated August 1, 2016 (ADAMS Accession No. ML16215A376). The staff provided its assessment of the Robinson SFP evaluation by letter dated September 30, 2016 (ADAMS Accession No. ML16230A535).

The completion of the NRC staff assessment for the reevaluated seismic hazard and the scheduling of Robinson SPRA submittal as described in the NRC's letter dated October 27, 2015, marked the fulfillment of the Phase 1 process for Robinson.

In its letter dated December 12, 2019, and supplements dated March 31, 2020, and June 19, 2020, Duke provided the SPRA submittal that was used for the NRC's Phase 2 decisionmaking process for Robinson. The NRC described this Phase 2 decisionmaking process in a guidance memorandum from the Director of the Division of Operating Reactor Licensing to the Director of the Office of Nuclear Reactor Regulation (NRR) dated March 2, 2020 (ADAMS Accession No. ML20043D958). This memorandum describes a Senior Management Review Panel (SMRP) consisting of NRR Division Directors that are expected to reach a screening decision for each plant submitting an SPRA. The SMRP is supported by appropriate technical staff who are responsible for consolidating relevant information and developing the recommendation for the screening decisions for consideration by the panel. In presenting recommendations to the SMRP, the supporting technical staff is expected to recommend placement of each SPRA plant into one of three groups:

1) **Group 1** includes plants for which available information indicates that further regulatory action is not warranted. For seismic hazards, Group 1 includes plants for which the mean seismic core damage frequency (SCDF) and mean seismic large early release frequency (SLERF) clearly demonstrate that a plant-specific backfit would not be warranted.

2) **Group 2** includes plants for which further regulatory action should be considered under the NRC's backfit provisions. This group may include plants with relatively large SCDF or SLERF, such that the event frequency in combination with other factors results in a risk to public health and safety for which a regulatory action is expected to provide a substantial safety enhancement.

3) **Group 3** includes plants for which further regulatory action may be needed, but for which more thorough consideration of both qualitative and quantitative risk insights is needed before determining whether a formal backfit analysis is warranted.

The evaluation performed to provide the basis for the staff's grouping recommendation to the SMRP for Robinson is described below.

EVALUATION

Upon receipt of the licensee's SPRA submittal, a technical team of NRC staff members performed a completeness review to determine if the necessary information to support Phase 2 decisionmaking had been included in the licensee's submittal. The technical team performing the review consisted of staff experts in the fields of seismic hazards, fragilities evaluations, and plant response/risk analysis. A week after the submittal, the technical team determined that sufficient information was available to perform the detailed technical review in support of the Phase 2 decisionmaking. Subsequently, the staff used the information in the submittal to determine that an immediate safety concern did not exist. NRC management agreed with the staff's determination that an immediate safety concern did not exist. Section 2.1 of Enclosure 4 of this staff assessment documents the staff's evaluation of an immediate safety concern.

As described in the 50.54(f) letter, the staff's detailed review focused on verifying the technical adequacy of the licensee's SPRA such that an appropriate level of confidence could be placed in the results and risk insights of the SPRA to support regulatory decisionmaking associated with the 50.54(f) letter. As stated in its submittal, the licensee developed and documented the SPRA to respond to Enclosure 1 of the 50.54(f) letter, Item 8(b) and Section 6.8 of the SPID. The SPRA included performance of an independent peer review against the Code Case Standard which is summarized in Appendix A of the licensee's submittal.

By letter dated July 6, 2017 (ADAMS Accession No. ML17177A446), the NRC issued a generic audit plan and entered into the audit process described in Office Instruction LIC -111, "Regulatory Audits," dated December 29, 2008 (ADAMS Accession No. ML082900195), to assist in the timely and efficient closure of activities associated with the 50.54(f) letter. The list of applicable licensees in Enclosure 1 of the July 6, 2017, letter included Duke as the licensee for the Robinson site. The staff exercised the audit process by reviewing selected licensee documents via an electronic reading room (eportal) as documented in Enclosure 3 to this letter.

During the audit process, the staff developed questions to clarify information in the licensee's submittal and to gain understanding of non-docketed information that supports the docketed SPRA submittal. The staff's clarification questions and request for supporting documents (ADAMS Accession Nos. ML19365A046, ML20009C092, ML20015A189, ML20027C204, ML20087N663 (public and ML20087N623 (non-public)), ML20085F931, and ML20086L533 (public and ML20086L226 (non-public)), were sent to the licensee to support the audit. The licensee subsequently provided those supporting documents and answers to the audit questions on the eportal, which the staff reviewed. The staff determined that the answers to the questions provided in the eportal served to confirm statements that the licensee made in its SPRA submittal and supplements.

Appendix A of the licensee's submittal also included the open SPRA finding level facts and observations (F&Os) along with the licensee's dispositions. These elements were reviewed by NRC staff in the context of the regulatory decisionmaking associated with the 50.54(f) letter. Since the licensee's internal events PRA (IEPRA) model was used as the basis for the development of the SPRA model, the NRC staff reviewed the IEPRA F&Os and the associated dispositions during the SPRA audit process to assess any potential impact on the SPRA submittal. The NRC staff identified no issues with the licensee's dispositions to these findings with respect to the SPRA submittal.

The staff's review process included the completion of the SPRA Submittal Technical Review Checklist (SPRA Checklist) contained in Enclosure 1 to this letter. As described in Enclosure 1, the SPRA Checklist is a document used to record the staff's review of licensee's SPRA submittals against the applicable guidance of the Code Case Standard, as described in the NRC letter to the NEI dated July 12, 2018. Enclosure 1 contains the staff's application of the SPRA checklist to Robinson's submittal. As documented in the checklist, the staff concluded that the Robinson SPRA meets the intent of the SPID guidance, including the documentation requirements of the Code Case Standard.

Based on the staff's review of the licensee's submittal, including the resolution of the peer review findings and completion of the SPRA checklist as described above, the NRC staff concluded that the licensee's SPRA submittal was technically acceptable to the extent necessary to support regulatory decisionmaking associated with Phase 2 of the 50.54(f) letter.

Following the staff's conclusion on the SPRA's technical adequacy, the staff confirmed the validity of its determination that an immediate safety concern did not exist.

To clarify and support its decisionmaking, the NRC staff performed a detailed technical evaluation of the insights from the Robinson SPRA and the audit information using the principles of risk-informed decisionmaking. The evaluation is documented in Enclosure 4 to this letter. Based on the information provided in the SPRA submittals and in the audit, the staff recommended increased senior management attention and communication with the licensee management.

Further, the staff used the screening criteria described in a staff memorandum dated August 29, 2017 (ADAMS Accession No. ML17146A200), titled, "Guidance for Determination of Appropriate Regulatory Action Based on Seismic Probabilistic Risk Assessment Submittals in Response to Near Term Task Force Recommendation 2.1: Seismic" to assist in determining the group in which the technical team would recommend placing Robinson to the SMRP. The criteria in the staff's guidance document includes thresholds to assist in determining whether to apply the backfit screening process described in Management Directive 8.4, "Management of Facility Specific Backfitting, Forward Fitting, Issue Finality, and Information Requests," dated September 20, 2019 (ADAMS Accession No. ML18093B087), to the SPRA submittal review. Enclosure 2 to this letter discusses the staff's evaluation.

As a part of the Phase 2 decisionmaking process for SPRAs, the NRC formed the Technical Review Board (TRB), a board of senior level NRC subject matter experts, to ensure consistency of review across the spectrum of plants that will be providing SPRA submittals. The technical review team provided the results of the Robinson review to the TRB. The TRB members assessed the information presented by the technical team and agreed that the team's evaluation should be presented to the SMRP. Subsequently, the technical review team met with the SMRP on multiple occasions to present the results of the review, including information provided by the licensee as part of the audit, and its recommendations. The SMRP members sought detailed information about the review and provided input to the technical team.

Based on the evaluation in Enclosures 2 and 4, and supported by the increased attention from the SMRP, the staff considered potential modifications, including those proposed by the licensee, to address the impacts demonstrated by the SPRA. As part of the audit process, the staff and the SMRP requested supporting details to understand the safety enhancement expected from the plant modifications proposed by the licensee and to understand the identification as well as implementation of interim actions during the completion of the proposed modifications. The staff received details of the modifications and interim actions as part of the audit process. In its letter dated June 19, 2020, the licensee proposed regulatory commitments to complete four (4) permanent plant modifications. The letter also identified interim actions that will be taken by the licensee until the proposed permanent modifications are completed. Enclosure 2 discusses the staff's evaluation of the information provided by the licensee on the proposed permanent modifications.

In addition, NRC's backfitting experts evaluated the available information on the proposed permanent modifications. The experts determined, using Section 2 of NUREG/BR-0058 as a guide, that whether to proceed to perform a detailed backfit analysis is a management (i.e., SMRP) decision.

Based on the available information and crediting the regulatory commitments provided by the licensee in its letter dated June 19, 2020, the SMRP decided to classify Robinson as a Group 1 plant (i.e., not to pursue further regulatory action for this SPRA submittal) because:

1. the proposed permanent modifications address the impacts of the dominant risk contributors demonstrated by the SPRA;

2. the completion schedule for the proposed permanent modifications will result in the safety enhancements being achieved with a relatively short turnaround;

3. the interim actions proposed by the licensee provide defense-in-depth and support mitigation of dominant sequences while the proposed permanent modifications are implemented;

4. the NRC staff can inspect the implementation of the proposed interim actions as well as the proposed permanent modifications; and

5. the permanent plant modifications and procedural changes will be incorporated into existing Diverse and Flexible Coping Strategies (FLEX) and Extensive Damage Mitigation Guidelines (EDMG) programs (i.e., incorporated the changes into the programs that are subject to 10 CFR 50.155 requirements).

The NRC inspection staff could choose to inspect these modifications as part of routine baseline inspection activities within the reactor oversight process (e.g., equipment alignment and plant modification samples).

AUDIT REPORT

The generic audit plan dated July 6, 2017, describes the NRC staff's intention to issue an audit report that summarizes and documents the NRC's regulatory audit of licensee's SPRA submittals associated with their reevaluated seismic hazard information. The NRC staff's audit included a review of licensee documents through an electronic reading room. An audit summary document is included as Enclosure 3 to this letter.

REGULATORY COMMITMENT

In its supplement letter dated June 19, 2020, the licensee proposed regulatory commitments to complete four (4) permanent plant modifications. The NRC staff notes that NEI 99-04 "Guidelines for Managing NRC Commitments" (ADAMS Accession No. ML003680088), as endorsed by the NRC in SECY-00-0045 "Acceptance of NEI 99-04, "Guidelines for Managing NRC Commitments"" (ADAMS Accession No. ML003679799), provides an acceptable method to manage commitments. If the licensee were to change these regulatory commitments, the staff expects to be informed in accordance with the process outlined in NEI 99-04, as endorsed by the NRC. If the commitments were to be changed, the staff may revisit its conclusion.

CONCLUSION

Based on the staff's review of the Robinson submittal against the endorsed SPID guidance, the NRC staff concludes that the licensee responded appropriately to Enclosure 1, Item (8) of the 50.54(f) letter. Additionally, the staff's review concluded that the SPRA is of sufficient technical adequacy to support Phase 2 regulatory decisionmaking in accordance with the intent of the

50.54(f) letter. Based on the results and risk insights of the SPRA submittal, the regulatory commitments to complete permanent plant modifications, and implementation of interim actions, the NRC staff decided to not pursue further regulatory action associated with NTTF Recommendation 2.1 “Seismic” for Robinson.

Application of this review and decision is limited to the review of the 10 CFR 50.54(f) response associated with NTTF Recommendation 2.1 “Seismic” review. The staff notes that assessment of the SPRA for use in other licensing applications, would warrant review of the SPRA for its intended application. The NRC may use insights from this SPRA assessment in its regulatory activities as appropriate.

If you have any questions, please contact Milton Valentin at (301) 415-2864 or via e-mail at Milton.Valentin@nrc.gov.

Sincerely,

/RA/

Gregory F. Suber, Deputy Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-261

Enclosures:
1. NRC Staff SPRA Submittal Technical Review Checklist
2. NRC Staff SPRA Submittal Detailed Screening Evaluation
3. NRC Staff Audit Summary
4. Risk-Informed Evaluation of Insights

cc w/redacted encls: Listserv

NRC Staff SPRA Submittal Technical Review Checklist

Several nuclear power plant licensees are performing seismic probabilistic risk assessments (SPRAs) as part of their submittals to satisfy Near-Term Task Force (NTTF) Recommendation 2.1: Seismic. These submittals are being prepared according to the guidance in the Electric Power Research Institute – Nuclear Energy Institute (EPRI-NEI) Screening, Prioritization, and Implementation Details (SPID) document (EPRI-SPID, 2012), which was endorsed by the U.S. Nuclear Regulatory Commission (NRC) staff for this purpose. The SPRA peer reviews are also expected to follow the guidance in NEI 12-13 (NEI, 2012) as supplemented by NRC staff comments in its acceptance letter dated March 7, 2018 (NRC, 2018a, 2018b).

The SPID indicates that an SPRA submitted for the purpose of satisfying NTTF Recommendation 2.1: Seismic (hereafter referred to as NTTF Recommendation 2.1) must meet the requirements in the American Society of Mechanical Engineers-American Nuclear Society (ASME-ANS) PRA Methodology Standard (the ASME-ANS Standard). According to the SPID, either the "Addendum A version" (ASME/ANS Addendum A, 2009) or the "Addendum B version" (ASME/ANS Addendum B, 2013) of the ASME-ANS Standard can be used.

Recently, the ASME-ANS Joint Committee on Nuclear Risk Management (JCNRM), which develops and maintains the PRA standards at issue, has issued a new set of requirements for Seismic PRAs, ASME/ANS RA-S Case 1 (ASME/ANS, 2017), herein called the "Code Case Standard." The Code Case Standard contains alternative requirements to Addendums A and B for Part 5 (SPRA) of the PRA Standard. The reasons for developing the Code Case Standard were to make the SPRA requirements more consistent in some areas with the rest of the standard, and to respond to comments from users concerning the scope or the level of detail of some of the requirements.

The use of the Code Case Standard by a licensee is voluntary, but it is the NRC staff's understanding that some nuclear power plant licensees will be developing and subsequently submitting their SPRAs in response to NTTF Recommendation 2.1 using the Code Case Standard instead of either the Addendum A or the Addendum B version.

The NRC staff wrote a letter to the JCNRM on March 12, 2018 (NRC, 2018), which states in part that, "The NRC staff finds the process for developing a PRA for seismic events proposed in the ASME/ANS RA-S Case 1 acceptable," while also setting forth some conditions that must be met by a licensee's submittal if the Code Case Standard is used. Specifically, an attachment to that letter contains detailed staff comments on the Code Case Standard that need to be addressed by any submittal that references the Code Case Standard. As stated in the staff's March 2018 letter "[l]icensees may choose to retain their facility's current SPRA approach or revise it consistent with the Code Case. Any licensee use of the Code Case is voluntary."

The purpose of this staff guidance document (checklist) is to provide guidance and a checklist to the staff for the review of prospective licensee submittals using the Code Case Standard, similar to the earlier guidance and checklist (NRC, 2017) covering submittals using either the 2009 Addendum A version or the 2013 Addendum B version of the Standard.

This new staff guidance document (and checklist) is a stand-alone document. It does, however, rely heavily on the guidance material in the earlier staff guidance and checklist document, and uses a vast majority of the material in the earlier document directly.

The following table provides a checklist covering each of the Supporting Requirements (SRs) in the Code Case Standard. For most SRs, the SPID guidance does not differ from the requirement in the Code Case Standard. However, because the guidance in the SPID and the criteria of the Code Case Standard differ in some areas, or the SPID does not explicitly address an SR, the staff has developed the checklist to help NRC reviewers to address and evaluate the differences, as well as to determine the appropriate technical requirement (Code Case Standard or SPID) against which the SPRA for NTTF Recommendation 2.1 submittals should be reviewed.

In general, the SPID allows for departures or differences from the ASME-ANS Standard in the following ways:

(i) In some technical areas, the SPID's requirements tell the SPRA analyst "how to perform" one aspect of the SPRA analysis, whereas the Code Case Standard's requirements generally cover "what to do" rather than "how to do it".

(ii) For some technical areas and issues the requirements in the SPID differ from those in the Code Case Standard.

(iii) The SPID has some requirements that are not in the Code Case Standard.

All of the technical positions in the SPID have been endorsed by the NRC staff for NTTF Recommendation 2.1 submittals, subject to certain conditions concerning peer review outlined in the staff's letter to NEI dated March 7, 2018 (NRC, 2018a, 2018b), which supersedes the staff's November 12, 2012, letter to NEI (NRC, 2012).

The checklist in this document is comprised of the 16 "Topics" that require additional staff guidance because the SPID contains specific guidance that differs from the Code Case Standard or expands on it. The earlier checklist covering staff review of submittals using Addendum A or Addendum B of the ASME-ANS Standard was discussed during a public meeting on December 7, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16350A181). Each topic is covered below under its own heading, "Topic 1," "2," etc.

- Topic 1: Seismic Hazard (SPID Sections 2.1, 2.2, and 2.3)
- Topic 2: Site Seismic Response (SPID Section 2.4)
- Topic 3: Definition of the Control Point for the SSE [Safe Shutdown Earthquake] - to - GMRS [Ground Motion Response Spectra] - Comparison Aspect of the Site Analysis (SPID Section 2.4.2)
- Topic 4: Adequacy of the Structural Model (SPID Section 6.3.1)
- Topic 5: Use of Fixed-Based Dynamic Seismic Analysis of Structures for Sites Previously Defined as "Rock" (SPID Section 6.3.3)
- Topic 6: Use of Seismic Response Scaling (SPID Section 6.3.2)

- Topic 7: Use of New Response Analysis for Building Response, ISRS [In-Structure Response Spectra], and Fragilities

- Topic 8: Screening by Capacity to Select SSCs [Structures, Systems, and Components] for Seismic Fragility Analysis (SPID Section 6.4.3)

- Topic 9: Use of the CDFM [Conservation Deterministic Failure Margin]/H Methodology for Fragility Analysis (SPID Section 6.4.1)

- Topic 10: Capacities of SSCs Sensitive to High-Frequencies (SPID Section 6.4.2)

- Topic 11: Capacities of Relays Sensitive to High-Frequencies (SPID Section 6.4.2)

- Topic 12: Selection of Dominant Risk Contributors that Require Fragility Analysis Using the Separation of Variables Methodology (SPID Section 6.4.1)

- Topic 13: Evaluation of LERF [Large Early Release Frequency] (SPID Section 6.5.1)

- Topic 14: Peer Review of the SPRA, Accounting for NEI 12-13 (SPID Section 6.7)

- Topic 15: Documentation of the SPRA (SPID Section 6.8)

- Topic 16: Review of Plant Modifications and Licensee Actions

TOPIC 1: Seismic Hazard (SPID Sections 2.1, 2.2, and 2.3)

The site under review has updated/revised its Probabilistic Seismic Hazard Analysis (PSHA) from what was submitted to NRC in response to the NTTF Recommendation 2.1: Seismic 50.54(f) letter.	NO
Notes from staff reviewer: None Deviation(s) or deficiency(ies) and Resolution: N/A Consequence(s): N/A	
The NRC staff concludes that:	
• the peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to the SHA requirements in the Code Case Standard, as well as to the requirements in the SPID.	YES
• although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.	N/A
• the guidance in the SPID was followed for developing the probabilistic seismic hazard for the site.	YES
• an alternate approach was used and is acceptable on a justified basis.	N/A

TOPIC 2: Site Seismic Response (SPID Section 2.4)

<table>
<tr><td>The site under review has updated/revised its site response analysis from what was submitted to NRC in response to the NTTF Recommendation 2.1: Seismic 50.54(f) letter.</td><td>NO</td></tr>
<tr><td colspan="2">Notes from staff reviewer:

To support its submittal, Duke Energy Progress, LLC (Duke), the licensee for the H.B. Robinson Steam Electric Plant, Unit No. 2 (Robinson or RNP), performed an extensive liquefaction analysis as part of its SPRA. The licensee's analysis focused on the susceptibility of large areas of the site to differential settlement and lateral spreading; the impact of such ground displacements on SSCs important to safety, and the susceptibility of the materials within and underlying the dam to liquefaction. The staff reviewed these analyses as part of the audit process and confirmed that the fragilities used in the SPRA are consistent with the material properties found at the site. There are no open F&Os related to the seismic hazard portions of the liquefaction analysis.

Deviation(s) or deficiency(ies) and Resolution: N/A

Consequence(s): N/A</td></tr>
<tr><td>The NRC staff concludes that:

• the peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to all SRs under HLR-SHA-E in the Code Case Standard, as well as to the requirements in the SPID.

• although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.

• the licensee's development of PSHA inputs and base rock hazard curves meets the intent of the SPID guidance or another acceptable approach.

• the licensee's development of a site profile for use in the analysis adequately meets the intent of the SPID guidance or another acceptable approach.

• although the licensee's development of a shear wave velocity (V_s) profile for use in the analysis does not meet the intent of the SPID guidance, it is acceptable on another justified basis.</td><td>YES

N/A

YES

YES

N/A</td></tr>
</table>

TOPIC 3: Definition of the Control Point for the SSE-to-GMRS-Comparison Aspect of the Site Analysis (SPID Section 2.4.2)

The issue is establishing the control point where the SSE is defined. Most sites have only one SSE, but some sites have more than one SSE, for example one at rock and one at the top of the soil layer. This control point is needed because it is used as part of the input information for the development of the seismic site-response analysis, which in turn is an important input for analyzing seismic fragilities in the SPRA. The SPID (Section 2.4.1) recommends one of two approaches for establishing the control point for a logical SSE-to-GMRS comparison:	
A) If the SSE control point(s) is defined in the final safety analysis report (FSAR), it should be used as defined.	NO
B) If the SSE control point is not defined in the FSAR, one of three criteria in the SPID (Section 2.4.1) should be used.	YES
C) An alternative method has been used for this site.	N/A
The control point used as input for the SPRA is identical to the control point used to establish the GMRS and previously accepted by the staff.	YES
If yes, the control point can be used in the SPRA and the NRC staff's earlier acceptance governs.	
If no, the NRC staff's previous reviews might not apply. The staff's review of the control point used in the SPRA is acceptable.	N/A
Notes from staff reviewer: None Deviation(s) or deficiency(ies) and Resolution: N/A Consequence(s): N/A	
The NRC staff concludes that:	

• The peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to the requirements in the SPID. No requirements in the Code Case Standard specifically address this topic.	YES
• Although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's definition of the control point for site response analysis adequately meets the intent of the SPID guidance.	YES
• The licensee's definition of the control point for site response analysis does not meet the intent of the SPID guidance, but is acceptable on another justified basis.	N/A

TOPIC 4: Adequacy of the Structural Model (SPID Section 6.3.1)

The NRC staff review of the structural model finds an acceptable demonstration of its adequacy	YES
Used an existing structural model	NO
Used an enhancement of an existing model	YES
Used an entirely new model	YES
Criteria 1 through 7 (SPID Section 6.3.1) are all met.	YES

Notes from staff reviewer:

According to Table 4-2 of the SPRA Submittal, both new and revised models were used to model structures. Section 4.3.4 further explains that new state of the art finite-element models (FEMs) were developed, with a few exceptions. The model of the Reactor Containment Building (RCB) included a new FEM representation of the internal structure, a recreated FEM of the Nuclear Steam Supply System (NSSS), and a revised version of an existing lumped-mass stick model (LMSM) representation of the containment shell. The recreated NSSS model meets SPID criteria and the original LMSM of the containment shell was modified to bring it into compliance with SPID requirements by revising the modulus of elasticity, refining model discretization, and including mass moments of inertia. An existing FEM of the Class I Turbine Building was used with certain modifications to make it a median-centered model satisfying SPID modeling criteria. According to the licensee, these models were developed in compliance with SPID guidance. The peer review team judged these structural models to be “generally realistic.”

The peer review team developed a Finding-level F&O against SR SFR-B3. The licensee’s disposition to this F&O provided the results of a sensitivity study using 4 percent damping compared to 2 percent damping used in the SPRA for the Class III Turbine Building and determined that the median capacity of the updated fragility changed by less than 5 percent and that this change is insignificant to the risk results. During the audit, the licensee also explained that using 2 percent damping is more appropriate based on American Society of Civil Engineers (ASCE) standards. The NRC staff concludes that the change in fragility is small and will have an insignificant impact on the SCDF and SLERF and, furthermore, that licensee’s justification for the use of 2 percent damping in the SPRA is the appropriate.

Deviation(s) or deficiency(ies) and Resolution: None.

Consequence(s): N/A

The NRC staff concludes that:	

• The peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to the SR SFR-B3 in the Code Case Standard, as well as to the requirements in the SPID.	YES
• Although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's structural model meets the intent of the SPID guidance.	YES
• The licensee's structural model does not meet the intent of the SPID guidance, but is acceptable on another justified basis.	N/A

TOPIC 5: Use of Fixed-Based Dynamic Seismic Analysis of Structures for Sites Previously Defined as "Rock" (SPID Section 6.3.3)

Fixed-based dynamic seismic analysis of structures was used, for sites previously defined as "rock." If no, this issue is moot. If yes, on which structure(s)? Structure name: N/A Structure #1: If used, is V_S > about 5,000 feet (ft.)/second (sec.)? If 3,500 ft./sec. < V_S < 5,000 ft./sec. was peak-broadening or peak shifting used? Potential Staff Finding: The demonstration of the appropriateness of using this approach is adequate.	NO N/A N/A
Notes from staff reviewer: This site was not previously defined as "rock," therefore this issue is not applicable. Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): N/A	
The NRC staff concludes that: • The peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to the requirements in the SPID. No requirements in the Code Case Standard specifically address this topic. • Although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis. • The licensee's use of fixed-based dynamic analysis of structures for a site previously defined as "rock" adequately meets the intent of the SPID guidance. • The licensee's use of fixed-based dynamic analysis of structures for a site previously defined as "rock" does not meet the intent of the SPID guidance, but is acceptable on another justified basis.	N/A N/A N/A N/A

TOPIC 6: Use of Seismic Response Scaling (SPID Section 6.3.2)

Seismic response scaling was used. If no, this issue is moot. If yes, on which structure(s)?	No
Potential Staff Findings: If a new UHS or RLE is used, the shape is approximately similar to the spectral shape previously used for ISRS generation.	N/A
If the shape is not similar, the justification for seismic response scaling is adequate.	N/A
Consideration of non-linear effects is adequate.	N/A
Notes from staff reviewer: During the audit the NRC staff reviewed the SPRA Full Scope Peer Review report wherein the peer reviewer concluded that SR SFR-B2 was not applicable to the RNP SPRA because seismic response analyses were performed based on either new or enhanced existing models for all RNP structures and, thus, scaling of existing response analysis was not performed. Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): N/A	
The NRC staff concludes that: • The peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to the SR SFR-B2 in the Code Case Standard, as well as to the requirements in the SPID.	N/A
• Although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's use of seismic response scaling adequately meets the intent of the SPID guidance.	N/A
• The licensee's use of seismic response scaling does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

TOPIC 7: Use of New Response Analysis for Building Response, ISRS, and Fragilities

The SPID does not provide specific guidance on performing new response analysis for use in developing ISRS and fragilities. The new response analysis is generally conducted when the criteria for use of existing models are not met or more realistic estimates are deemed necessary. The requirements for new analysis are included in the standard. See all of the SR under HLR-SFR-B in the Code Case Standard. One of the key areas of review is consistency between the hazard and response analyses. Specifically, this means that there must be consistency among the ground motion equations, the soil-structure-interaction analysis (for soil sites), the analysis of how the seismic energy enters the base level of a given building, and the in-structure-response-spectrum analysis. Said another way, an acceptable SPRA must use these analysis pieces together in a consistent way. The following are high-level key elements that should have been considered:	
1. Foundation Input Response Spectra (FIRS) site response developed with appropriate building specific soil velocity profiles. Structure #1 name: Reactor Containment Building (RCB) Structure #2 name: RCB Piles Structure #3 name: Lake Robinson Dam Spillway Structure #4 name: Lake Robinson Dam ground surface Structure #5 name: FLEX Building Are all structures appropriately considered?	YES
	YES

2. Are models adequate to provide realistic structural loads and response spectra for use in the SPRA? 1. Is the SSI analysis capable of capturing uncertainties and realistic? 2. Is the probabilistic response analysis capable of providing the full distribution of the responses?	YES YES
Notes from Reviewer: The GMRS and shear wave velocity profiles used in the submittal to develop the FIRS were previously reviewed by the NRC staff in the seismic hazard information submitted to the NRC in response to the NTTF 2.1 seismic information request (ADAMS Accession No. ML15280A199). Consistent with the guidance in the SPID, the site response analysis included consideration of both aleatory and epistemic uncertainty in the development of the median shear wave velocity for the RNP site, which included development of three alternative median shear wave velocity profiles that were weighted to develop the profile used in the submittal. See Topic 4 for discussion of a Finding-level F&O against SR SFR-B3 and associated disposition by the licensee. As noted in Topic 6, the SPRA peer review team determined that SR SFR-B2 was not applicable. Table A-2 of the SPRA Submittal describes Finding-level F&O 30-1 against SR SFR-B5. The licensee's disposition to this SR concluded it was documentation only and provided the additional documentation incorporated into the SPRA notebooks for the SSI analysis of the Reactor Auxiliary Building (RAB) and RCB. Based on the peer reviewers suggested resolution, the NRC staff concludes the licensee's disposition is reasonable. Based on additional information provided during the audit, there were no other Finding-level F&Os against HLF-SFR-B SRs and all SRs were assessed to be either CC-I/II or CC-II as applicable, unless determined not to be applicable to the RNP SPRA as discussed above. Deviation(s) or deficiency(ies) and Resolution: None Consequence(s): N/A	
The NRC staff concludes:	

• The peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to all SRs under HLR-SFR-B in the Code Case Standard, as well as to the requirements in the SPID.	YES
• Although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's FIRS modeling is consistent with the prior NRC review of the GMRS and soil velocity information.	YES
• The licensee's structural model meets the intent of the SPID guidance and the Standard's requirements.	YES
• The response analysis accounts for uncertainties in accordance with the SPID guidance and the Standard's requirements.	YES
• The NRC staff concludes that an acceptable consistency has been achieved among the various analysis pieces of the overall analysis of site response and structural response.	YES
• The licensee's structural model does not meet the intent of the SPID guidance and the Standard's requirements but is acceptable on another justified basis.	YES

TOPIC 8: Screening by Capacity to Select SSCs for Seismic Fragility Analysis (SPID Section 6.4.3)

The selection of SSCs for seismic fragility analysis used a screening approach by capacity following Section 6.4.3 of the SPID. If no, see items D and E. If yes, see items A, B, and C. Potential Staff Findings:	YES
A) The recommendations in Section 6.4.3 of the SPID were followed for the screening aspect of the analysis, using the screening criteria therein.	YES
B) The approach for retaining certain SSCs in the model with a screening-level seismic capacity follows the recommendations in Section 6.4.3 of the SPID and has been appropriately justified.	YES
C) The approach for screening out certain SSCs from the model based on their inherent seismic ruggedness follows the recommendations in Section 6.4.3 of the SPID and has been appropriately justified.	YES
D) The Standard has been followed.	N/A
E) An alternative method has been used and its use has been appropriately justified.	N/A

Notes from staff reviewer:

According to Section 4.4.1 of the SPRA Submittal, a screening level high confidence of low probability of failure (HCLPF) of 0.75g, corresponding to a seismic core damage frequency (SCDF) of 5E-7/reactor-year, was developed and SSCs assigned this screening level fragility were retained in the SPRA model. This approach is in accordance with SPID guidance.

See Topic 7 for discussion of Finding-level F&O 30-1 against SR SFR-B5 and associated disposition by the licensee.

Table A-2 of the SPRA Submittal describes a Finding-level F&O against SR SFR-C1. In F&O 29-1 against SR SFR-C1 and SFR-E1, the peer review team requested the licensee perform a review of valves ranked as having high capacity based on their design criteria to confirm that all have a capacity of at least 3g in each horizontal direction and 2g vertical as assumed in the fragilities developed for these valves. The licensee conducted a comprehensive review of both safety and non-safety valves for the appropriateness of their assigned fragility. For the single safety-related valve (FCV-6416) in which the fragility was updated as a result of this review, the NRC staff confirmed during the audit that the updated fragility is documented in the fragility notebooks and is based on failure of the yoke rather than exceeding the allowable valve acceleration per the design criteria. With regards to the licensee's comprehensive

review of all the non-safety valves, the NRC staff determined during the audit that the fragilities for several valves were revised downward, but that these changes were not reflected in the fragility notebooks. During the audit the licensee confirmed that these revised fragilities were incorporated in the SPRA used in the submittal and that the SPRA documentation is outdated and needs to be updated to reflect these revised fragilities. The licensee further clarified that all the resolutions to the SPRA F&Os were incorporated in the SPRA used in the submittal. Deviation(s) or deficiency(ies) and Resolution: None Consequence(s): N/A	
The NRC staff concludes:	
• The peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to the SRs SFR-C1, SFR-C2, and SPR-B5 in the Code Case Standard, as well as to the requirements in the SPID.	YES
• Although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's use of a screening approach for selecting SSCs for fragility analysis meets the intent of the SPID guidance.	YES
• The licensee's use of a screening approach for selecting SSCs for fragility analysis does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

TOPIC 9: Use of the CDFM/Hybrid Methodology for Fragility Analysis (SPID Section 6.4.1)

The CDFM/Hybrid method was used for seismic fragility analysis. If no, See item C) below and next issue. If yes: Potential Staff Findings: A) The recommendations in Section 6.4.1 of the SPID were followed appropriately for developing the CDFM High Confidence Low Probability of Failure capacities.	YES YES
B) The Hybrid methodology in Section 6.4.1 and Table 6-2 of the SPID was used appropriately for developing the full seismic fragility curves.	YES
C) An alternative method has been used appropriately for developing full seismic fragility curves.	YES

Notes from staff reviewer:

According to the summary of the SPRA peer review conclusions provided in Section A.4 of the SPRA submittal, representative fragilities using the EPRI Hybrid approach were calculated for SSCs that were not dominant risk contributors. During the audit, the licensee further explained that representative fragilities were developed using several approaches, including: 1) using the CDFM/Hybrid methodology, including determination of the HCLPF capacity and use of SPID Table 6-2 for defining the variability parameters (e.g., relays, building structures), 2) determining the capacity of the governing failure mode by scaling the results of previous response spectrum performed for the DBE to the GMRS (e.g., CST piping), 3) justifying the applicability of a generic fragility for similar components from industry/NRC documents (e.g., HVAC ducts), and 4) assigning a screening level fragility based on walkdown observations, past earthquake and/or test experience, and past seismic evaluations.

For the Robinson site, the licensee determined that soil liquefaction (settlement and lateral spreading) is risk significant. Because failure probabilities from these hazards do not fit a double lognormal distribution typical of standard fragilities, failures of SSCs due to liquefaction-induced settlement and lateral spreading displacements were expressed as mean conditional probabilities of failures at multiple hazard levels. Sections 3.1.5, 3.1.6, and 4.4.2 of the SPRA Submittal describes the development of these distributions.

The staff confirmed that the CDFM/Hybrid methodology generally followed EPRI TR-1002988, "Seismic Fragility Applications Guide," EPRI NP 6041-SL, "A Methodology for Assessment of Nuclear Plant Seismic Margin," EPRI TR-103959, “Methodology for Developing Seismic Fragilities,” and EPRI 3002000709, “Seismic Probabilistic Risk Assessment Implementation Guide.” In addition, the NRC staff reviewed the resolution of finding-level F&O 29-3 against SR SFR-E4 and SFR-F1 and concluded that the licensee's disposition is sufficient for this submittal.

Liquefaction-induced settlement is the failure mode for several dominant risk contributors for SCDF (Tables 5.4-2 in the SPRA submittal) and SLERF (Table 5.5-2 in the SPRA submittal). The licensee evaluated liquefaction and its effects at the HB Robinson nuclear power plant. This included evaluations of liquefaction susceptibility and triggering; potential for a site-wide, continuous, liquefiable soil layer; liquefaction-induced settlement; and lateral spread displacements. The NRC staff used the audit process to review the license's seismic liquefaction evaluation at the power plant and FLEX storage area.

The licensee evaluated liquefaction triggering using standard penetration test (SPT) explorations located within the main plant area where safety related SSCs are located. The licensee used a sensitivity study to capture the range of effects from assumptions made within the triggering evaluation. Based on its evaluations, the licensee concluded that there was an absence of a continuous liquefiable material in the main plant area. In response to an NRC staff audit question, the licensee stated that the susceptible soil is present in lenses that vary spatially in textural composition and relative density. The NRC staff verified that the licensee used acceptable methodology and site-specific data in support of their conclusion. The licensee also estimated the liquefaction-induced settlement and lateral spreading for areas identified by the liquefaction triggering evaluation. The FLEX storage area north of the main plant area was evaluated only for liquefaction susceptibility and triggering, but the licensee concluded that although settlement could occur in isolated zones around the FLEX storage area, it was unlikely to suffer site-wide soil liquefaction. In response to an audit question from the NRC staff, the licensee discussed mitigation strategies in the event of settlement in the FLEX deployment paths and staging areas within the main plant area, including use of debris removal equipment and using alternate paths or off-path routing. Based on its review of the liquefaction related information, the NRC staff concludes that the licensee's liquefaction evaluations for the plant and FLEX storage area are technically acceptable for this SPRA submittal because the licensee: (1) used triggering, settlement, and lateral spread displacement models that are appropriate for those evaluations, (2) justified their assumptions and used sensitivity analyses where practical, (3) applied acceptable probabilistic approaches, and (4) adequately developed uncertainty distributions for the model parameters in the probabilistic computations. The staff's evaluation of the liquefaction induced failure of the Robinson Dam is discussed in detail under Topic 12.

Deviation(s) or deficiency(ies) and Resolution: None.

Consequence(s): N/A

The NRC staff concludes that: • The peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to the requirements in the SPID. No requirements in the Code Case Standard specifically address this Topic. • Although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis. • The licensee's use of the CDFM/Hybrid method for seismic fragility analysis meets the intent of the SPID guidance. • The licensee's use of the CDFM/Hybrid method for seismic fragility analysis does not meet the intent of the SPID guidance, but is acceptable on another justified basis	 YES N/A YES N/A

TOPIC 10: Capacities of SSCs Sensitive to High-Frequencies (SPID Section 6.4.2)

The SPID requires that certain SSCs that are sensitive to high-frequency seismic motion must be analyzed in the SPRA for their seismic fragility using a methodology described in Section 6.4.2 of the SPID. <u>Potential Staff Findings</u>: The NRC staff review of the SPRA's fragility analysis of SSCs sensitive to high frequency seismic motion finds that the analysis is acceptable. The flow chart in Figure 6-7 of the SPID was followed. The flow chart was not followed but the analysis is acceptable on another justified basis.	 YES YES N/A
Notes from staff reviewer: Section 4.1.2 of the SPRA submittal describes the relay evaluation that was performed in accordance with the SPID. During the audit the licensee provided additional plant documentation on the relay evaluation performed for the SPRA development. This analysis considered seismic-induced contact chatter failure of electro-mechanical relays, electro-mechanical contacts, mercury switches, control switches, process switches, and circuit breakers. The Seismic Equipment List (SEL) was used to establish the scope of the circuit analysis performed, from which a list of relay contacts was developed to determine where relay chatter could adversely impact the credited function of an SEL item. Relays in which chatter was determined to have an impact were evaluated further for modeling in the SPRA (no relays were replaced with higher capacity components). Representative fragilities were developed using the CDFM/Hybrid method for most of the unscreened relays, while detailed fragilities were developed for a select few relays determined to be risk contributors using the separation of variable (SoV) method. There were no peer review F&Os against SR SFR-E5. F&O 28-4 against SR SFR-F2 requested justification for the representative fragilities developed for relays. The NRC staff reviewed the licensee's disposition of this F&O and concluded it to be sufficient for this submittal. Deviation(s) or deficiency(ies) and Resolution: None Consequence(s): N/A	
The NRC staff concludes that: • The peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to the SR SFR-E5 in the Code Case Standard, as well as to the requirements in the SPID.	 YES N/A

• Although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis. • The licensee's fragility analysis of SSCs sensitive to high frequency seismic motion meets the intent of the SPID guidance. • The licensee's fragility analysis of SSCs sensitive to high-frequency motion does not meet the intent of the SPID guidance, but is acceptable on another justified basis.	YES N/A

TOPIC 11: Capacities of Relays Sensitive to High-Frequencies (SPID Section 6.4.2)

The SPID requires that certain relays and related devices (generically, "relays") that are sensitive to high-frequency seismic motion must be analyzed in the SPRA for their seismic fragility. Although following the Standard is generally acceptable for the fragility analysis of these components, the SPID (Section 6.4.2) contains additional guidance when either circuit analysis or operator-action analysis is used as part of the SPRA to understand a given relay's role in plant safety. When one or both are used, the NRC reviewer should use the following elements of the checklist.	
i) <u>Circuit analysis</u>: The seismic relay-chatter analysis of some relays relies on circuit analysis to assure that safety is maintained. (A) If <u>no</u>, then (B) is moot. (B) If <u>yes:</u> <u>Potential Staff Finding</u>: The approach to circuit analysis for maintaining safety after seismic relay chatter is acceptable.	YES YES
ii) <u>Operator actions</u>: The relay-chatter analysis of some relays relies on operator actions to assure that safety is maintained. (A) If <u>no</u>, then (B) is moot. (B) If <u>yes:</u> <u>Potential Staff Finding</u>: The approach to analyzing operator actions for maintaining safety after seismic relay chatter is acceptable.	NO YES
Notes from staff reviewer: Topic 10 discusses how the SEL was used to identify SSCs, including relays, potentially sensitive to failure via contact chatter. From information provided during the audit, the circuit analysis performed for the RNP Fire PRA was used in the relay chatter assessment. It was determined that just one SSC from the SEL was not already evaluated in the Fire PRA. Also, while Section 4.1.2 of the SPRA Submittal discusses the potential for crediting operator actions to reset relays failed by chatter, information provided during the audit indicates that operator actions were not credited either in the relay screening evaluation or in the SPRA quantification to recover relays failed due to chatter. There were no peer review F&Os against SR SPR-B6.	

Deviation(s) or deficiency(ies) and Resolution: None Consequence(s): N/A	
The NRC staff concludes that:	
• the peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The relevant peer review findings are those that relate to SR SPR-B6 in the Code Case Standard, as well as to the requirements in the SPID.	YES
• although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.	N/A
• the licensee's analysis of seismic relay-chatter effects meets the intent of the SPID guidance.	YES
• the licensee's analysis of seismic relay-chatter effects does not meet the intent of the SPID guidance, but is acceptable on another justified basis.	N/A

TOPIC 12: Selection of Dominant Risk Contributors that Require Fragility Analysis Using the Separation of Variables Methodology (SPID Section 6.4.1)

<table>
<tr><td>The CDFM methodology has been used in the SPRA for analysis of the bulk of the SSCs requiring seismic fragility analysis.

If <u>no</u>, the staff review will concentrate on how the fragility analysis was performed, to support one or the other of the “potential staff findings” noted just below.

If <u>yes</u>, significant risk contributors for which use of SOV fragility calculations would make a significant difference in the SPRA results have been selected for SOV calculations.”</td><td>NO</td></tr>
<tr><td><u>Potential Staff Findings</u>:
A) The recommendations in Section 6.4.1 of the SPID were followed concerning the selection of the “dominant risk contributors” that require additional seismic fragility analysis using the separation-of-variables methodology.</td><td>YES</td></tr>
<tr><td>B) The recommendations in Section 6.4.1 were not followed, but the analysis is acceptable on another justified basis.</td><td>N/A</td></tr>
<tr><td colspan="2">Notes from staff reviewer:

The SPRA submittal describes the process used by the licensee for selecting dominant risk contributors for more detailed fragility analysis. Specifically, Section 4.4.2 states the initial SPRA quantification was performed using representative fragilities developed for all SEL components and that these representative fragilities were developed with a “slight conservative bias” and/or based on generic data. For the dominant risk contributors, more detailed fragilities were developed using the SoV methodology. The SPRA was then requantified with the revised fragilities, and the iterative process of SPRA quantification and development of more detailed fragilities repeated until “the top risk contributors were ultimately characterized with a realistic and plant-specific fragility”. The fragilities of most dominant risk contributors reported in the SPRA submittal are based on detailed SoV fragilities as noted in Tables 5.4-2 and 5.5-2 of the SPRA submittal. During the audit, the NRC staff confirmed that a select set of SSCs were modeled with detailed fragilities calculated using the SoV approach.

Section 4.4.2 of the SPRA submittal further explains that the representative fragilities were developed based on a combination of design information and calculations, past RNP seismic evaluations, judgements from RNP seismic walkdowns, past fragility estimates and experience, recent RNP assessments addressing NTTF requirements, and the results of the RNP seismic response analysis for structures described in Section 4.3 of the SPRA submittal. Section 4.2.1 of the submittal further clarifies that, during the seismic walkdowns, expert judgement was used to rank the seismic capacity of each SSC as either “Rugged,” “High,” “Medium,” or “Low” seismic capacity and describes the criteria used to make these rankings. A combination of the results of these rankings, and the other information described above, were used to develop the representative fragilities.</td></tr>
</table>

Table A-2 of the SPRA Submittal describes finding-level F&O 2-1 against SR SFR-E3 regarding including overtopping as a failure mode in the SPRA and combining post-earthquake deformations with the deformations during shaking in the development of the fragility for the Robinson Dam in a manner consistent with recent studies. The licensee's disposition to this F&O is that overtopping was determined not to be likely based on the analytical results (i.e., the calculated crest settlements were less than the available freeboard). The licensee also stated that the studies referred to in the F&O were overly conservative and inappropriate for use in an SPRA. During the audit, the licensee made available for NRC staff review a substantial amount of information on the constitutive models, publications that support the applicability and reliability of the constitutive models, geotechnical and other model input parameters, and model verification and testing applicable to the liquefaction analysis of the Robinson Dam, and additional justification for not including post-seismic deformation in the calculated fragility for the Robinson Dam. The information provided by the licensee in the audit included results of a sensitivity analysis for the contraction rate parameter used in the constitutive models, which showed the licensee's analysis to be slightly conservative. The licensee also provided the results of an analysis that showed that including the post-seismic deformation contribution would only marginally increase the crest settlement and not result in overtopping. The licensee provided adequate discussion on the validation approach used in the modeling. Based on its review, the NRC staff determined that the licensee's evaluation of Lake Robinson Dam fragility is technically acceptable for this SPRA submittal because (1) evaluated potential failure modes appropriate for embankment dams, (2) characterized the properties of the alluvium layer and dam based on past and recent geotechnical studies, (3) calibrated model parameters and post-earthquake deformation in a technically defensible manner, (4) adequately addressed the uncertainty distributions for the model parameters, and (5) applied acceptable probabilistic approaches and historical data to evaluate the fragility.

The staff notes that the licensee's analysis for post-seismic deformation contribution used a methodology different than that recommended by the peer reviewer in F&O 2-1. As described in the F&O, applying the peer reviewer's proposed methodology can potentially result in overtopping. Because of the varying results of different methodologies, the NRC staff considers the treatment of post-seismic deformation to be a "key" source of uncertainty in the licensee's SPRA. However, based on its review, the NRC staff concludes that (1) failure of the Robinson Dam is already a dominant risk contributor and accounting for potential overtopping failure due to post-earthquake crest settlement would not change the primary insights from the SPRA, and (2) the licensee's proposed modification, discussed further in Enclosure 2, while not well defined, identifies the availability and reliability of a water source, including alternatives to the use of Lake Robinson, as an important feature of the modification. For these reasons, the NRC staff finds that the licensee's treatment of liquefaction failure of the Robinson Dam is acceptable for the decisionmaking for this submittal and that the impact of the "key" uncertainty in the modeling of the Robinson Dam is understood by the staff.

The NRC staff finds the licensee's methodology acceptable because it is consistent with Section 6.4.3 of the SPID to utilize screening fragilities initially followed by detailed fragility analysis of SSCs that are significant contributors to seismic risk.

Deviation(s) or deficiency(ies) and Resolution: None

Consequence(s): N/A

The NRC staff concludes:	
• the peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to SFR-E3 in the Code Case Standard and the requirements in the SPID.	YES
• although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.	N/A
• the licensee's method for selecting the "dominant risk contributors" for further seismic fragilities analysis using the separation-of-variables methodology meets the intent of the SPID guidance.	N/A
• the licensee's method for selecting the "dominant risk contributors" for further seismic fragilities analysis using the separation-of-variables methodology does not meet the intent of the SPID guidance, but is acceptable on another justified basis.	YES

TOPIC 13: Evaluation of LERF (SPID Section 6.5.1)

The NRC staff review of the SPRA's analysis of LERF finds an acceptable demonstration of its adequacy. <u>Potential Staff Findings</u>: A) The analysis follows each of the elements of guidance for LERF analysis in Section 6.5.1 of the SPID, including in Table 6-3. B) The LERF analysis does not follow the guidance in Table 6-3 but the analysis is acceptable on another justified basis.	YES YES N/A
Notes from staff reviewer: Section 4.1 of the submittal describes the development of an SEL for RNP, including identifying SSCs for achieving safe shutdown following a seismic event and those that mitigate radioactivity releases if core damage occurs. Section 5.3.2 of the submittal states that the technical basis of the SPRA is based on the internal event models for CDF and LERF. Specifically, with regards to LERF, this section also explains that the SPRA includes failure of key structures that lead directly to core damage and to a large early release. All the top 25 LERF cutsets identified in Table 5.5-1 are containment isolation failures due to either seismic-induced failure of the RCB or loss of containment cooling caused by seismic-induced loss of power. Table A-2 of the submittal identifies no finding-level F&Os against SRs SPR-E1 and SPR-E5 and four F&Os (24-7, 24-21, 24-22, and 24-23) against SR SPR-E6. The NRC staff reviewed the licensee's disposition to F&Os 24-7, 24-21, and 24-22 and determined them to be acceptable for the SPRA submittal. Finding-level F&O 24-23 states that the SPRA uses the internal events LERF model as its basis and that a number of the LERF SRs in the internal events PRA (IEPRA) are only met at CC-I. During the audit, the licensee identified that the following six LERF SRs were assessed by the IEPRA peer review to be CC-I: LE-C2, LE-C4, LE-C11, LE-C12, LE-D6, and LE-F1. The licensee explained that subsequent to the IEPRA peer review, four of these SRs were resolved as closed and to meet CC-II or CC-I/II as applicable by an Independent Assessment Team (see Topic 14 for further discussion of this closure assessment). One of these SRs had a Finding-level F&O against it in which the resolution was assessed to be a PRA upgrade by the Independent Assessment Team. A subsequent focused-scope peer review assessed the SR to meet CC-II did not develop any Finding-level F&Os (see Topic 14 for further discussion of this focused-scope peer review). The Finding-level F&O against one SR, LE-C4, was left open by the Independent Assessment Team and the SR continues to be assessed as CC-I. The licensee explained that resolution of this F&O would not impact the SPRA because it is on interfacing systems LOCA (ISLOCA) modeling which, in the SPRA, is caused by failure of the containment building and assumed to go directly to LERF. The NRC staff finds the licensee's disposition to F&O 24-23 acceptable for the SPRA submittal because five of the six SRs are now assessed as CC-II using an NRC-accepted processes (see Topic 14) and the resolution to the F&O associated with the sixth SR will not impact the staff's decision on this SPRA submittal.	

During the audit, the NRC staff reviewed the 2010 peer review report of the IEPRA and the subsequent 2017 Independent Assessment Team report (see Topic 14 for further discussion of these reviews). The 2010 peer review identified F&O LE-E1-1 to provide appropriate justification for parameter values selected for equipment and operator response in the accident progression analysis and for the RNP containment event tree. The 2017 Independent Assessment Team left this F&O open because of insufficient documentation of the parameter values. The NRC staff noted that previous risk-informed license amendment requests for RNP (e.g., NFPA 805, Title 10 of the Code of Federal Regulations (10 CFR) 50.69, ILRT) have indicated that many of the Level 2 PRA parameters are based on expert judgement, which is a source of uncertainty. To address this concern, the licensee identified two non-seismic failures that are top contributors to SLERF, Plant Damage States 3P and 3J, both of which are split fractions based on severe accident phenomenology that determine the fraction of CDF accident sequences that progress to LERF. Based on a review of the input parameters, the licensee identified two judgement-based parameters whose uncertainty could potentially affect these LERF split fractions: 1) probability of hydrogen burn under both low and high steam environments due to high pressure melt ejection causing direct containment heating (HPME/DCH) and; 2) probability of containment failure due to the associated pressure loading. The licensee explained that for the SPRA, both judgement-based parameters were replaced by the results of plant-specific thermal-hydraulics analyses to provide more realistic estimates of the probability of global hydrogen burn, peak containment loading, and containment failure probability given the pressure loading. Based on the results of these analyses, containment failure due to HPME/DCH was screened from the SLERF model for both PDS 3P and 3J, which decreased SLERF by about 20 percent. The plant-specific thermal-hydraulics analyses cited by the licensee were the subject of a focused-scope peer-review which did not result in any finding level F&Os. Based on the results of the licensee's plant-specific analyses and the focused-scope peer-review for those analyses, the NRC staff finds the licensee's disposition to F&O LE-E1-1 reasonable for the SPRA submittal.

In summary, based on its review of the submittal and the information available through the audit, the NRC staff finds the licensee's SLERF model acceptable for the SPRA submittal.

Deviation(s) or deficiency(ies) and Resolution: None

Consequence(s): N/A

The NRC staff concludes that:	
• the peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The relevant peer review findings are those that relate to the SRs SPR-E1, E5, and E6 in the Code Case Standard, as well as to the requirements in the SPID.	YES
• although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.	N/A

• the licensee's analysis of LERF meets the intent of the SPID guidance.	YES
• the licensee's analysis of LERF does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

TOPIC 14: Peer Review of the SPRA, Accounting for NEI 12-13 (SPID Section 6.7)

The NRC staff review of the SPRA's peer review findings, observations, and their resolution finds an acceptable demonstration of the peer review's adequacy.	YES
Potential Staff Findings: A) The analysis follows each of the elements of the peer review guidance in Section 6.7 of the SPID as supplemented by NRC staff comments in the NRC letter dated March 7, 2018 (NRC 2018a, 2018b).	YES
B) The composition of the peer review team meets the SPID guidance as supplemented by NRC staff comments in the NRC letter dated March 7, 2018 (NRC 2018a, 2018b).	YES
C) The peer reviewers focusing on seismic response and fragility analysis have successfully completed the Seismic Qualifications Utility Group (SQUG) training course or equivalent (see SPID Section 6.7).	YES
In what follows, a distinction is made between an "in-process" peer review and an "end-of-process" peer review of the completed SPRA report. If an in-process peer review is used, go to (D) and then skip (E). If an end-of-process peer review is used, skip (D) and go to (E).	
D) The "in-process" peer-review process followed the "in-process" peer review guidance in the SPID (Section 6.7), including the three "bullets" and the guidance related to NRC's additional input in the paragraph immediately following those three bullets. These three bullets are: • the SPRA findings should be based on a consensus process, and not based on a single peer review team member • a final review by the entire peer review team must occur after the completion of the SPRA project • an "in-process" peer review must assure that peer reviewers remain independent throughout the SPRA development activity.	N/A
If no, go to (F).	

If yes, the “in process” peer review approach is acceptable. Go to (G). E) The “end-of-process” peer review process followed the peer review guidance in the SPID (Section 6.7) as supplemented by NRC staff comments in the NRC letter dated March 7, 2018 (NRC 2018a, 2018b). If no, go to (F). If yes, the “end-of-process” peer review approach is acceptable. Go to (G).	YES
F) The peer-review process does not follow the guidance in the SPID as supplemented by NRC staff comments in the NRC letter dated March 7, 2018 (NRC 2018a, 2018b), but is acceptable on another justified basis.	N/A
G) The licensee peer-review F&Os were satisfactorily resolved or were determined not to be significant to the SPRA conclusions for this review application.	YES
Notes from staff reviewer: The full-scope peer review of the SPRA was conducted in November 2018 against the CC-II supporting requirements of PRA Standard ASME/ANS RA-S Case 1 (Ref. ASME/ANS, 2017), and associated NRC clarifications (Ref. NRC, 2018), and in accordance with the peer review characteristics and attributes described in RG 1.200, Revision 2 (Ref. NRC, 2009). ASME/ANS RA-S Case 1 has been approved by the NRC for use in regulatory applications, subject to certain conditions and clarifications (Ref. NRC, 2018). The peer review team utilized the peer review process defined in NEI 12-13 (Ref. NEI, 2012) and associated NRC clarifications (Ref. NRC, 2018a and 2018b). The SPRA submittal states that the peer review team members met the required qualifications for peer reviewers in accordance with the ASME/ANS PRA Standard (Ref. ASME/ANS Addendum A, 2009 and ASME/ANS Addendum B, 2013) and the guidelines of NEI 12-13 (Ref. NEI, 2012). During the audit the NRC staff confirmed that the peer reviewers were independent of the RNP PRA development for any hazard. Concurrence on the assignment of capability categories to each SR was based on a consensus process involving all members of the review team. During the audit, the NRC staff reviewed the SPRA Peer Review Report which provided detailed resumes for all of the peer reviewers. The lead reviewer of the plant response model has successfully completed the seismic qualification user group (SQUG) training course and participated in the plant walkdown. The resume for the fragility reviewer who participated in the plant walkdown showed significance experience with SPRAs, seismic qualification of plant equipment, and EPRI SQUG involvement and leadership. The resumes for the lead fragility reviewer and second plant response model reviewer also showed substantial experience with plant structural analysis (including seismic qualification of SSCs) and PRA development, respectively. It is NRC staff judgement that the peer review team members meet the SPID composition and qualification requirements.	

All elements of the SPRA were peer reviewed, including those identified in Section 6.7 of the SPID. A total of 49 Finding-level SPRA Facts and Observations (F&Os) generated by the peer review are documented in Table A-2 of the SPRA submittal along with dispositions for the submittal. During the audit it was confirmed that the complete text of the Findings, the basis for the Findings, and the resolutions suggested by the peer review are provided in Table A-2.

A focused-scope peer review of the Robinson Dam fragility analysis was conducted in September 2019 against the CC-II supporting requirements of PRA Standard ASME/ANS RA-S Case 1 (Ref. ASME/ANS, 2017) and associated NRC clarifications (Ref. NRC, 2018). The peer reviewer utilized the peer review process defined in NEI 12-13 (Ref. NEI, 2012). Thirteen SRs were assessed during this peer review, resulting in two finding-level F&Os, which are documented in Table A-2 of the SPRA submittal along with dispositions for the submittal. During the audit the NRC staff confirmed that the peer reviewer was independent of the RNP PRA development for any hazard and had substantial experience with seismic analysis of dams. It was also confirmed that the complete text of the Findings, the basis for the Findings, and the resolutions suggested by the peer reviewer are provided in Table A-2.

A focused-scope peer review was also conducted in September 2019 of an upgrade to the LERF model to more realistically model high-pressure melt injection/direct containment heating and subsequent hydrogen combustion. The peer review assessed seven LE SRs from the IEPRA standard and two SPR SRs from the SPRA standard. The IEPRA SRs were peer reviewed against the CC-II supporting requirements of PRA Standard ASME/ANS RA-Sa-2009 (Ref. ASME/ANS, 2009), and associated NRC clarifications in RG 1.200, Revision 2 (Ref. NRC, 2009). The SPRA SRs were peer reviewed against the CC-II supporting requirements of PRA Standard ASME/ANS RA-S Case 1 (Ref. ASME/ANS, 2017), and associated NRC clarifications (Ref. NRC, 2018). No finding-level F&Os resulted from this peer review. The NRC staff reviewed the licensee's disposition to each of the peer review and focused-scope peer review finding-level F&Os.

Table A-2 of the submittal identifies finding-level F&O 24-8 from the SPRA full-scope peer-review that challenges the SPRA assumption that steam-driven auxiliary feedwater (SDAFW) fails 50 percent of the time the Class III Turbine Building (TB) fails. The assumption appears to be a "key" assumption for the SPRA and is identified as such by the peer-review team and the licensee (Section A.8 of Appendix A of the submittal). In fact, according to information available to the staff as part of its audit, the event TB-CLASS3-1 which represents the 50 percent assumption contributes approximately 15 percent to the total SCDF. The licensee's disposition to this F&O provides a qualitative justification for retaining the 50 percent assumption based on a post peer-review walkdown. During the audit the licensee provided further justification for the 50 percent assumption as follows: 1) Latin Hypercube simulations of the collapse of the Class III TB due to ground shaking show the dominant collapse direction to not be toward the Class I TB where SDAFW is located and 2) modeling of the collapse of the Class III TB due to pounding with the RAB show that the Class III TB will collapse with equal likelihood either toward or away from the Class I TB. Based on the results of the licensee's analyses, the NRC staff finds that there is a reasonable basis for concluding that the Class I TB would likely collapse less than 50 percent of the time that the Class III TB collapses due to seismic events.

However, SDAFW functionality could still be impacted in cases where the Class III TB collapses and the Class I TB does not collapse. During the audit the licensee justified that this is unlikely to be the case because post-peer review walkdowns and interviews with the AFW system engineer determined that 1) the SDAFW pump was judged to have an adequate separation distance from the Class III TB to not be impacted by its collapse, 2) the loss of certain SDAFW support SSCs do not impact successful operation of the SDAFW pump, and 3) SSCs necessary for successful operation of the SDAFW pump were judged to be shielded by the SDAFW pump skid and/or Class I TB structure. However, the licensee's assessment did not address why feedwater piping from the CST to the SDAFW, which passes through the Class III TB in a pipe trench, does not fail when the Class III TB collapses.

In Table 5.7-1 of the SPRA submittal, the licensee provided the results of a sensitivity analysis assuming guaranteed failure of SDAFW when the Class III TB fails (Sensitivity Case SY-2e). These results show that SCDF and SLERF increase by a modest 14 percent and nine percent, respectively. During the audit, the licensee provided the importance analysis results for this sensitivity case, which showed increased importance of the Class III TB and a corresponding decrease in the importance of the TB Gantry Crane (seismic failure of which impacts availability of SDAFW) and failure of SDAFW due to liquefaction-induced settlement. The other top contributors to SCDF and SLERF did not change significantly. The licensee also provided the results from a sensitivity that assumed guaranteed failure of the SDAFW when the Class III TB fails in conjunction with the credit for the modification proposed by the licensee in the submittal. The sensitivity demonstrated that the impact of the proposed modification, as modeled in the sensitivity, did not change significantly based on the assumption related to failure of SDAFW when the Class III TB fails.

Based on these sensitivities, the NRC staff understands the impact of the key assumption that collapse of the Class III TB fails the Class I TB (and therefore SDAFW) 50 percent of the time and that the assumption does not alter the staff's decision regarding this SPRA submittal.

The disposition to F&O 24-1 provided in Table A-2 of the SPRA submittal states that no changes were made to the SPRA model based on the licensee's review of the IEPRA model assumptions. Based on information reviewed during the audit this included assumptions on the timing of an operator action to align deepwell water for AFW supply. The licensee explained that no adjustments to the HFE for this operator action were necessary based on 1) with the CST available there are well over 12 hours available before deepwell water is needed and 2) the deepwell water supply to the containment coolers is not credited in the PRA, and 3) this action is not relied on in the sensitivity analysis of the enhanced FLEX modification credited in the SPRA submittal. The NRC staff finds the licensee's disposition to F&O 24-1 reasonable for this SPRA submittal.

The IEPRA model-of-record as of June 2015 was used as the basis for the development of the SPRA model. However, the submittal provided no information about the technical adequacy of the IEPRA model. NRC staff position #4 related to NEI 12-13 specifies that the SPRA peer review team is required to review all of the IEPRA F&Os and determine whether the resolutions were appropriate and in accordance with the PRA standard. Furthermore, SR SPR-B2 in ASME/ANS RA-S Case 1 requires the seismic peer review team to assess the status of the IEPRA model F&Os relevant to the SPRA. During the audit, the NRC staff reviewed the SPRA peer review report, IEPRA peer review report,

and the IEPRA F&O closure report. The IEPRA (excluding external flooding) was peer reviewed in October 2009 against the CC-II supporting requirements of PRA Standard ASME/ANS RA-Sa-2009 (Ref. ASME/ANS, 2009), and associated NRC clarifications in RG 1.200, Revision 1 (Ref. NRC, 2007). While RG 1.200, Revision 2 (Ref. NRC, 2009) is the current NRC guidance, the NRC staff concludes that the differences between the two RG 1.200 revisions is not expected to be significant to the SPRA submittal based on the results of previous gap assessments. An Independent Assessment Team (IAT) evaluated the licensee's resolutions to each of the peer review finding-level F&Os in July 2017 and closed all but six F&Os and assessed one F&O resolution to be an upgrade. The IAT F&O closure process was conducted in accordance with NRC guidance in letters dated May 1, 2017 (Ref. NRC, 2017a) and May 3, 2017 (Ref. NRC, 2017b). A focused-scope peer review of the identified upgrade was subsequently performed with no resultant finding-level F&Os (this upgrade was for a revised steam generator tube rupture analysis, which is not significant to the SPRA submittal). During the audit the NRC staff reviewed the licensee's disposition to each of the IEPRA open peer review Finding-level F&Os. With the exception of F&O LE-E1-1, the NRC staff found the licensee's resolutions to these open F&Os are not likely to impact the SPRA submittal. Refer to Topic 14 for the NRC staff's evaluation of the licensee's disposition to F&O LE-E1-1. The IAT F&O closure assessment was conducted in July 2017, while the SPRA model was developed using the June 2015 IEPRA model of record. During the audit the NRC staff identified that resolutions to several IEPRA F&Os that were closed by the IAT could impact the SPRA. The license confirmed that the changes made to the IEPRA model to address these F&Os were made prior to June 2015, which is the date of the IEPRA model files and notebooks that were provided to the IAT for review. Deviation(s) or deficiency(ies) and Resolution: None Consequence(s): N/A	
The NRC staff concludes: • the licensee's peer-review process meets the intent of the SPID guidance as supplemented by NRC staff comments in the NRC letter dated March 7, 2018 (NRC 2018a, 2018b). • the licensee's peer-review process does not meet the intent of the SPID guidance as supplemented by NRC staff comments	YES N/A

in the NRC letter dated March 7, 2018 (NRC 2018a, 2018b), but is acceptable on another justified basis.	

TOPIC 15: Documentation of the SPRA (SPID Section 6.8)

The NRC staff review of the SPRA's documentation as submitted finds an acceptable demonstration of its adequacy. The documentation should include all the items of specific information contained in the 50.54(f) letter as described in Section 6.8 of the SPID.	YES YES
Notes from staff reviewer: Tables 2-1 and 2-2 of the submittal provide a cross-reference of information required by 10 CFR 50.54(f) and specified in Section 6.8 of the SPID to the sections of the submittal where the information can be found. The level-of-detail of the information provided appears to be generally consistent with that specified in Section 6.8 of the SPID. The SPID requires that there should be sufficient information to assess the results to all key aspects of the analysis. Section 5.3.2 of the submittal identifies and discusses several SPRA quantification assumptions. Sections 5.4 and 5.5 of the submittal presents and discusses the results. It is noted, however, that not all the information identified in Section 6.8 of the SPID with regard to what was submitted for the Individual Plant Examination of External Events (IPEEE) program is included in the submittal (e.g., all functional/systemic event trees). However, the SPID only identifies this IPEEE information as guidance for consideration in the 50.54(f) response. The submittal explains that the SPRA model reflects the as-built/as-operated RNP as of the freeze date for the internal events model (June 2015). Section 5.6 of the submittal presents the SPRA quantification uncertainty results for SCDF and SLERF (i.e., the median (50 percent), and the 5th and 95th percentiles). However, according to the NRC staff memorandum dated August 29, 2017 (ADAMS Accession No. ML17146A200), the NRC staff also utilizes the mean SCDF and SLERF to develop a recommendation on whether the plant should move forward as a Group 1, 2, or 3 plant. During the audit, the license provided the mean CDF and LERF as 1.3E-04 per year and 2.5E-05 per year, respectively. Based on the information available to the staff as part of its audit, the seismic event tree includes an event questioning whether offsite power is available (S-OSP), which occurs after the event questioning whether the Class III TB has failed (S-TB-CLASS3). The failure of the Class III TB results in a consequential loss-of-offsite power (LOOP) due to the switchgear in that building. The licensee confirmed that the availability of offsite power question is not needed once the Class III TB is failed because offsite power is consequentially failed as well without question. The licensee also confirmed that this logic was correctly implemented in the SPRA used in the submittal. Section 5 of the SPRA submittal identifies that the SPRA includes an assumption of failure of the Condensate Storage Tank (CST) 75 percent of the time the TB Gantry Crane fails, yet failure of the TB Gantry Crane does not fail the Turbine Building. Also, based on the information available to the staff as part of its audit, failure of the CST independent of the failure of the TB Class I does not appear to have been accounted for in the SPRA (i.e., for the 50 percent of the time TB Class III fails and does not interact with TB Class I). During the audit the licensee explained that the SPRA fragility team and an independent reviewer qualitatively judged the correlated failure probability to be between 50 and 100 percent with a best estimate of 75 percent based on 1) the crane girder extends outside the footprint of the turbine building but only partially over the CST	

and 2) failure of the crane is expected to be in a direction not toward the CST. It is the NRC staff's assessment that this assumption is a key source of uncertainty, but that it is reasonable to assume that the TB Gantry Crane will not fail the CST 100 percent of the time because of the location of the CST with respect to the TB Gantry Crane. The licensee also provided the results of a sensitivity analysis that assumed guaranteed failure of the CST if the TB Gantry Crane failed. This analysis showed a small increase in SCDF and SLERF of 8 percent and 6 percent, respectively, and no significant changes to the importance analysis results for the top risk contributors. Based on the results of this sensitivity analysis, the NRC staff finds that the assumption that collapse of the TB Gantry Crane fails the CST 75 percent of the time does not impact the staff's conclusions regarding the submittal because the dominant risk contributors and the staff's decisionmaking for this submittal are not significantly impacted by the assumption.

With regard to the assumption that failure of the CST independent of the failure of TB Class I is not accounted for in the SPRA (i.e., for the 50 percent of the time TB Class III fails and does not interact with TB Class I – see the NRC staff's evaluation of this assumption under Topic 14 for F&O 24-8), the licensee explained during the audit that failures of the TB Gantry Crane and TB Class III are highly correlated because failure of the TB Gantry Crane could cause a cascading failure of TB Class III and because the fragilities for the TB Gantry Crane and TB Class III are similar. For these reasons, the TB Gantry Crane fragility is used as a surrogate for the composite fragility to represent the high degree of correlation between the crane and TB Class III fragilities. The licensee further explained that because of this correlation it is unrealistic to assume the CST fails 100 percent of the time when the TB Class III fails and assume the CST fails 75 percent of the time when the TB Gantry Crane fails because it would result in an excessive total CST failure probability. The NRC staff's review determined that there is a defensible basis for the correlation between the failure of the TB Gantry Crane and TB Class III.

According to Tables 5.4-2 and 5.5-2 of the SPRA submittal the dominant risk contributor for both SCDF and SLERF is SF-TB-CLASS-3-POUND, which represents the failure of the TB Class III due to seismic-induced pounding with the RAB that induces cracking and splitting of the mezzanine floor slab resulting in loss of structural integrity. During the audit the NRC staff requested the licensee to explain how the impact of this seismic-induced pounding on safety-important SSCs located in the RAB (e.g., control room, safeguards room, cable spreading rooms, Heating Ventilation and Air Conditioning [HVAC] room, direct current power system, and component cooling water storage tank) was accounted for in the SPRA. The licensee explained that the impact loads delivered to the RAB was considered as part of the fragility assessment of the equipment within the RAB and, based on a qualitative assessment, it was determined that the associated fragilities would not be significantly affected because:

- building impacts result in high frequency loads (i.e., much higher than 20 Hertz) imparted into each structure and, as discussed in the SPID (Ref. EPRI-SPID, 2012), high frequency shock loads such as those generated from building impact only have the potential to significantly affect components that are subject to intermittent states while SSCs that have strain- or stress-based potential failure modes are generally not affected by high frequency shock loads,

- the RAB is a reinforced concrete structure that was designed as a Class I building and the shock loads between these two structures will be more significant in the Class III TB and much more reduced within the comparatively rugged and heavier RAB, and

- building impact loads that occur at one floor level are filtered for the floors that are lower or higher in elevation away from the impact floor location.

The licensee conducted a sensitivity analysis to assess the effects of reducing the fragilities of components on the second floor of the RAB where the impact occurs and which have the potential for intermittent states (e.g., relays, switches, and breakers). The licensee reduced the fragilities of these types of components by a factor of two. The results of this analysis showed less than 1 percent increase in SCDF and SLERF, and no significant change in the associated top risk contributors from the importance analysis.

The staff finds that quantitatively accounting for the impact on SSCs in the RAB due to pounding with the TB Class III will not change the NRC staff's conclusions regarding the submittal because (1) the RAB is heavier than the TB Class III by more than a factor of 3.5, (2) the RAB is a significantly more robust structure (reinforced concrete) than the TB Class III (steel-framed), (3) the licensee's seismic response analysis shows that the lateral displacement of the RAB for the GMRS input motions is about 10 percent of the TB Class III displacement at the time of maximum relative displacement, (4) the RAB moves as a rigid body with practically no amplification at elevations above the foundations and the pounding impact will be attenuated at distances away from the pounding impact area, (5) the pounding impact results in high frequency loads, and (6) the licensee's sensitivity analysis of the potential impact on the fragilities of SSCs sensitive to high frequency shock loads shows negligible impact on the results and therefore, the decisionmaking for this submittal.

During the audit, the licensee explained that the SPRA credits ex-control room operator actions after failure of TB Class III and liquefaction-induced settlement or spreading. The licensee provided justification for such actions stating that (1) there would be sufficient number of operators that are not working or stationed in the TB to feasibly perform credited actions even in the event of operator fatalities due to failure of the TB, and (2) there were multiple pathways other than through the TB or liquefaction-induced settlement or spreading areas that operators can take to reach the action locations. The staff's review notes that the licensee's response does not discuss the diversion of resources due to fatalities and the need for retrieval of personnel. Further, it is unclear to the NRC staff how operators will have advance knowledge of liquefaction and lateral spreading 'zones' and the whether the impact of such field decisions on the human error probabilities were included. However, Table 5.7-1 of the submittal provides the results of Sensitivity Case HR-2b where all HEPs and JHEPS were set to their 95 percentile values, which showed that results were not impacted (increase of SCDF of about two percent and increase of SLERF of 0.5 percent). Tables 5.4-3 and 5.5-3 do not identify operator actions as dominant risk contributors. As a result, the NRC staff finds that credit for operator actions in the SPRA following failure of the TB Class III and liquefaction-induced settlement or spreading does not change the NRC staff's decisionmaking on the results of the SPRA provided in this submittal. The impact of uncertainty from human actions on the proposed modification is discussed in Enclosure 2.

During the audit, the licensee provided the results of a sensitivity analysis in which the %G02 and %G03 seismic hazard bins were divided into three bins to provide a better estimation of the contribution to seismic risk from accelerations at and below the SSE or

<table>
<tr><td colspan="2">0.2g. The results showed that the contribution to SCDF from these earthquakes is approximately 4.3 percent. In addition, the licensee explained that the Robinson Dam fragility was incorrectly reported in the SPRA submittal, provided a corrected fragility in its supplement (ADAMS Accession No. ML20092E957, non-public), and explained that the correct value was used in the SPRA used for the submittal. Therefore, the results reported in the submittal are not impacted by the change (i.e., the change was only a documentation error).

The NRC staff reviewed the licensee's disposition to finding-level F&Os against HLRs SHA-J (F&Os 26-3, 26-4, 26-5, and 26-6), SFR-F (F&Os 28-4, 29-3, 29-4, 29-6, 29-7, and 2-2), and SPR-F (F&Os 19-2, 24-1, 24-6, 24-7, 24-13, 24-20, and 25-6). These F&Os were against SRs related to documentation. The NRC staff review determined that the licensee's disposition was adequate for this submittal.

Deviation(s) or deficiency(ies) and Resolution: None

Consequence(s): N/A</td></tr>
<tr><td>The NRC staff concludes:

• The licensee's documentation meets the intent of the SPID guidance. The documentation requirements in the Code Case Standard can be found in HLR-SHA-J, HLR-SFR-F, and HLR-SPR-F.

• The licensee's documentation does not meet the intent of the SPID guidance but is acceptable on another justified basis.</td><td>YES

N/A</td></tr>
</table>

Topic 16: Review of Plant Modifications and Licensee Actions, If Any

The licensee: • identified modifications necessary to achieve seismic risk improvements • provided a schedule to implement such modifications (if any), consistent with the intent of the guidance • provided Regulatory Commitment to complete modifications • provided Regulatory Commitment to report completion of modifications.	YES YES YES YES
Plant will: • complete modifications by • report completion of modifications by	12/31/2021 1/31/2022
Notes from the Reviewer: On June 19, 2020 (ADAMS Accession No. ML20171A761), the licensee supplemented its SPRA submittal with regulatory commitments to complete four (4) plant modifications. The licensee also committed to report completion of plant modifications. The supplement also identified interim actions that the licensee will implement. Refer to Enclosure 2 for the detailed screening evaluation. Deviation(s) or Deficiency(ies), and Resolution: Refer to Enclosure 2 for the detailed screening evaluation. Consequences: Refer to Enclosure 2 for the detailed screening evaluation.	
The NRC staff concludes that the licensee: • identified plant modifications necessary to achieve the appropriate risk profile • provided a schedule to implement the modifications (if any) with appropriate consideration of plant risk and outage scheduling	YES YES

REFERENCES

ASME/ANS Addendum A, 2009: Standard ASME/ANS RA-Sa-2009, Addenda A to ASME/ANS RA-S-2008, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," American Society of Mechanical Engineers and American Nuclear Society, 2009

ASME/ANS Addendum B, 2013: Standard ASME/ANS RA-Sb-2013, Addenda B to ASME/ANS RA-S-2008, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," American Society of Mechanical Engineers and American Nuclear Society, 2013

ASME/ANS, 2017: Case 1 for Standard ASME/ANS RA-Sb-2013 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," American Society of Mechanical Engineers and American Nuclear Society, 2017

EPRI-SPID, 2012: "Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," Electric Power Research Institute, EPRI report 1025287, November 2012 (ADAMS Accession No. ML12333A170) as endorsed by the NRC in a February 15, 2013, letter (ADAMS Accession No. ML12319A074).

NEI, 2012: NEI 12-13 "External Hazards PRA Peer Review Process Guidelines," Nuclear Energy Institute, August 2012

NRC, 2007: "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Regulatory Guide 1.200, Revision 1, January 2007 (ADAMS Accession No. ML070240001)

NRC, 2012: "U.S. Nuclear Regulatory Commission Comments on NEI 12-13, 'External Hazards PRA Peer Review Process Guidelines' Dated August 2012," NRC letter to Nuclear Energy Institute, November 16, 2012 (ADAMS Accession No. ML12321A280)

NRC, 2017: "NRC Staff Review Guidance for Seismic PRA Submittals and Technical Review Checklist," February 10, 2017 (ADAMS Accession No. ML17041A342)

NRC, 2017a: Rosenberg, Stacey L., U.S. Nuclear Regulatory Commission, memo to Risk Informed Steering Committee, "U.S. Nuclear Regulatory Commission Staff Expectations for an Industry Facts and Observations Independent Assessment Process," dated May 1, 2017 (ADAMS Accession No. ML17121A271).

NRC, 2017b: Giitter, Joseph, and Ross-Lee, Mary Jane, U.S. Nuclear Regulatory Commission, letter to Krueger, Greg, Nuclear Energy Institute, "U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 07-12, and 12-13, Close-Out of Facts and Observations (F&Os)," dated May 3, 2017 (ADAMS Accession No. ML17079A427).

NRC, 2018: "US Nuclear Regulatory Commission Acceptance of ASME/ANS RA-S Case 1," NRC letter from Brian Thomas (NRC Standards Executive) to C.R. Grantom and R.J. Budnitz, March 12, 2018 (ADAMS Accession No. ML18017A963)

NRC, 2018a: "US Nuclear Regulatory Commission Acceptance of Nuclear Energy Institute (NEI) Guidance NEI 12-13, "External Hazards PRA Peer Review Process Guidelines" (August 2012" NRC letter to Nuclear Energy Institute, March 7, 2018 (ADAMS Accession No. ML18025C025)

NRC, 2018b: "US Nuclear Regulatory Commission Acceptance of Nuclear Energy Institute (NEI) Guidance NEI 12-13, "External Hazards PRA Peer Review Process Guidelines" (August 2102," tabular compilation of NRC staff comments, appended to (NRC, 2018a), (ADAMS Accession No. ML18025C024)

NRC Staff SPRA Submittal Detailed Screening Evaluation

Introduction

The H.B. Robinson Steam Electric Plant, Unit No. 2 (Robinson) Seismic Probabilistic Risk Assessment (SPRA) report (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML20084P290 (public) and ML19346E204 (non-public)), supplemented by letters dated March 31, 2020 (ADAMS Accession Nos. ML20094K843 (public) and ML20092E957 (non-public)) and June 19, 2020 (ADAMS Accession No. ML20171A761), indicates that the point estimate seismic core damage frequency (SCDF) is 9.27E-05/reactor-year (/rx-yr) and the point estimate seismic large early release frequency (SLERF) is 2.02E-05/rx-yr. The mean SCDF and SLERF values are not provided in the SPRA report. However, these values, mean SCDF of 1.3E-04/rx-yr and mean SLERF of 2.5E-05/rx-yr, were provided by the licensee during the audit. The NRC staff compared these values against the guidance in NRC staff memorandum dated August 29, 2017 (ADAMS Accession No. ML17146A200), titled, "Guidance for Determination of Appropriate Regulatory Action Based on Seismic Probabilistic Risk Assessment Submittals in Response to Near Term Task Force Recommendation 2.1: Seismic" (hereafter referred to as the SPRA Screening Guidance), which establishes a process the NRC staff uses to develop a recommendation on whether the plant should move forward as a Group 1, 2, or 3 plant.[1]

The SPRA Screening Guidance is based on NUREG/BR-0058, Revision 4, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," (ADAMS Accession No. ML042820192), NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook," (ADAMS Accession No. ML050190193), and NUREG-1409, "Backfitting Guidelines," (ADAMS Accession No. ML032230247), as informed by Nuclear Energy Institute (NEI) 05-01, "Severe Accident Mitigation Alternatives (SAMA) Analysis Guidance Document" (ADAMS Accession No. ML060530203). In order to determine the significance of proposed modifications in terms of safety improvement, NUREG/BR-0058 uses screening criteria based on the estimated reduction in core damage frequency, as well as the conditional probability of early containment failure or bypass. Per NUREG/BR-0058, the conditional probability of early containment failure or bypass is a measure of containment performance and the purpose of its inclusion in the screening criteria is to achieve a measure of balance between accident prevention and mitigation. The NUREG/BR-0058 uses a screening criterion of 0.1 or greater for conditional probability of early containment failure or bypass. In the context of the SPRA reviews, the staff guidance uses SCDF and SLERF as the screening criteria where SLERF is directly related to the conditional probability of early containment failure or bypass. Following NUREG/BR-0058, the threshold for the screening criterion in the staff guidance for SLERF is (1.0E-6/rx-yr), or 0.1 times the threshold for the screening criterion for SCDF (1.0E-5/rx-yr).

The NRC staff found that because the SCDF and SLERF for Robinson were above the initial screening values of 1.0E-5/rx-yr and 1.0E-6/rx-yr, respectively, a detailed screening following the SPRA Screening Guidance was performed.

The SPRA submittal proposed a plant modification that could potentially provide substantial safety enhancement and reduce the risk associated with the reevaluated seismic hazard. As

[1] The groups are defined as follows: regulatory action not warranted (termed Group 1), regulatory action should be considered (termed Group 2), and more thorough analysis is needed to determine if regulatory action should be considered (termed Group 3).

part of the audit process, the staff requested supporting details to understand the safety enhancement expected from the proposed modification and to understand the identification as well as implementation of interim actions during the completion of the proposed modifications. The staff received details of the modifications as part of the audit process. In its letter dated June 19, 2020, the licensee proposed regulatory commitments to complete the four (4) permanent plant modifications. The letter also identified interim actions that will be taken by the licensee for the duration that the proposed permanent modifications are completed.

Detailed Screening

The NRC staff did not identify concerns that would require immediate action to avoid undue risk to public health and safety. A discussion of the staff's determination that an immediate safety concern does not exist based on the results from the SPRA is provided in Enclosure 4. In addition, there were no issues identified as non-compliances with the Robinson license, or the rules and orders of the Commission.

The detailed screening uses information provided in the Robinson SPRA report, particularly the importance measures, SCDF, and SLERF, as well as other information described below, to establish threshold and target values that are used to identify areas where potential cost-justified substantial safety improvements might be identified. The detailed screening process makes several simplifying assumptions, similar to a Phase 1 SAMA analysis (NEI 05-01, ADAMS Accession No. ML060530203) used for license renewal applications. The detailed screening process uses risk importance values as defined in NUREG/CR-3385, "Measures of Risk Importance and Their Applications" (ADAMS Accession No. ML071690031). The NUREG/CR-3385 states that the risk reduction worth (RRW) importance value is useful for prioritizing feature improvements that can most reduce the risk. The Robinson SPRA report provides Fussell-Vesely (F-V) importance values, which were converted to RRW values by the NRC staff for this screening evaluation using a standard relationship formulation. Data used to develop the maximum averted cost-risk (MACR) for the severe accident mitigation alternative (SAMA) analysis provided in the *Docket Number 50-261 Robinson Nuclear Plant Unit 2 Carolina Power & Light Company Facility Operating License DPR-23 – License Renewal Application*, dated June 14, 2002 (ADAMS Accession Nos. ML021690663, ML021690656, ML021690696, and ML021700129), and the *Proposed Amendment to Technical Specification 5.5.16 for the Adoption of Option B of 10 CFR 50, Appendix J for Type B and Type C Testing and the Permanent Change in 10 CFR 50, Appendix J, Integrated Leak Rate Test Interval and Type C Leak Rate Testing*, dated November 19, 2015 (ADAMS Accession No. ML15323A085), were used to calculate the RRW threshold. For this analysis, the NRC staff determined the RRW threshold from the SCDF-based MACR to be 1.031. The MACR calculation includes estimation of offsite exposures and offsite property damage, which captures the impact of SLERF. Therefore, separate SLERF-based MACR calculations were not performed. The target RRW values (as defined by the SPRA screening guidance) based on the mean and 95th percentile SCDF and SLERF were calculated by the NRC staff to be between 1.01 and 1.08.

Section 5 of the Robinson SPRA report included tables listing and describing the events and fragility groups that are the most significant seismic failure contributors to SCDF and SLERF. Similar tables were also provided for the most significant contributors due to random failure of SSCs and due to failure of operator actions. The descriptions of the significant contributors included the F-V for each. The NRC staff utilized the F-V values presented in the supplement to calculate the RRW and the contribution to SCDF or SLERF of each contributor. The results are provided in Table 1 for the SCDF contributors and Table 2 for the SLERF contributors. The listed seismic-induced failures that contribute to SCDF and SLERF have an RRW greater than about 1.02. These tables provide the following information by column: (1) Description of the component, (2) Failure Mode, (3) RRW, and (4) maximum SCDF or SLERF reduction (MCR) from eliminating the failure. Three single SPRA model elements or contributors exceeded the mean target RRW for SCDF and seven seismically-induced failures exceeded the mean target RRW for SLERF.

The NRC staff considered both single and combinations of basic events in accordance with the SPRA Screening Guidance. It is not the intent of that aspect of the guidance to aggregate several disparate basic events that individually have RRW values close to the mean target RRW. The total SCDF of the SPRA model seismically-failed elements identified in Table 1 is over 1.2E-04/rx-yr. The total SLERF of the SPRA model seismically-failed elements identified in Table 2 is about 2.0E-05/rx-yr. For both SCDF and SLERF, the top contributor is failure of Turbine Building Class 3 due to seismic-induced pounding of the Reactor Auxiliary Building (RAB) and the Turbine Building, which fails auxiliary feedwater. According to Tables 5.4-2 and 5.5-2 of the submittal, this failure is a contributor to 18 of the top 25 SCDF and SLERF cutsets. Furthermore, review of Tables 1 and 2 reveals that there are also several other failures that are common top contributors to both SCDF and SLERF (namely, Turbine Building Gantry Crane, Robinson Dam, Steam-Driven Auxiliary Feedwater pump, and diesel fuel oil tank piping). According to Tables 5.4-2 and 5.5-2 of the submittal, these failures contribute, in aggregate, to all of the top 25 SCDF and SLERF cutsets.

Duke Energy Progress, LLC (Duke, the licensee), in Table 6-1 of its SPRA submittal, identified a plant modification to address the high seismic risk. According to the licensee, the plant modification would provide auxiliary feed water (AFW) using a modified FLEX strategy and could potentially reduce SCDF and SLERF by approximately 40 percent and 30 percent, respectively. However, the submittal did not provide details of the modification which would support the staff's understanding of its objectives, feasibility, and effectiveness. Therefore, the staff sought and evaluated additional information from the licensee on (1) the rationale for selecting the proposed modifications over the other options explored by the licensee, and (2) the details of the proposed modification, including assumptions and sources of uncertainty.

In its audit responses, the licensee explained sensitivity analyses were performed to estimate risk reduction from different modification options and combinations of modifications. The licensee explained that there are several challenges with providing an equivalent level of defense-in-depth and safety margin for a plant in which the original design bases did not address the seismic hazard being evaluated in the SPRA. The licensee provided information about feasibility issues with different potential modifications including those for the Class III TB. These included operational risk from any excavation, risk of damaging the structure during implementation, and interferences from existing systems and structural features for bracing and trimming the Class III TB.

The licensee also performed scoping of the modifications in terms of physical and economic feasibility, which included a cost estimate based on the initial modification definition and the likely range of costs associated with each modification. The licensee provided the cost estimate for each modification explored by the licensee, which were classified as "low" which corresponded to cost estimate of less than \$2 million; "medium" which corresponded to cost estimate between \$2 and \$10 million; and "high" which corresponded to cost estimate greater than \$10 million. The licensee explained that the Class III TB structural modification and other dominant risk contributor modifications were not being pursued based on the comparison of the modification cost estimates to the proposed modification strategy cost estimate. The licensee selected the proposed modifications based on the comparison of a combination of the cost estimate and risk reduction from completion of the modifications addressing the dominant risk contributors against the corresponding information for the proposed modifications.

The licensee provided details of four (4) permanent modifications that were collectively proposed to address the impacts demonstrated by the SPRA. These proposed permanent modifications are:

- Reconfiguration of the existing pre-staged discharge hose from the FLEX AFW pumps to include a branch which routes to the existing FLEX AFW connection (AFW-165) located inside the Motor-Driven AFW Pump Room. The current configuration connects to the FLEX AFW connection near the Steam-Driven AFW Pump (AFW-166). The new configuration will feature isolation of the two branches such that the branch which connects to AFW-165 will not be vulnerable to a potential failure of the Turbine Building Class III structure. This modification provides an AFW flow path that is not vulnerable to seismic failure of the Class III TB. The completion time for this modification is July 30, 2021.

- Installation of isolation valves, hose connections and pipe for the existing deepwell pumps A, B and C, to protect the above ground portion of the piping from the seismic event. This modification will address the liquefaction failure mode of supply from deepwell pumps, A, B and C. The completion time for this modification is May 29, 2021.

- Routing of hose and/or pipe from each of the existing deepwell pumps A, B and C, to the suction header for the existing FLEX AFW pumps. The shortest and most seismically preferable routes will be utilized, as best possible. For high traffic areas, hose will be permanently staged in lieu of permanent installation. To mitigate probable damage, the routes will include sections to allow for easy disassembly and connection of new sections. This modification supplements proposed permanent modification discussed in the second item above. The completion time for this modification is June 30, 2021.

- Install a new deepwell pump E to provide suction directly to the existing FLEX AFW pumps. The construction of the new deepwell pump will include similar seismically rugged features as existing deepwell pump D. The new deepwell pump E and appurtenances will be located, or otherwise protected, to prevent adverse seismic impact from other structures or equipment. The new deepwell pump will be located and configured to support time sensitive operator actions per the guidance of NEI 12-06, Revision 2, “Diverse and Flexible Coping Strategies (FLEX) Implementation Guide.” Power for the new deepwell pump will be provided by normal AC power, and a new diesel generator, similar to existing deepwell pumps A, B and C. This modification will address the high failure probability of the existing tanks which provide suction to the FLEX AFW pumps. The completion time for this modification is December 31, 2021.

Note that the above modifications are not included in the SPRA in the submittal and therefore, are not listed in Tables 1 and 2 of this enclosure.

The licensee provided a regulatory commitment in a supplemental letter dated June 19, 2020, to implement the four (4) permanent plant modifications described above. The licensee stated that a sensitivity study estimated the combined risk reduction from all the above mentioned four (4) plant modifications to be approximately 5.6E-05/rx-yr and 7.2E-06/rx-yr for SCDF and SLERF, respectively. The results provided by the licensee appear to be based on point estimates and not mean values (i.e., including uncertainties). However, the NRC staff finds the sensitivity's estimate of the direction and approximate magnitude of the impact from the proposed modification applicable to the mean values.

The sensitivity evaluation performed by the licensee approximated the failure probability of the water source using the Robinson Dam fragility and a random failure of a low-pressure FLEX pump and manual valves in the flow path. The licensee stated that the remaining SSCs required for the modifications were assumed to have no failure probability due to seismic shaking in the modification sensitivity and seismic interaction failure (failures of enclosures or structures housing the equipment) was not included in the sensitivity. The licensee stated that the SSCs for the proposed modification, other than the water source(s), would be designed such that seismic shaking and lateral spreading will not contribute significantly to the failure probability of the function during risk-significant seismic events. The licensee stated that the SSCs would be in an area that precludes seismic interactions.

The licensee stated that the proposed modification will depend on operator actions and the sensitivity evaluation used existing FLEX-related human failure events (HFEs) and corresponding human error probabilities (HEPs) as surrogates. The selected HFEs were deemed by the licensee to be similar, though not identical, to the need for an operator to manually start a pre-staged system that provides AFW to the SGs with provision for long-term action to ensure sufficient inventories of water, power, and fuel. The licensee stated that the uncertainty in the HEPs would directly impact the importance of the modification. In its supplemental letter dated June 19, 2020, the licensee stated that the licensee will ensure that operator actions associated with the permanent plant modifications are validated in accordance with NEI 12-06, Appendix E, Revision 2, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide" (ADAMS Accession No. ML16005A625).

The NRC staff's review of the proposed permanent modifications, aided by a discussion on the modifications with the licensee during the audit, concludes that the proposed four (4) permanent modifications collectively appropriately address the impacts of the dominant risk contributors demonstrated by the SPRA. While the modifications do not directly address the dominant risk contributors, they address the impact of the failure of those contributors by providing alternative means to supply AFW to the SGs. The alternative means to provide AFW to the SGs would achieve the key safety function of decay heat removal. Further, the NRC staff concludes that the compressed schedule for completion of the proposed permanent modifications is positive because it achieves the safety benefits earlier.

In its supplement dated June 19, 2020, the licensee identified interim actions that will be taken by the licensee for the duration that the proposed permanent modifications are completed. These interim actions would provide defense-in-depth and support mitigation of dominant sequences while the proposed permanent modifications are implemented. The NRC staff's review of the proposed permanent modifications, aided by a discussion on the modifications with the licensee during the audit, concludes that the interim actions are appropriate to manage the incremental risk demonstrated by the SPRA during completion of the permanent modifications.

The NRC's backfitting experts evaluated the available information on the proposed permanent modifications and determined that, using Section 2 of NUREG/BR-0058 as a guide, the risk reduction falls within a band that provides a substantial additional overall protection such that whether to proceed to perform a detailed backfit analysis is a management (i.e., SMRP) decision.

Based on the technical evaluation of the insights from Robinson SPRA submittal (Enclosure 4), the detailed screening evaluation described above, and the available information regarding the proposed plant modification and interim actions, the staff concludes that the proposed modifications provide substantial additional overall protection. The SMRP credited the regulatory commitments provided by the licensee in supplement dated June 19, 2020, and decided not to pursue further regulatory action for this SPRA submittal because:

1. the proposed permanent modifications address the impacts of the dominant risk contributors demonstrated by the SPRA;

2. the completion schedule for the proposed permanent modifications will result in the safety enhancements being achieved with a relatively short turnaround;

3. the interim actions proposed by the licensee provide defense-in-depth and support mitigation of dominant sequences while the proposed permanent modifications are implemented; and

4. the NRC staff can inspect the implementation of the proposed interim actions as well as the proposed permanent modifications; and

5. the modifications are very unlikely to be undone and will be incorporated into existing FLEX and EDMG programs (i.e., incorporated the changes into the programs that are subject to 10 CFR 50.155 requirements).

The NRC staff reviewed the results of the Individual Plant Examination of External Events (IPEEE) and SAMA analyses previously completed for Robinson to identify additional substantial safety improvements that would be cost justified. No additional potential modifications were identified based on the IPEEE and SAMA information and the staff's conclusions regarding this SPRA submittal were not impacted.

<u>Conclusion of Detailed Screening</u>

Based on the analysis of the submittal and supplemental information discussed in detail above:

- The staff did not identify potential modifications necessary for adequate protection or compliance with existing regulations;

- The SMRP credited the four (4) proposed permanent modifications submitted as regulatory commitments by the licensee in supplement dated June 19, 2020, as well as the interim actions described in the same supplement in its decision to classify Robinson as a Group 1 plant for this SPRA submittal (i.e., not pursue further regulatory action for this SPRA submittal).

- The staff did not identify any additional cost-justified substantial safety improvement either based on reduction of SCDF and/or SLERF or based on consideration of containment performance, as described in NUREG/BR-0058.

Table 1. Importance Analysis Results of Top Contributors to SCDF

Description	Failure Mode	RRW	MCR (/yr)
Seismically-failed SSCs			
Turbine Building Class 3 - Pounding-induced cracking	RAB pounding induced cracking and splitting of the mezzanine floor slab resulting in loss of structural integrity.	1.777	5.86E-05
DFOST Liquefaction-Induced Settlement	Failure of EDG-B pipe at RAB penetration.	1.144	1.68E-05
Turbine Building Gantry Crane	Failure of A-frame anchor bolts.	1.134	1.58E-05
SDAFW Liquefaction-Induced Settlement	Failure of AFW discharge piping where the 4 in. diameter pipe meets the 6x4 reducing elbow.	1.075	9.39E-06
Robinson Dam Liquefaction-Induced Settlement	Uncontrolled release of the reservoir given the performance of the dam during and after strong shaking.	1.072	9.02E-06
Underground Cable Trays at Intake Liquefaction-Induced Settlement	Failure at Intake Structure.	1.027	3.47E-06
Relay Chatter - DG-A,B-AUX-PNL_Barksdale Controls_D2T-M18SS	Relay malfunction due to earthquake shaking.	1.019	2.51E-06
Relay Chatter - DG-A,B-ENG-PNL_Barksdale Controls_D2T-M80SS	Relay malfunction due to earthquake shaking.	1.019	2.51E-06
North Header Intake Mech 3 Liquefaction-Induced Settlement	Failure of North Header - Mechanism 3.	1.017	2.28E-06
Seismic-Induced Loss of Offsite Power	Functional failure.	1.016	2.06E-06
Randomly-failed SSCs			
TURBINE-DRIVEN PUMP FAILS TO RUN	Not Applicable	1.007	9.78E-07
CONDENSER WATERBOX INLET MOTOR PUMP CIW-L FAILS TO START (FLEX)	Not Applicable	1.006	7.64E-07
TURBINE-DRIVEN PUMP FAILS TO START	Not Applicable	1.004	5.76E-07
AFW TD PUMP TRAIN C UNAVAILABLE	Not Applicable	1.003	4.02E-07
PORTABLE LAKE PUMP FAILS TO START (FLEX)	Not Applicable	1.003	3.48E-07
Human Failure Events			
Failure to align and start pre-staged pumps for SG makeup - Condenser Inlet Waterbox (FLEX)	Not Applicable	1.007	9.11E-07
Failure to align and start portable pumps to lake for long-term water source - SG makeup (FLEX)	Not Applicable	1.004	5.63E-07
Dependent HEP for OPER-35,OPER-68,OPER-18B-S1,OPER-64,OPER-01S	Not Applicable	1.003	4.42E-07
OPERATOR FAILS TO TRANSFER POWER TO DEEPWELL PUMP DIESEL	Not Applicable	1.003	4.42E-07
OPERATOR FAILS TO ALIGN AFW PUMP C	Not Applicable	1.002	2.41E-07

Table 2. Importance Analysis Results of Top Contributors to SLERF

Description	Failure Mode	RRW	MCR (/yr)
Seismically-failed SSCs			
Turbine Building Class 3 - Pounding-induced cracking	RAB pounding induced cracking and splitting of the mezzanine floor slab resulting in loss of structural integrity.	1.420	7.43E-06
Reactor Containment Building (RCB)	Nonlinear rotation of the socketed like connection at the underside of the basemat due to lateral structural displacement.	1.134	2.97E-06
DFOST Liquefaction-Induced Settlement	Failure of EDG-B pipe at RAB penetration.	1.087	2.02E-06
Turbine Building Gantry Crane	Failure of A-frame anchor bolts.	1.083	1.93E-06
Liquefaction-Induced Lateral Spreading Distance Category 2 (DC2)	Liquefaction-Induced Lateral Spreading.	1.053	1.26E-06
Robinson Dam Liquefaction-Induced Settlement	Uncontrolled release of the reservoir given the performance of the dam during and after strong shaking.	1.049	1.18E-06
SDAFW Liquefaction-Induced Settlement	Failure of AFW discharge piping where the 4 in. diameter pipe meets the 6x4 reducing elbow.	1.044	1.05E-06
Reactor Auxiliary Building (RAB)	Flexural failure of piles.	1.028	6.88E-07
Underground Cable Trays at Intake Liquefaction-Induced Settlement	Failure at Intake Structure.	1.025	6.15E-07
Deepwell Pump D Liquefaction-Induced Settlement	Bending failure of a bolted flange connection.	1.021	5.20E-07
Randomly-failed SSCs			
PERSONNEL HATCH INNER DOOR GASKETS FAILS	Not Applicable	1.011	2.74E-07
TURBINE-DRIVEN PUMP FAILS TO RUN	Not Applicable	1.003	8.03E-08
FAILURE OF PERSONNEL HATCH DOOR SEALS	Not Applicable	1.003	8.03E-08
ELECTRICAL PENETRATIONS FAILS OPEN	Not Applicable	1.003	8.03E-08
PRE-INITIATOR IMPORTANCE SCOPING EVENT FOR CI - P-44/45 BYPASS LEFT OPEN	Not Applicable	1.003	7.28E-08
TURBINE-DRIVEN PUMP FAILS TO START	Not Applicable	1.002	4.52E-08
PRE-INITIATOR IMPORTANCE SCOPING EVENT FOR CI - PERSN HATCHES LEFT OPEN	Not Applicable	1.001	3.51E-08
Human Failure Events			
Failure to align and start pre-staged pumps for SG makeup - Condenser Inlet Waterbox (FLEX)	Not Applicable	1.003	7.53E-08
Failure to align and start portable pumps to lake for long-term water source - SG makeup (FLEX)	Not Applicable	1.002	4.02E-08
Failure to supply AFW with SW	Not Applicable	1.002	3.77E-08

AUDIT SUMMARY BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO

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

SUBMITTAL OF SEISMIC PROBABILISTIC RISK ASSESSMENT ASSOCIATED WITH

REEVALUATED SEISMIC HAZARD IMPLEMENTATION OF THE

NEAR-TERM TASK FORCE RECOMMENDATION 2.1: SEISMIC

(EPID NO. L-2019-JLD-0022)

BACKGROUND AND AUDIT BASIS

By letter dated March 12, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12053A340), the U.S. Nuclear Regulatory Commission (NRC) issued a request for information pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.54(f) (hereafter referred to as the 50.54(f) letter). Enclosure 1 to the 50.54(f) letter requested that licensees reevaluate the seismic hazards for their sites using present-day methods and regulatory guidance used by the NRC staff when reviewing applications for early site permits and combined licenses.

By letter dated October 27, 2015 (ADAMS Accession No. ML15194A015), the NRC made a determination of which licensees were to perform: (1) a Seismic Probabilistic Risk Assessment (SPRA), (2) limited scope evaluations, or (3) no further actions based on a comparison of the reevaluated seismic hazard and the site's design-basis earthquake. (Note: Some plant-specific changes regarding whether an SPRA was needed or limited scope evaluations were needed at certain sites have occurred since the issuance of the October 27, 2015, letter).

By letter dated July 6, 2017 (ADAMS Accession No. ML17177A446), the NRC issued a generic audit plan and entered into the audit process described in Office Instruction LIC-111, "Regulatory Audits," dated December 29, 2008 (ADAMS Accession No. ML082900195), to assist in the timely and efficient closure of activities associated with the letter issued pursuant to 10 CFR Part 50, Section 50.54(f). The list of applicable licensees in Enclosure 1 of the July 6, 2017, letter included Duke Energy Progress, LLC (Duke, the licensee) as the licensee for the HB Robinson Steam Electric Plant, Unit No. 2 (Robinson) site.

REGULATORY AUDIT SCOPE AND METHODOLOGY

The areas of focus for the regulatory audit are the information contained in the SPRA submittal (ADAMS Accession Nos. ML20084P290 (public) and ML19346E204 (non-public)), supplemented by letters dated March 31, 2020 (ADAMS Accession Nos. ML20094K843 (public) and ML20092E957 (non-public)) and June 19, 2020 (ADAMS Accession No. ML20171A761) and all associated and relevant supporting documentation used in the development of the SPRA submittal including, but not limited to, methodology, process information, calculations, computer models, etc.

AUDIT ACTIVITIES

The NRC staff developed questions to verify information in the licensee's submittals and to gain understanding of non-docketed information that supports the docketed SPRA submittals. The staff's clarification questions and request for supporting documents (ADAMS Accession Nos. ML19365A046, ML20009C092, ML20015A189, ML20027C204, ML20087N663 (public and ML20087N623 (non-public)), ML20085F931, and ML20086L533 (public and ML20086L226 (non-public)), were sent to the licensee to support the audit. The licensee provided clarifying information in the following areas:

- Information describing the Internal Events PRA acceptability, resolution for finding level facts and observations (F&Os) and their effect over the SPRA.
- Discussion on fragilities for certain structures, systems, and components (SSCs).
- Uncertainties on fragility calculations for different SSCs.
- Modeling of seismic event trees used in the SPRA and selection of failure sequences.
- Overall liquefaction analysis for the site, including the Lake Robinson Dam.
- Refinements made to the SPRA to achieve realism.
- Failure probabilities and uncertainties associated with liquefaction risk contributors.
- Sensitivities to quantify the effect of assumptions used in the analysis.
- Supporting information about FLEX mitigation strategies and human reliability analysis.
- Clarification of risk contribution from the reevaluated seismic hazard at the design basis level.
- Impact of top risk contributors in human actions for response and mitigation.
- Evaluation of potential plant modifications (assumptions, cost, and selection criteria) that could address dominant risk contributors and re-establish defense-in-depth, safety margins and reduce SCDF and SLERF.
- Additional details about the proposed plant modification to provide auxiliary feedwater to the steam generators and resulting potential reduction of SCDF and SLERF.

The licensee's response to the questions aided in the staff's understanding of the Robinson SPRA docketed submittal. Following the review of the licensee's response and the supporting documents provided by the licensee on the eportal, the staff determined that no additional documentation or information was needed to supplement the docketed SPRA submittal.

DOCUMENTS AUDITED

- Duke Energy Calc. No. RNP-C/FLEX-0026, Attachment 17, Revision 1, "Flex Building

Liquefaction Potential at Flex -C GMRS," Amec Foster Wheeler Calc. No. 4150-GEO-031 for Robinson Unit 2, October 10, 2017.

- Duke Energy Calc. No. RNP-C/FLEX-0027, Revision 1, "Response Analysis of the Robinson Unit 2 Intake Structure: Phase 2 SPRA Project," September 16, 2019.

- Duke Energy Document RNP-F/PSA-0062, Revision 4, Appendix 21, "RNP Seismic PRA 2018 (Full Scope) and 2019 (Focused Scope) Peer Review F&O Resolution Notebook," Jensen-Hughes Document #004047-CALC-10 for Robinson Unit 2, November 26, 2019.

- Duke Energy Document RNP-F/PSA-0062, Revision 3, Appendix 15, "RNP Internal Events and Internal Flood Model F&O Closure," Jensen-Hughes Document #025085-RPT-03 for Robinson Unit 2, November 26, 2019.

- Duke Energy Calc. No. RNP-C/FLEX-0041, Revision 0, "Relay Contact Chatter Analysis," August 31, 2017.

- Duke Energy Calc. No. RNP-C/FLEX-0056, Revision 1, "Response Analysis Notebook SPRA Project," Robinson Unit 2, September 16, 2019.

- Duke Energy Calc. No. RNP-F/PSA-0131, Revision 2, "Robinson SPRA Model Notebook," Jensen-Hughes Document 004047-CALC-03 for Robinson Unit 2, December 2, 2019.

- Duke Energy Calc. No. RNP-C/FLEX-0006, Revision 4, "Geotechnical Analysis Report: Robinson Fukushima Seismic PRA," AMEC Foster Wheeler for Robinson Unit 2, May 7, 2018.

- Duke Energy Document FSG-005, Revision 0, "Initial Assessment and FLEX Equipment Staging," Robinson Unit 2.

- WEC Letter No. LTR-RAM-19-106, Revision 0, "Robinson Focused Scope Peer Review of LERF and Dam Failure Fragility," November 6, 2019.

- Duke Energy Document RNP-C/FLEX-0007, Revision 7, "Final Seismic Analysis Report: Robinson Fukushima Seismic Support," AMEC Foster Wheeler Document for Robinson Unit 2, December 17, 2018.

- Duke Energy Document RNP-C/FLEX-0031, Revision 1, "Liquefaction Settlement: Seismic Probabilistic Risk Assessment Project," SGH Document 158039-CA-055 Revision 0A for Robinson Unit 2, May 4, 2018.

- Duke Energy Document RNP-C/FLEX-0055, Revision 1, "Seismic Fragility of Pounding between the Class III Turbine Bldg and the Reactor Aux Bldg by Nonlinear Analysis," SGH Document 158039-CA-157 Revision 1 for Robinson Unit 2, September 18, 2019.

- Duke Energy Document RNP-C/FLEX-0053, Revision 1, "Seismic Fragility of the Deep Well Pump D Piping for the Effects of Liquefaction-Induced Soil Settlement," SGH Document 158039-CA-111 for Robinson Unit 2, August 20, 2018.

- Duke Energy Document RNP-C/FLEX-0062, Revision 1, “Seismic Fragility Evaluation Notebook,” SGH Document 158039-R-06 for Robinson Unit 2, September 30, 2019.

- Duke Energy Document RNP-F/PSA-0129, Revision 1, “Robinson SPRA Quantification Notebook,” Jensen-Hughes Document 004047-CALC-06 Revision 1 for Robinson Unit 2, December 2, 2019.

- Duke Energy Document RNP-C/FLEX-0050, Revision 2, “Robinson Representative Fragilities Overview,” SGH Document 158039-R-03 for Robinson Unit 2, September 30, 2019.

- WEC Document PWROG-18063-P, Revision 0, ‘’Peer Review of the Robinson Unit 2 Seismic Probabilistic Risk Assessment,” Risk Management Committee PA-RMSC-1476, March 2019.

- Duke Energy Document RNP-C/FLEX-0006, Revision 2, Attachment 9, “Dam and Foundation Soil Properties for Liquefaction,” AMEC Foster Wheeler Document for Robinson Unit 2, November 3, 2017.

- Duke Energy Document FSG-002, Revision 2, “Alternate SDAFW Suction Source,” Robinson Unit 2.

- Duke Energy Document CSD-EG-RNP-8888, Revision 0, “Flexible Response to Extended Loss of All AC Power,” Robinson Unit 2, January 17, 2017.

- Duke Energy Engineering Change Package 90625R0, “Design and Construction of the Permanent FLEX Storage Building,” Robinson Unit 2.

- Calc. No. 4150-GEO-015, Revision 2, “Liquefaction Analysis for Robinson Plant,” Amec Foster Wheeler Project No. 6468-14-4150, July 15, 2016.

- Duke Energy Document RNP-C/STRU-1376, Revision 0, “RNP Permanent FLEX Storage Building Structural Analysis,” SGT Document IND13143 for Robinson Unit 2.

- Duke Energy Document RNP-C/FLEX-0026, Revision 2, Attachment 19, “Probability of Continuous Liquefaction Layer,” AMEC Foster Wheeler Document 4150-GEO-023, Revision 0 for Robinson Unit 2, December 2, 2016.

- Duke Energy Document RNP-F/PSA-0128, Revision 1, “Robinson SPRA Uncertainty and Sensitivity Notebook,” Jensen-Hughes Document 004047-CALC-07 Revision 1 for Robinson Unit 2, December 2, 2019.

- Duke Energy Document RNP-C/FLEX-0049, Revision 0, “Robinson Site-Wide Lateral Spreading Assessment: SPRA Project,” AMEC Foster Wheeler Document 4150-GEO-024 Revision 0 for Robinson Unit 2, March 6, 2018.

- Duke Energy Document RNP-F/PSA-0018, Revision 10, “PSA Model Appendix A – System Notebooks,” Engineering Planning and Management, Inc. Document P3126-2003-01 Revision 0 for Robinson Unit 2, February 27, 2019.

- Duke Energy Document RNP-C/FLEX-0060, Revision 1, “Fragility Evaluation of Robinson Dam,” SGH Document 158039-CA-056 Revision 0 for Robinson Unit 2, September 16, 2019.
- Duke Energy Document FSG-003, Revision 0, “Alternate Feedwater,” Robinson Unit 2.
- Duke Energy Document RNP-C/FLEX-0026, Revision 0, Attachment 2, “Site-Wide Liquefaction Evaluation,” AMEC Foster Wheeler Document 4150-GEO-018 Revision 1 for Robinson Unit 2, July 26, 2016.
- Duke Energy Document RNP-F/PSA-0130, Revision 1, “Robinson SPRA Human Reliability Analysis Notebook,” Jensen-Hughes Document 004047-CALC-02 Revision 1 for Robinson Unit 2, December 2, 2019.

OPEN ITEMS AND REQUEST FOR INFORMATION

There were no open items identified by the NRC staff that required proposed closure paths and there were no requests for information discussed or planned to be issued based on the audit.

DEVIATIONS FROM AUDIT PLAN

There were no deviations from the generic audit plan dated July 6, 2017.

AUDIT CONCLUSION

The issuance of this document, containing the staff’s review of the submittals, concludes the SPRA audit process for Robinson.

RISK-INFORMED EVALUATION OF INSIGHTS FROM THE

ROBINSON SEISMIC PROBABILISTIC RISK ASSESSMENT UNDER 10 CFR 50.54(f)

1. Background

The licensee developed the reevaluated seismic hazard for its site in response to items (1) through (7) of the 50.54(f) letter and submitted the same to the NRC (ADAMS Accession No. ML15243A061). The licensee compared the re-evaluated hazard with its design basis. The comparison concluded that the ground motion response spectra (GMRS), which was developed based on the re-evaluated seismic hazard, exceeded the design basis seismic response spectrum in the 1 to 10 Hertz (Hz) range, and a seismic risk assessment was warranted. A comparison of the GMRS against the design basis Safe Shutdown Earthquake (SSE) is shown in Figure 1.

The staff assessment of the re-evaluated hazard (ADAMS Accession No. ML15280A199) concluded that the licensee conducted the hazard reevaluation using present-day methodologies and regulatory guidance, appropriately characterized the site given the information available, met the intent of the guidance for determining the reevaluated seismic hazard, and that the licensee's reevaluated seismic hazard was acceptable to address other actions associated with NTTF Recommendation 2.1: “Seismic.” The staff also confirmed that the licensee's GMRS for the Robinson site exceeds the SSE over the frequency range of approximately 0.5 to 100 Hz and that a seismic risk assessment was merited.

The licensee developed a SPRA to perform the seismic risk assessment for Robinson in response to the 50.54(f) letter, specifically item (8) in Enclosure 1 of the 50.54(f) letter and submitted the same to the NRC on December 12, 2019 (ADAMS Accession No. ML19346E204, non-public).

In its December 12, 2019, submittal the licensee stated that significant efforts were expended to eliminate or minimize conservatisms in its SPRA 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. Based on the submittal, as well as information available to the staff as part of its audit, the licensee considers its SPRA to be a “high fidelity as-built, as-operated model that best represents the physical and operating characteristics of the plant.” The licensee’s SPRA has undergone a full-scope peer-review by an independent group of experts against NRC-endorsed consensus PRA Standard for SPRAs. The peer-review followed NRC-accepted guidance which includes an independent walkdown of certain areas of the plant. The SPRA is expected to be a highly detailed and technical reflection of the plant response, and consequently, to demonstrate the risk of seismic events at Robinson. The review has identified certain key assumptions that need further information from the licensee.

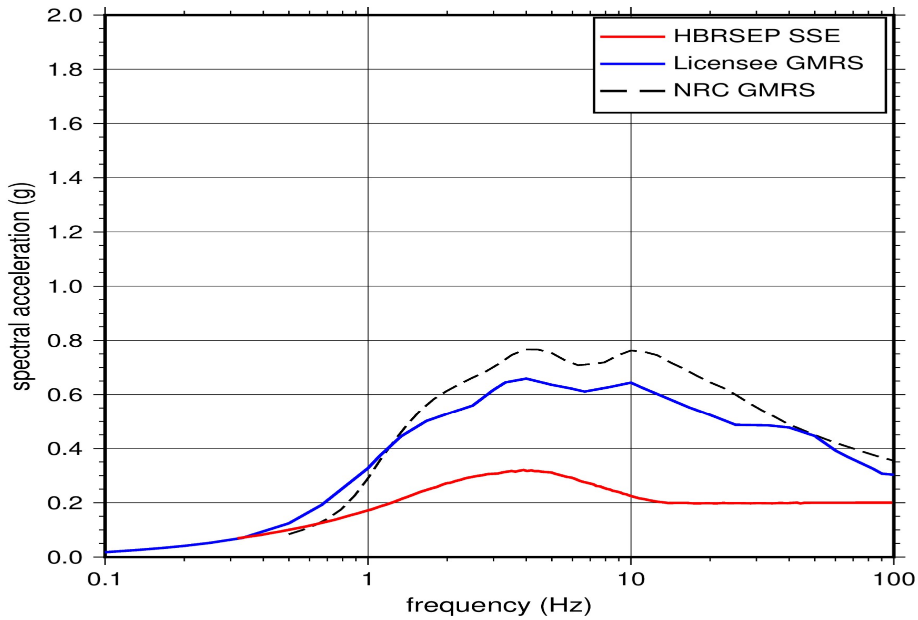

Figure 1. Comparison of the Ground Motion Response Spectrum (GMRS) from the re-evaluated hazard and the Safe Shutdown Earthquake (SSE) for Robinson

The SPRA for Robinson revealed the following dominant risk contributors based on the re-evaluated hazard:

- Failure of the Class III turbine building (TB) structure due to its pounding[1] against the Reactor Auxiliary Building (RAB) and subsequent collapse of the Class III TB,
- Failure of the TB Gantry Crane,
- Failure of the Robinson dam due to liquefaction,[2]
- Failure of the steam-driven auxiliary feedwater (SDAFW) system due to liquefaction,
- Failure of the diesel fuel oil storage tank piping due to liquefaction,
- Failure of piping from all four deep wells due to liquefaction.

Figure 2 and Figure 3 shown the layout of the TB at Robinson at the ground and mezzanine floor level, respectively. Figure 2 and Figure 3 also identify certain SSCs relied upon for mitigation to demonstrate location and proximity to the Class III TB. Figure 4 shows a cross section of the Robinson dam.

[1] Seismic pounding is defined as the collision of adjacent buildings during earthquakes.
[2] Liquefaction is the process by which saturated, unconsolidated soil or sand acts as a fluid because it loses strength during an earthquake.

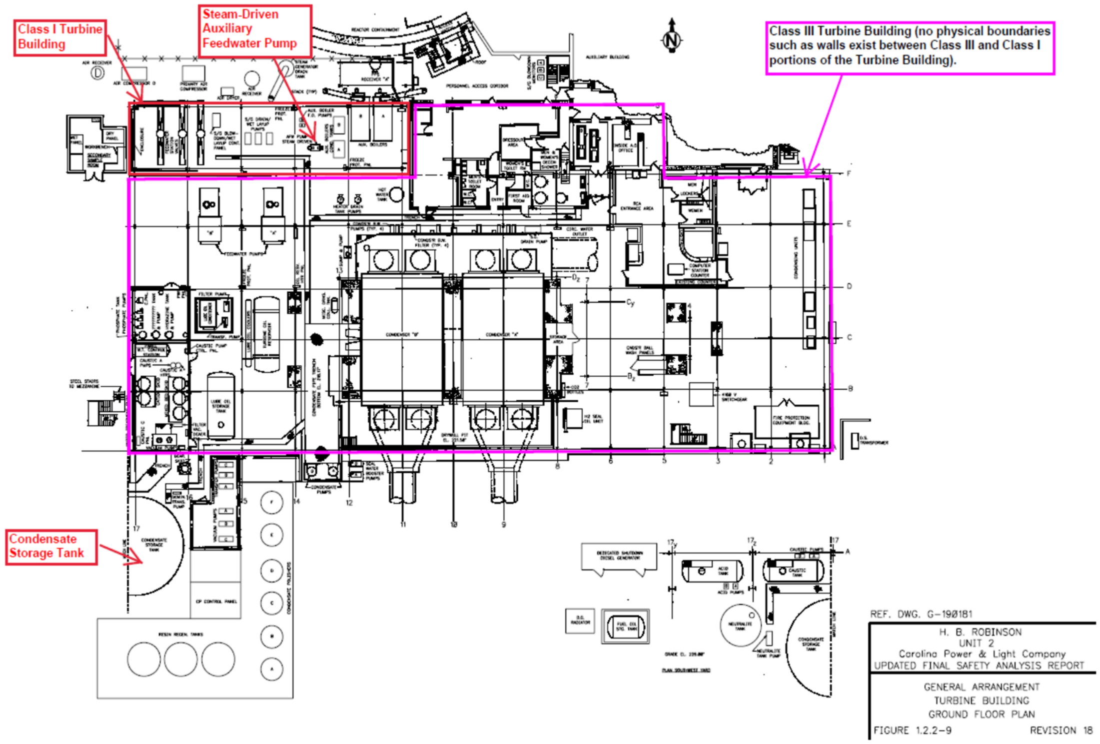

Figure 2 Layout of Turbine Building at the Ground Floor Level

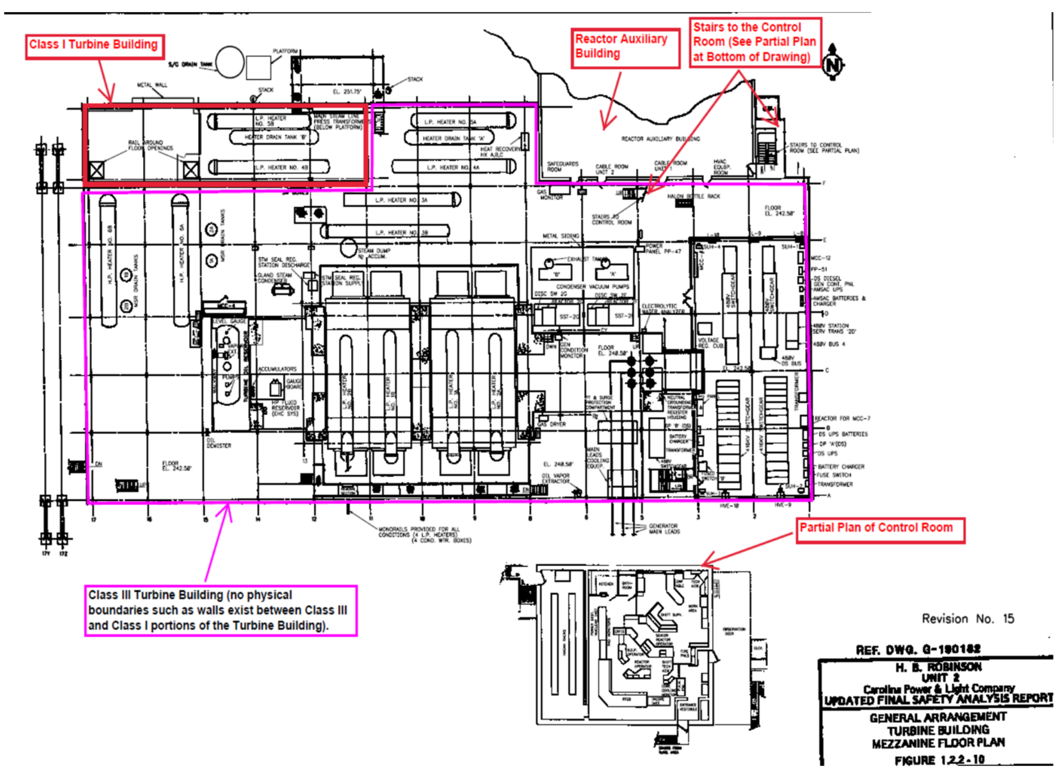

Figure 3 Layout of the Turbine Building at the Mezzanine Level

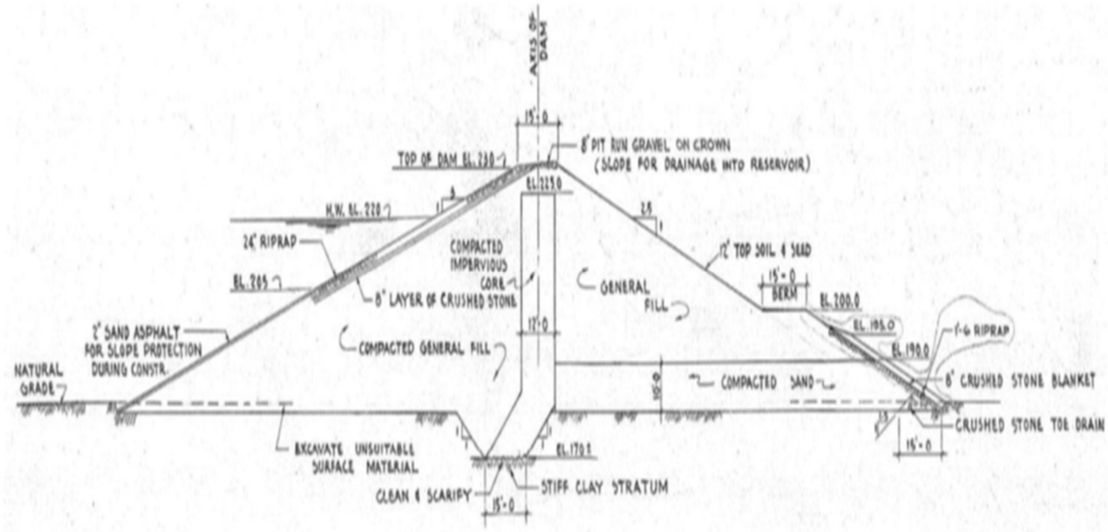

Figure 4 Cross-section of the Robinson Dam

2. Technical Evaluation Approach

The staff is following a structured and systematic approach for reviewing the licensee's SPRA. The approach is consistent with that used for review of multiple SPRAs submitted in response to the 50.54(f) letter.

Upon receipt of the licensee's December 12, 2019, SPRA submittal, a technical team of staff performed a completeness review to determine if the licensee's submittal includes all information requested by the 50.54(f) letter. The technical team performing the review consisted of staff experts in the fields of seismic hazards, fragilities evaluations, and plant response/risk analysis. On January 16, 2020, the technical team determined that the licensee has provided all information requested by the 50.54(f) letter.

As described in the 50.54(f) letter, the staff's detailed review focuses on (1) verifying the technical adequacy of the licensee's SPRA such that an appropriate level of confidence could be placed in the results and risk insights of the SPRA to support regulatory decisionmaking associated with the 50.54(f) letter, and (2) the risk and safety insights contained in the licensee's SPRA submittal to determine the need for further regulatory action. The NRC staff's detailed review also includes an audit of relevant licensee documents and responses to staff's questions through an electronic reading room.

The staff's review of the results and insights from the Robinson SPRA will evaluate the impact of a seismic events (known hazard) in terms of the risk, relevant failures of SSCs, the modes of failures, and the defense-in-depth philosophy. The staff's technical evaluation approach is consistent with the PRA Policy Statement which states,

> The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

> Where appropriate, PRA should be used to support the proposal for additional regulatory requirements in accordance with 10 CFR 50.109 (Backfit Rule).

The staff also concluded that its technical evaluation approach and recommendations are consistent with the Principles of Good Regulation (POGR), especially: (i) basing final decisions on objective, unbiased assessments of all information, (ii) documenting them with appropriate rationale, and (iii) ensuring regulatory activities are consistent with the degree of risk reduction they achieve.

2.1. Immediate Safety Concern

The staff followed the risk-informed approach for evaluating emergent issues described in LIC-504, "Integrated Risk-Informed Decision-Making Process for Emergent Issues," Revision 5 (ADAMS Accession No. ML19253D401). Section 4.2.1 of LIC-504 provides guidelines and states that any of those guidelines can be used to question whether immediate regulatory action (e.g., an immediate shutdown order) is required. The guidelines are: defense-in-depth is significantly degraded (e.g., multiple barriers are moderately to significantly degraded, functional redundancy or diversity is significantly compromised, or vulnerability to single failures is significantly increased); significant loss of safety margin (e.g., the calculated ASME code structural factors for a component are equal to or less than 1); and the risk impact from internal or external events is high, as determined using risk metrics. LIC-504, Revision 5, provides examples of various metrics and corresponding numerical values for use in decisionmaking.

Based on the available information available, the staff concludes that an immediate safety concern is not identified because:

- Functional redundancy and diversity are degraded at the seismic acceleration corresponding to the plant's SSE but not significantly compromised. Similarly, safety margins are degraded at the acceleration corresponding to the plant's SSE but are not significantly lost. The degradation in defense-in-depth and safety margins for seismic accelerations at and below the SSE does not demonstrate guaranteed or extremely high probability of loss of functional redundancy and diversity.

- The risk impact from seismic events is high but not in the range that supports immediate regulatory action (i.e., in the range of 1E-3 per year for CDF and 1E-4 per year for LERF per LIC-504, Revision 5). Due to the inclusion of safety and non-safety systems in the licensee's SPRA, the risk quantification includes consideration of all potential failures and impacts on mitigation. While the risk from seismic events exceeds the RG 1.174, Revision 3, guidelines, it does not rise to the level of risk from past events that have necessitated immediate regulatory action. The key assumptions in the licensee's SPRA will not challenge the conclusion on the risk impact.

- The Robinson dam failure contributes about 7 percent to the total SCDF and about the same contribution to SLERF. Therefore, even if the entire contribution to SCDF from dam failure (about 1E-5 per year) is added to LERF it would not reach the range that supports immediate regulatory action per LIC-504, Revision 5.

- The acceleration corresponding to the plant's SSE has a 1 in 3,790-year return period based on the re-evaluated hazard. While the return period is high compared to 1 in 100,000-year seismic event, it is not on the order of 1 in 100-year period.

The staff continued to use a risk-informed decisionmaking approach for its detailed evaluation of the risk insights and results demonstrated by the SPRA. The purpose of the evaluation was to identify and support its recommendations to the SMRP. Commission policy articulated in SECY-98-144 (ADAMS Accession No. ML003753601) defines a risk-informed approach to regulatory decisionmaking as one that represents a philosophy whereby risk insights are considered together with other factors to establish requirements that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to public health and safety. The staff used the principles of risk-informed decisionmaking discussed in Section C of RG 1.174, Revision 3 to arrive at its recommendation in an integrated manner. A discussion, and the NRC staff's conclusions on the five principles of risk-informed decisionmaking, is provided below.

2.2. Principle 1 (Compliance with Regulations)

To the best of the staff's knowledge, the licensee is currently in compliance with existing regulations as well as its licensing basis, specifically those related to seismic design.

2.3. Principle 2 (Consideration of Defense-in-Depth)

The Commission has long considered the defense-in-depth philosophy as an important element of plant design and operation. Commission policy articulated in SRM-SECY-98-144 states:

> Risk insights can make the elements of defense-in-depth more clear by quantifying them to the extent practicable...Decisions on the adequacy of or the necessity for elements of defense should reflect risk insights gained through identification of the individual performance of each defense system in relation to overall performance.

The risk insights from the Robinson SPRA demonstrate that, based on the re-evaluated hazard[3], the non-trivial failure likelihood of the seismic vulnerabilities at the plant[4] degrade the plant's defense-in-depth by causing the potential failure of redundant, diverse, and independent SSCs even for seismic accelerations at and below the SSE level. Based on the results of a sensitivity performed by the licensee and provided in response to NRC staff's audit question, the decrease in defense-in-depth at and below the SSE level is manifested in an approximately 4 percent contribution to SCDF from accelerations at and below the SSE level. The sensitivity also demonstrated that the 'plant level' high confidence of low probability of failure (HCLPF); the acceleration at which there is 95 percent confidence that the failure probability is 5 percent of less) was 0.17g which is lower than the SSE level of 0.2g. The results of the sensitivity showed that the cumulative conditional core damage probabilistic at the SSE of 0.2g is approximately 5 percent.

[3] The insights from Robinson's SPRA are based on the re-evaluated hazard and does not represent any staff analysis to change or alter the licensee submitted values. Therefore, all discussions of contributions and failure probabilities from the SPRA are based on the re-evaluated hazard. The staff's use of the licensee submitted values does not represent a finding that the numbers are adequate for use here or any other regulatory action. This qualification is not repeated hereafter for simplicity unless stated otherwise.
[4] Henceforth, the terms "seismic vulnerability(ies)" and "dominant risk contributor(s)" are used interchangeably in this document.

Table 1 provides the contribution to total SCDF from several SPRAs submitted in response to the 50.54(f) letter from accelerations at and below the SSE. The defense-in-depth at the acceleration of the SSE is demonstrated by the negligible, if any, contribution to SCDF from accelerations at and below a plant's SSE, even from the re-evaluated hazard. Robinson's SPRA demonstrates a non-trivial contribution to SCDF from accelerations at and below the SSE level. The contribution translates into a SCDF of approximately 5.6E-6 per year[5] which is approximately six times higher than that for Plant 3[6].

Table 1 Comparison of contribution of seismic acceleration at and below SSE to total SCDF for different operating plants

Plant	SSE at Peak Ground Acceleration (g)	Contribution to SCDF [Cumulative from accelerations at and below SSE]	
		%	per year
Plant 1	0.15	0.0	~0
Plant 2	0.12	0.0	~0
Plant 3	0.75	3.0	~9E-7
Plant 4	0.12	0.0	~0
Plant 5	0.125	0.0	~0
Plant 6	0.18	0.0	~0
Plant 7	0.10	0.0	~0
Plant 8	0.20	0.0	~0
Robinson	0.20	4.0	5E-6

The dominant risk contributors identified by the licensee's SPRA include:

- Failure of the Class III turbine building (TB) structure due to its pounding against the Reactor Auxiliary Building (RAB) and subsequent collapse of the Class III TB
- Failure of the TB Gantry Crane leading to failure of the condensate storage tank (CST)
- Failure of the Robinson dam due to liquefaction
- Failure of the SDAFW system due to liquefaction
- Failure of the diesel fuel oil storage tank piping due to liquefaction
- Failure of piping from all four deep wells due to liquefaction.

The staff determined that the dominant risk contributors represent seismic vulnerabilities as explained in the 50.54(f) letter. If the Class III TB fails (failure probability of 12% at a seismic

[5] The risk metrics from the Robinson SPRA are per reactor-year (i.e., period the reactor is operating which is indicated by its capacity factor). The simpler term 'per year' is used for the risk metrics from the Robinson SPRA in this evaluation with the same intent.
[6] The mean SCDF for Plant 3 is 2.8E-5 per year. The contribution of 3 percent from accelerations at and below Plant 3's SSE translates to 8.4E-7 per year.

event with return period of about 1 in 3,800 years), it results in the functional failure of several SSCs that are important to safety and therefore, represents a seismic vulnerability. The relative contribution of the dominant contributors to SCDF and SLERF is represented in Figure 5. Certain seismic vulnerabilities are dominant contributors to both SCDF and SLERF thereby impacting SSCs that prevent plant challenges from progressing to core damage as well as SSCs supporting the containment function such as containment fan coolers and sprays. The first five dominant risk contributors in Figure 6 contribute approximately 82 percent to SCDF and approximately 61 percent to SLERF. Additional details on dominant SLERF sequences is provided in 2.5.

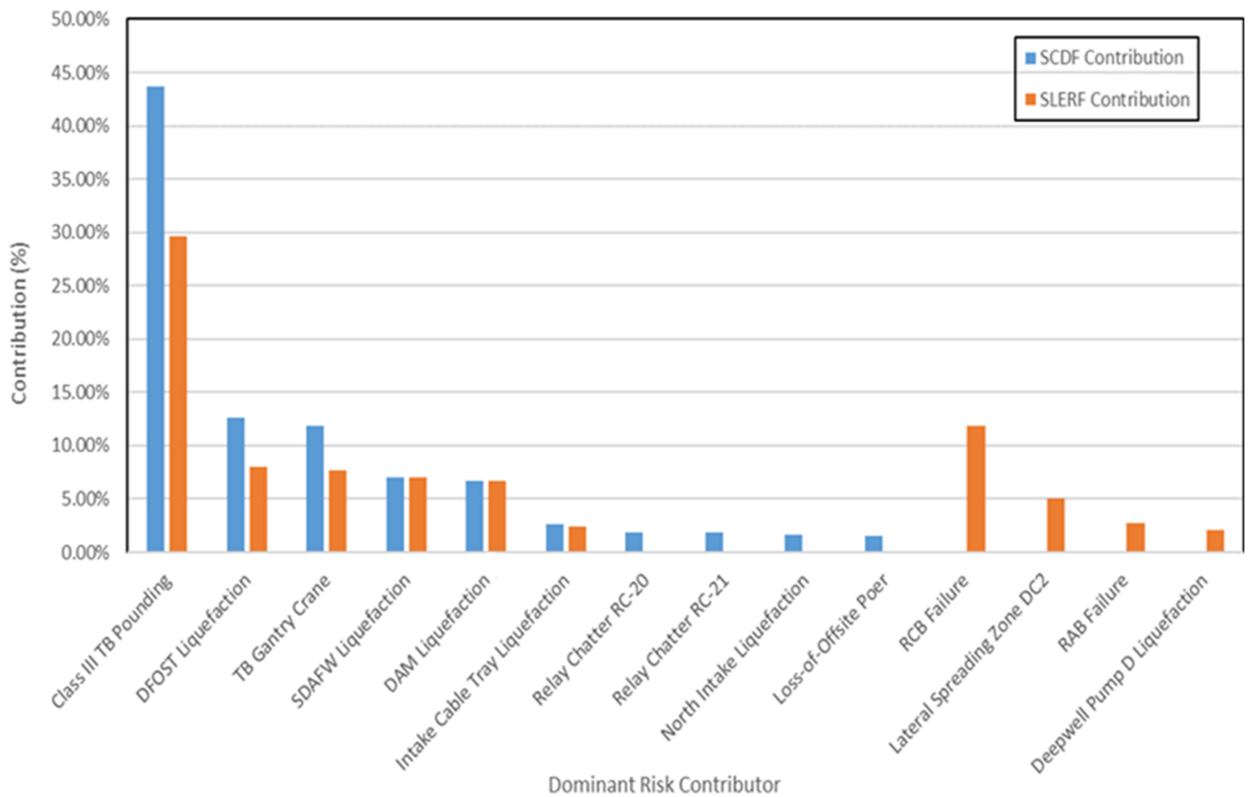

Figure 5 Relative contribution of dominant risk contributors (seismic vulnerabilities) to SCDF and SLERF

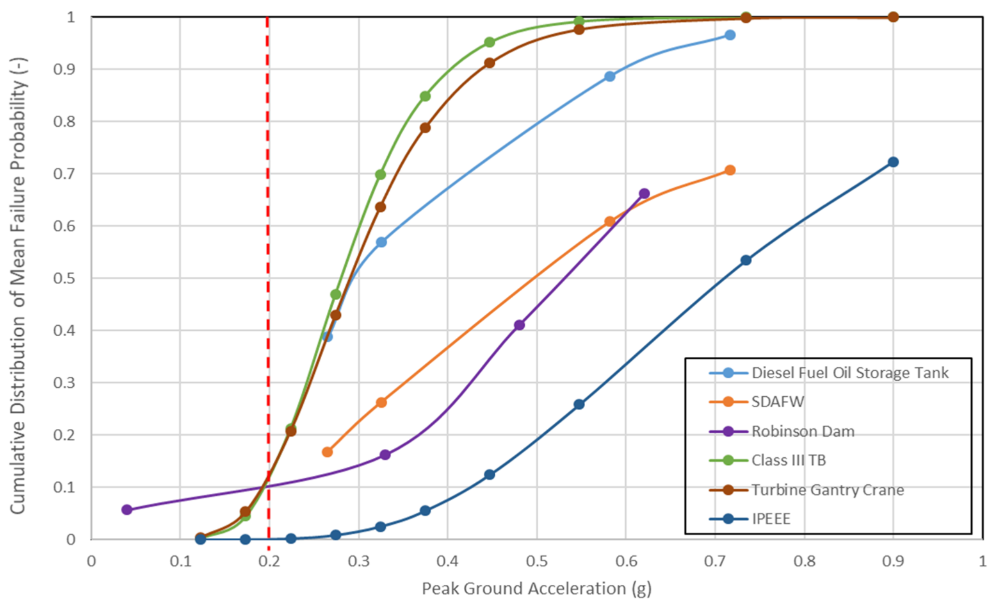

Figure 6 Cumulative distribution of the mean failure probabilities for certain dominant risk contributors as a function of seismic acceleration

The SPRA includes a probabilistic treatment of the dominant risk contributors (i.e., probability of failure as a function of seismic acceleration). The analytically derived failure probabilities, using modern day techniques and subjected to a peer-review, demonstrate non-negligible values for the dominant contributors even at the acceleration corresponding to the plant's SSE of 0.2g peak ground acceleration (PGA)[7] as shown in Figure 6 for certain dominant risk contributors. Table 2 provides a description of the dominant risk contributors whose failure probabilities are shown in Figure 6 as well as the corresponding failure mode. The failure probabilities of the seismic vulnerabilities, which are non-trivial at the acceleration corresponding to the plant's SSE, continue to increase with seismic acceleration. The Individual Plant Examination for External Events (IPEEE) curve in Figure 6 is based on the licensee's IPEEE submittal.

The impact of certain dominant risk contributors on various mitigating SSCs is provided in Table 2. The dominant risk contributors result in the direct or indirect impact on several SSCs representing redundant, independent, and diverse means of achieving safe shutdown.
Table 2 also lists the failure probabilities of dominant risk contributors at the plant's SSE acceleration level (at PGA) of 0.2g.
Table 2 also provides failure probabilities at the seismic acceleration of 0.3g because that acceleration is the PGA for the re-evaluated hazard as shown in Figure 1. It is evident from Table 2 that the dominant risk contributors listed above have non-trivial failure probabilities even at the acceleration corresponding to the plant's SSE. Non-trivial failure probabilities of SSEs at these accelerations are particularly significant due to relatively high frequency of earthquakes at those accelerations. The failure probability for the Robinson dam, which is the plant's Ultimate Heat Sink (UHS), is approximately 12 percent at the acceleration corresponding to the plant's SSE.

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[7] It is state-of-practice in seismic design and risk assessments to use the peak ground acceleration or PGA, which is the acceleration at a frequency of 100 Hz, as the common reference to perform analysis and compare results. The accelerations mentioned in this evaluation are PGA unless otherwise specified.

Table 2 Impact of seismic vulnerabilities on accident prevention and mitigation SSCs

Seismic Vulnerability	*Failure Mode*	*Failure Probability (%)*		*Direct Impact on Prevention and Mitigation*
		At 0.2g (return period of 1 in 3,800 years)	*At 0.3g (return period of 1 in 11,200 years)*	
Class III Turbine Building	Pounding	12	60	Loss-of-Offsite Power (LOOP) due to loss of corresponding switchgear
				Loss of transformers powering emergency buses from offsite power
				Loss of AFW-A and -B through LOOP
				Loss of AFW-C (with dedicated diesel generator)
				Loss of Steam Driven AFW (SDAFW)
				Loss of Condensate Storage Tank (CST) which is primary water source for all AFWs
				Loss of Condenser Hotwell (an alternate AFW or service water source)
				Loss of Dedicated Shutdown Diesel Generator (DSDG) function due to loss of corresponding switchgear
				Loss of Diverse and Flexible Coping Strategies (FLEX) connections and equipment (stored in turbine building)
Robinson Dam	Liquefaction	12	15	Loss of Ultimate Heat Sink (UHS)
				Loss of service water (SW)

Seismic Vulnerability	*Failure Mode*	*Failure Probability (%)*		*Direct Impact on Prevention and Mitigation*
		At 0.2g (return period of 1 in 3,800 years)	*At 0.3g (return period of 1 in 11,200 years)*	
				Loss of Emergency Diesel Generators (EDGs) through loss of SW
				Loss of SDAFW through loss of SW
				Loss of backup water source for all AFWs
Turbine Building Gantry Crane	Anchorage	12	54	Loss of CST which is primary water source for all AFWs
SDAFW	Liquefaction	6	22	Loss of SDAFW
Diesel Fuel Oil Storage Tank Piping	Liquefaction	19	49	Loss of EDGs through fuel oil depletion
				Loss of AFW-A and -B through loss of SW and loss of emergency power
				Loss of Deepwell D through loss of emergency power
Deepwell D Piping	Liquefaction	34	65	Loss of backup source for all AFWs
				Loss of backup source of SW for one EDG and SDAFW (i.e., loss of one EDG and SDAFW)

An explanation of the impacts of seismic vulnerabilities is as follows:

- The Robinson dam, an earthen dam which is the UHS for the plant and the primary source of service water (SW) suffers liquefaction-induced failure with an approximately 12% failure probability at the acceleration corresponding to the plant's SSE and approximately 15 percent at 0.3g. If the Robinson dam fails, it results in the draining of Lake Robinson and the loss of the UHS as well as primary source of SW. UHS is necessary for the operation of the plant and SW is necessary for the operation of several plant SSCs, including the emergency diesel generators (EDGs) and the SDAFW. In addition, SW also provides backup supply to the CST.

- The structural failure of the Class III TB structure has a failure probability of 12 percent at the acceleration corresponding to the plant's SSE and 60 percent at 0.3g. If the Class III TB fails, it results in (i) the loss-of-offsite power (LOOP) due to the loss of equipment housed in the Class III TB that provides offsite power to plant SSCs, which is compounded by a low likelihood of offsite power recovery due to the nature of the seismic event; (ii) the failure of Auxiliary Feedwater Pump-C (AFW-C) because the pump and its diesel driven power source are housed in the Class III TB; (iii) a high probability of failure of the SDAFW pump and/or the CST; (iv) failure of the SDAFW system occurs either directly due to interaction with the failed Class III TB or due to the loss of alternate suction source from the condenser waterbox; and (v) the failure of the DSDG function because the DSDG is connected to the emergency buses through switchgear in the Class III TB. The structural failure of the Class III TB introduces a new common-cause failure mechanism for the SSCs impacted by its failure.

- The TB Gantry Crane has a 12 percent probability of failure at the acceleration corresponding to the plant's SSE and 54 percent at 0.3g. Upon its failure, the CST fails leading to the failure of all AFW pumps including SDAFW system.

- The SDAFW pump has another failure mode via its settlement due to liquefaction (the failure mode is in the dominant risk contributors).

- The underground piping from the diesel fuel oil storage tank (DFOST) which provides fuel to the day tanks for operation of the EDGs suffers liquefaction-induced failure with a 6 percent at 0.2g and 22 percent at 0.3g. If the DFOST piping fails, it results in the loss of supply of fuel oil to the EDGs, thereby failing the function of the EDGs.

- Deepwell pump D can act as an alternative SW source for one of the EDGs and the lube oil heat exchanger for continued operation of the SDAFW system. If the Deepwell pump D piping fails due to liquefaction, which has a 6 percent failure probability at 0.2g and 65 percent at 0.3g, it eliminates the ability of Deepwell pump D to perform its functions.

- The loss of offsite power, the EDGs, Lake Robinson, CST and Deepwell D results in a combined extended loss of alternating current (ac) power (ELAP) and loss of UHS (LUHS).

- Diverse and Flexible Coping Strategies (FLEX) have been developed because of orders to cope with an ELAP concurrent with LUHS. The Robinson SPRA includes credit for the plant's FLEX strategies. The SPRA demonstrates that FLEX equipment provides marginal, if any, benefit to the plant's defense-in-depth. This is due to the failures discussed above which challenge the FLEX equipment's ability to provide mitigation (e.g., the failure of the Robinson dam which provides water supply for the FLEX strategies). Sensitivity performed by the licensee to determine the impact of using FLEX equipment for alternate feedwater injection results in only a 2 percent decrease in SCDF. Therefore, based on the re-evaluated hazard, there is high probability that FLEX equipment would not be able to provide mitigation for the dominant seismic accident sequences.

The acceleration of 0.3g is the PGA for the GMRS based on the re-evaluated hazard. The reevaluated hazard and the resulting GMRS represents the seismic hazard at the plant based on the best available knowledge and technology. The failure probability and resulting impacts from the above-mentioned dominant risk contributors increases, in several cases above 50 percent, at the acceleration of 0.3g. Those failure probabilities represent a further degradation of the defense-in-depth.

The information provided in Table 2 shows that the seismic vulnerabilities impact the key safety functions (KSFs) of decay heat removal and inventory control resulting in non-negligible failure probabilities for those KSFs at the acceleration corresponding to the plant's SSE. Figure 7, reproduced from NEI 12-06, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," Revision 4 (ADAMS Accession No. ML16354B421), shows the different defense-in-depth strategies available at U.S. nuclear power plants.

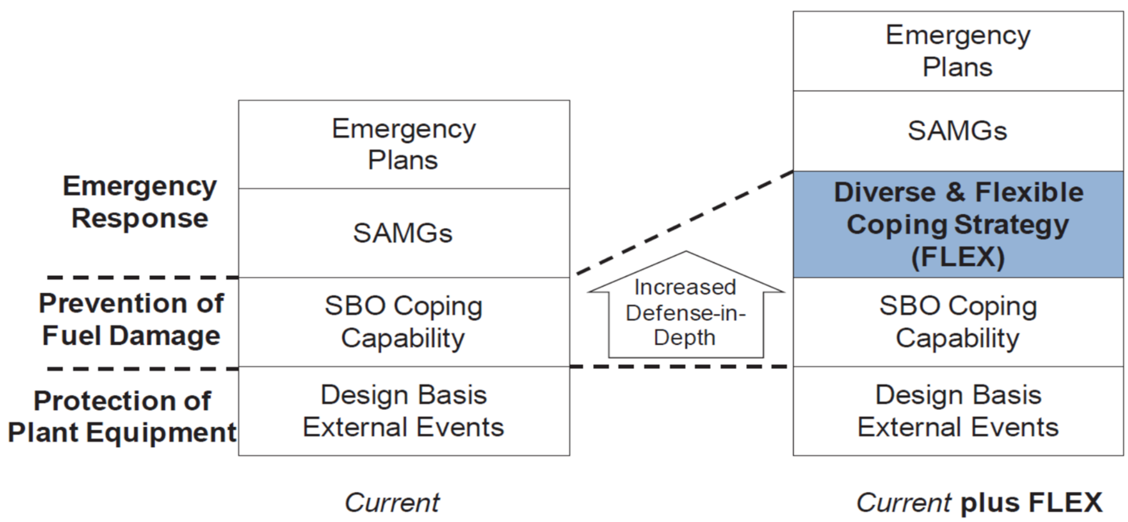

Figure 7 Defense-in-Depth Strategies Available at U.S. Nuclear Power Plants (reproduced from NEI 12-06, Revision 4)

As discussed above, protection and the ability of plant equipment (i.e., based on abnormal and emergency operating procedures) as well as FLEX strategies to mitigate seismic events are challenged with non-trivial failure probabilities at low seismic accelerations. Based on the available information, the plant's station blackout (SBO) coping depends on the dedicated DSDG as well as the SDAFW system. As noted in Table 2, the dominant failure modes impact both the DSDG switchgear (and therefore, its ability to power any equipment) as well as the SDAFW. The dominant risk contributors also impact availability of SW, which would in turn affect the operation of the loads from the DSDG. In addition, the plant's batteries have a 1-hour battery life which, upon deep direct current (dc) load shedding, would be extended to approximately 3.75 hours. Multiple dominant risk contributors, including the settlement of the SDAFW pump itself, upon failure, result in the loss of function of the SDAFW system. Therefore, the plant's SBO coping is also challenged based on the non-trivial failure probabilities of the dominant risk contributors (even at seismic accelerations of 0.2g and 0.3g) based on the insights from the SPRA. The severe accident mitigation guidelines (SAMGs) for the plant also rely on equipment whose function will be lost upon failure of the dominant risk contributors. As an example, the SAMG for SG injection identifies the SDAFW, the CST, and the Robinson Dam. Therefore, SAMG strategies are also degraded with non-trivial probability. Further, as noted in Section 2.7, if the failure of the dominant risk contributors were to occur, it would also result in a degradation of the emergency plans. Therefore, the results and insights from the SPRA identify non-trivial degradation of all defense-in-depth strategies even at seismic accelerations of 0.2g and 0.3g (which have return periods of approximately 1 in 3,790 years and 1 in 11,200 years, respectively).

Section C.2.1.1.3 of Regulatory Guide (RG) 1.174, Revision 3 (ADAMS Accession No. ML17317A256) provides guidance for consideration of defense-in-depth in risk-informed decisions. The seismic vulnerabilities impact several defense-in-depth considerations in RG 1.174, Revision 3. The non-negligible failure probabilities of the seismic vulnerabilities even at the acceleration corresponding to the plant's SSE result in reduction in the system redundancy, independence, and diversity due to the failure of multiple independent and redundant SSCs. Further, if the Class III TB collapses and/or the Robinson Dam fails due to a seismic event it introduces a new common-cause failure that impacts redundant and diverse SSCs as shown in Table 2. This results in a challenge to the guidance in RG 1.174, Revision 3 which indicates that the evaluation "should demonstrate that the change does not result in a significant reduction of existing CCF defenses or introduce new CCF dependencies."

Figure 8 shows the contribution of each of the dominant risk contributors to the total SCDF at different PGA values, which correspond to the seismic 'bins'[8] in the SPRA. Figure 8 shows that the seismic vulnerabilities have non-negligible contributions (i.e., impact the plant response) across the seismic accelerations considered in the SPRA, including low seismic accelerations such as 0.17g through 0.4g. These seismic accelerations have a return period of 1 in 2000 years (for 0.17g) through 1 in 19,000 years (for 0.4g). As shown in Figure 13, these results are markedly different from those reviewed by the staff for several SPRAs submitted in response to the 50.54(f) letter, where the contribution of the dominant risk contributors is from higher seismic accelerations (e.g., 0.5g and above) which have return periods of 1 in 100,000 through 1 in 1,000,000 years.

[8] It is state-of-practice in seismic PRAs to discretize the seismic hazard curve to facilitate processing and quantification. The number of 'bins' are chosen based on state-of-practice as well as sensitivities to determine the impact of more refined discretization. The discretization used in seismic PRAs is peer-reviewed for appropriateness.

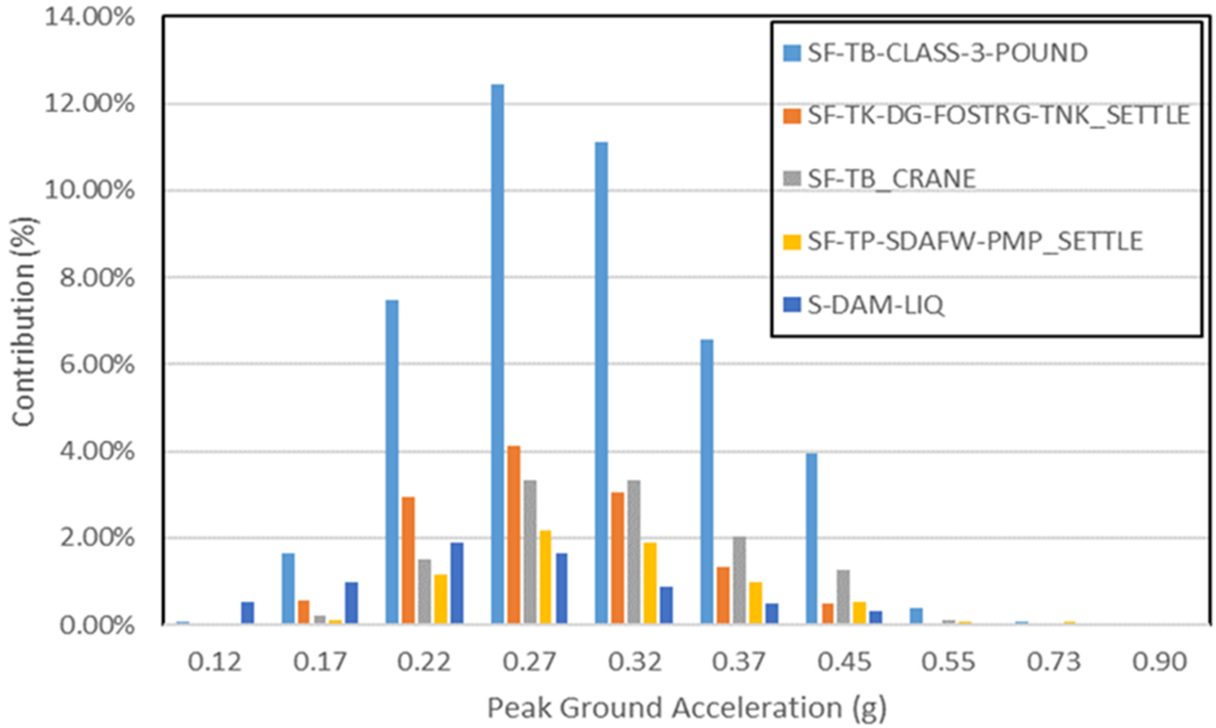

Figure 8 Contribution of seismic vulnerabilities to the risk from each 'bin' in the SPRA

The dominant risk contributors from the licensee's SPRA are different from several other SPRAs submitted in response to the 50.54(f) letter and reviewed by the staff in their failure modes, impacts, and failure probabilities at low seismic accelerations. Pounding and liquefaction-induced failures have not been identified as dominant risk contributors in any of the nearly 18 SPRAs reviewed by the staff for decisionmaking for the 50.54(f) letter. These failure modes are such that if they occur, the function of SSCs impacted by them cannot be recovered. This is especially true for structural and liquefaction failures which have an impact only at high seismic accelerations (e.g., 0.75g and higher) in other SPRAs. Further, as highlighted in Table 2**Error! Reference source not found.**, the dominant risk contributors from the Robinson SPRA, upon their failure, impact several redundant, diverse, and independent SSCs in contrast to dominant risk contributors in other SPRAs that impact individual SSCs.

It is important to understand that multiple seismic vulnerabilities impact the function of SSCs necessary for accident prevention and mitigation and therefore, addressing a single seismic vulnerability, by itself, appears to be ineffective.

In summary, based on the re-evaluated hazard, the dominant risk contributors have non-negligible failure probabilities even at low seismic accelerations from 0.17g to 0.4g. If failure of dominant risk contributors were to occur, it would result in the failure of front-line and back-up systems including UHS, SW, offsite power, all alternate feedwater systems, the CST, the EDGs, and the FLEX strategies, thereby resulting in a non-negligible failure probability of redundant, diverse, and independent SSCs and key safety functions. Different seismic vulnerabilities impact the same SSCs, such as SDAFW, CST, and EDGs, thereby resulting in the increased likelihood of losing the function of those SSCs. The failures described above result in an ELAP concurrent with LUHS. The impact of failures challenges different defense-in-depth strategies including emergency procedures, SBO coping, FLEX, SAMGs, and EDMGs. The impact of the non-negligible failure probability of the dominant risk contributors and their impact on defense-in-depth at the acceleration corresponding to the plant's SSE is manifested in a contribution of approximately 4 percent to total SCDF from accelerations at and below the acceleration corresponding to the plant's SSE, which has a return period of approximately 1 in 3,790 years based on the re-evaluated hazard. The contribution to total SCDF increases to approximately 45-50 percent from accelerations at and below 0.3g, which has a return period of approximately 1 in 11,200 years.

Based on the above discussion and the guidance in RG 1.174, Revision 3, the staff concludes that the defense-in-depth of the plant, the second principle of risk-informed decisionmaking, is degraded.

2.4. Principle 3 (Consideration of Safety Margins)

NUREG-2122, "Glossary of Risk-Related Terms in Support of Risk-Informed Decisionmaking," (ADAMS Accession No. ML13311A353) defines safety margin as the extra capacity factored into the design of a structure, system, or component so that it can cope with conditions beyond the expected to compensate for uncertainty. Safety margins can be considered a part of, or complementary to, defense-in-depth in that they provide extra (redundant) capacity. Incorporation of safety margins is one of the ways designers deal with the uncertainty of the challenges that the designed SSCs face. The design of plant SSCs includes conservatisms in various codes and practices which results in safety margins to achieving regulatory requirements.

In SPRAs, the safety margin in the design, construction, and response of SSCs to seismic events is explicitly captured in the analytical evaluation of the fragility of an SSC. The fragility represents the conditional probability of failure of an SSC at a particular seismic acceleration. The analytical evaluation to determine the fragility of an SSC includes consideration of the SSC's capacity arising from design, construction, testing, installation, and location. Therefore, the fragility and the resulting failure probability used in the SPRA for an SSC represents physical failure of the SSC to perform its function and not just the exceedance of a regulatory or design code limit.

Section 2.5 of the Updated Final Safety Assessment Report (UFSAR) for Robinson includes a discussion of the seismic evaluations performed as part of the siting and determination of the SSE. The seismic evaluations included taking borings and laboratory examinations of the soil at the site. Section 2.5.2.6 provides the SSE for the site and states "[i]t is important to note that even if an earthquake comparable to the Charleston shock were to occur 35 miles from the site, the ground acceleration would not exceed 0.2g." Section 2.5.5 includes a discussion of the analyses performed for the Robinson dam and concludes that "[t]hese analyses indicate a very large margin against gross failure of the dam and appurtenances…The evaluation of the dam indicates that it will not fail when subjected to the maximum credible earthquake or flood."

The licensee's IPEEE submittal for seismic hazards used the seismic margins analysis (SMA) approach for a review level earthquake of 0.3g PGA. The Robinson IPEEE included an evaluation of the potential for interaction between the Class III and Class I portions of the TB and concluded that "[t]he total maximum displacement for both portions of the Turbine Building do not exceed the spacing between both portions of the building." The IPEEE submittal also included an evaluation of soil liquefaction and lateral spreading and concluded "liquefaction is judged not to be a likely concern for the Robinson Dam at 0.3g and the dam is considered acceptable for a 0.3g earthquake." The NRC staff safety evaluation for the licensee's IPEEE submittal stated that "the seismic IPEEE needed a more detailed review because of specific concerns related to the seismic analyses (e.g., potential soil failures, selection of earthquake magnitude for the review level earthquake, seismic stability of Lake Robinson Dam, seismic capacity calculations, containment walkdown)." The NRC staff reviewed additional information on soil liquefaction provided by the licensee as part of its IPEEE submittal and concurred with the licensee's assessment that liquefaction, and other geotechnical hazards, would not "adversely affect the performance of buried pipelines and structures as a result of the 0.3g peak horizontal ground acceleration review level earthquake at [the] site."

The insights from the licensee's SPRA submitted in response to the 50.54(f) letter challenges the evaluations in the licensee's UFSAR and IPEEE submittal. The interaction between the Class III and Class I portions of the TB and the gross failure of the Robinson dam as well as buried piping due to liquefaction has a non-trivial failure probability, even at the acceleration corresponding to the plant's SSE, then previously known or expected. As discussed in 2.3, specifically Table 2 and Figure 6, these failure modes have high failure probabilities at low seismic accelerations ranging 0.2g through 0.4g which are in spite of the inherent design margins. If the structural failure and soil liquefaction failure occur, they result in exceedance of the design margins of several SSCs because the load due to structural collapse is not included during the design of the SSCs. As an example, the load due to the Class III TB structural failure is not expected to have been a consideration in the design of the AFW-C, the Class I TB, and the SDAFW, all of which fail due to the collapse of the Class III TB. The structural failure of the Class III TB collapses due to a seismic event introduces a new common-cause failure mechanism that degrades the safety margins of the impacted SSCs. Therefore, the impact of the dominant risk contributors on the safety margin of SSCs is not limited to a single SSC but can be considered to be plant-wide per the discussion in LIC-504, Revision 5.

Based on the above discussion, the staff concludes that the safety margins in the SSCs at the plant that are necessary for mitigation of seismic events are degraded due to the re-evaluated hazard, even at low seismic accelerations from 0.2g to 0.4g. The insights from the licensee's SPRA challenges the evaluations in the licensee's UFSAR and IPEEE submittal.

2.5. Principle 4 (Demonstration of Acceptable Level of Risk)

The mean SCDF and SLERF based on the quantification of the Robinson SPRA are 1.3E-4 per year and 2.5E-5 per year, respectively. The mean values are based on the distribution of SCDF and SLERF derived from propagation of parametric uncertainties in the SPRA and therefore, includes consideration of such uncertainties. In addition, the mean values also account for the plant capacity factor (i.e., fraction of the year that the plant is operating at full-power). The licensee's SCDF is the highest among the plants that have submitted SPRAs in response to the 10 CFR 50.54(f) request for information issued following the Fukushima Dai-ichi nuclear power plant accident. As shown in Table 3 and Figure 9, the SCDF and SLERF are the highest contributors to the total quantified plant risk at Robinson. The SCDF contributes approximately 70 percent of the total plant CDF and the SLERF contributes approximately 80 percent to the total plant LERF; both these large contributions to plant risk are atypical of the staff's experience with plant risk for operating power plants.

Table 3 Quantified Risk from PRAs for Different Hazards for Robinson

Hazard	CDF (per year)	LERF (per year)
Seismic	1.30E-04	2.50E-05
Internal Fire	4.60E-05	5.40E-06
Internal Events	3.20E-06	5.80E-07
Internal Flooding	1.70E-06	3.20E-07
Total	1.81E-04	3.13E-05

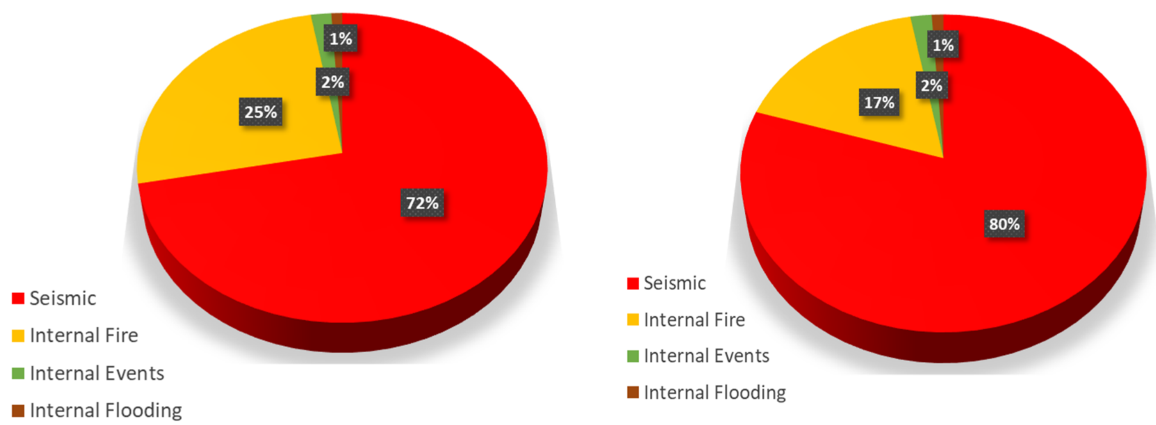

Figure 9 Relative contribution of risk from different hazards to the total plant risk

The CDF and LERF are recognized and utilized as surrogates to the Quantitative Health Objectives (QHOs) in the Commission's Safety Goal Policy Statement. A value of 1E-4 per year and 1E-5 per year has been assigned as the CDF and LERF guidelines, respectively, that meets the QHOs with acceptable margin. Several agency processes such as risk-informed licensing basis changes via RG 1.174, Revision 3, and the Significance Determination Process (SDP) in the Reactor Oversight Process (ROP) use these guidelines. The licensee's SPRA by itself demonstrates a non-trivial exceedance of the 1E-4 per year CDF as well as the 1E-5 per year LERF guidelines. The staff has previously accepted exceedances of 1E-4 per year for CDF for licensing applications where the exceedances were based on the cumulative sum of CDF from all hazards modeled using a PRA. The staff has usually cited known conservatisms in the analysis performed for different licensing applications (e.g., conservatisms in fire PRAs for amendment requests to adopt 10 CFR 50.48(c) for use of National Fire Protection Association [NFPA] Standard 805). According to the licensee, Robinson's SPRA is the result of significant efforts to eliminate or minimize conservatisms involving numerous refinement iterations and the licensee considers its SPRA to be "high fidelity as-built, as-operated model that best represents the physical and operating characteristics of the plant." In addition, the SPRA is based on the licensee's internal events PRA model and includes credit for the low-leakage reactor coolant pump (RCP) seals. Further, the quantification of the Robinson SPRA, by itself, results in an exceedance of the 1E-4 per year and 1E-5 per year guidelines for CDF and LERF as opposed to a cumulative contribution from multiple hazards. Since these guidelines are related to the Commission's Safety Goals, the exceedance of the guidelines also represents a decrease in the margins to the corresponding Safety Goals.

Table 4 provides the contribution of each seismic 'bin' to the total SCDF, on an individual bin and cumulative basis, which is represented graphically in Figure 11 and Figure 12. Table 4 as well as Figure 11 and Figure 12 demonstrate that seismic accelerations at and below 0.45g (which has a return period of approximately 1 in 28,000 years) contribute 83 percent of the SCDF and 61 percent of the SLERF. Further, the GMRS based on the re-evaluated hazard has a PGA of 0.3g and the contribution from accelerations at and below that level is approximately 45-50 percent. These insights are atypical compared to SPRAs submitted in response to the 50.54(f) letter.

Table 4 Contribution to total SCDF and SLERF from different 'bins' (with corresponding seismic accelerations

Bin	Representative Acceleration (g)	Occurrence Frequency (per year)	Cumulative Contribution to Total SCDF (%)	Cumulative Contribution to Total SLERF (%)
%G01	0.12	5.67E-04	1%	0%
%G02	0.17	2.49E-04	3%	1%
%G03	0.22	1.16E-04	13%	6%
%G04	0.27	5.86E-05	31%	16%
%G05	0.32	3.26E-05	50%	29%
%G06	0.37	1.94E-05	65%	41%
%G07	0.45	1.98E-05	83%	59%
%G08	0.55	8.47E-06	91%	72%
%G09	0.73	7.16E-06	98%	93%
%G10	0.90	1.88E-06	100%	100%

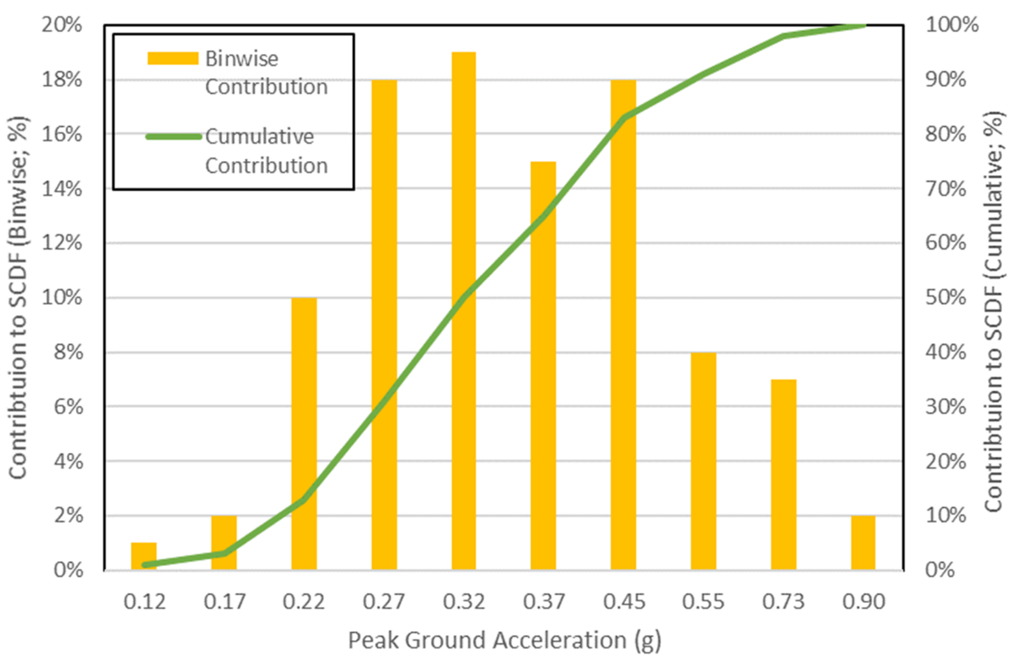

Figure 10 Contribution of each 'bin' in the SPRA to the total SCDF ('bin' accelerations are used to represent each 'bin')

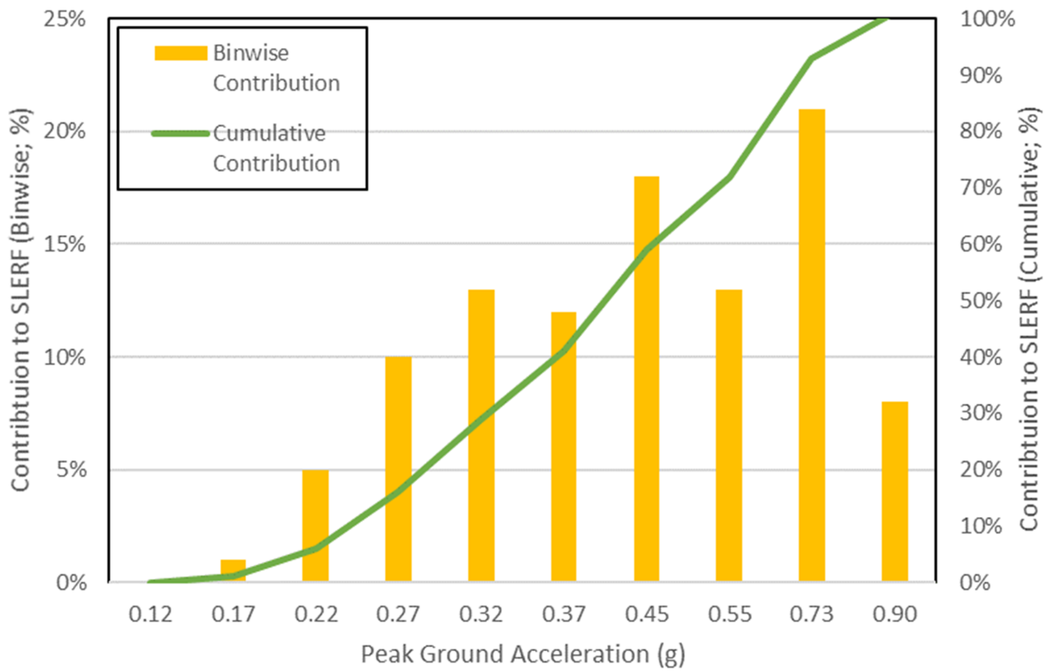

Figure 11 Contribution of each 'bin' in the SPRA to the total SLERF ('bin' accelerations are used to represent each 'bin')

The seismic accelerations at and below 0.45g that contribute 83 percent to the SCDF have occurrence frequencies ranging from approximately 9E-4 per year to approximately 3.5E-5 per year range ('bins' %G01 through %G07) as shown in Figure 12. These occurrence frequencies correspond to return periods of 1 in 1,000 years to 1 in 28,000 years. The occurrence frequency for the acceleration corresponding to the plant's SSE level (and above) is approximately 2.64E-4 per year, which corresponds to a return period of approximately 1 in 3,790 years[9]. The review level earthquake selected for the licensee's seismic margins analysis for the IPEEE was 0.3g. Accelerations at and below 0.3g contribute approximately 49 percent to the total SCDF and 29 percent to the total SLERF. Further, the occurrence frequency for such an earthquake is approximately 8.9E-5 per year, which corresponds to a return frequency of 1 in 11,200 years.

Figure 13 and Figure 14 compares the cumulative contribution to SCDF and SLERF, respectively, from Robinson's SPRA against seven different plants that have submitted SPRAs in response to the 10 CFR 50.54(f) request for information resulting from the accident at the Fukushima Dai-ichi nuclear power plant. As shown in Figure 13 and Figure 14, the contribution to Robinson's SCDF is from relatively low acceleration earthquakes which is atypical. As an example, Figure 15 shows that accelerations at or below 0.32g contribute 50 percent to the SCDF for Robinson as compared to less than 10 percent for 7 other plants. In addition, if comparison is made between the contribution to SCDF from accelerations at and below the occurrence frequency of 1E-4 per year, that contribution for a west coast plant (acceleration of

[9] The hazard curve provides the occurrence frequency for a selected seismic acceleration and the accelerations greater than the selected value (i.e., for accelerations equal to or greater than the selected value). This detail is implied and not repeated here onwards whenever the occurrence frequency of a selected acceleration is mentioned.

approximately 0.8g) is approximately 3.5 percent, for an east coast plant that experienced an earthquake (acceleration of approximately 0.38g) is about 10 percent, and for Robinson (acceleration of about 0.3g) is 30-35 percent. Therefore, the seismic vulnerabilities and their impact demonstrated by the Robinson SPRA is atypical compared to other plants.

Based on the information available to the staff, the combination of the core damage bin (CDB) and the containment safeguards event tree (CSET) which results in largest contribution to quantified SLERF is the damage state termed PDS3P. Damage state PDS3P contributes approximately 61 percent to the quantified SLERF. The CDB '3' includes sequences with (i) no SG cooling, (ii) cycling relief rate, (iii) early reactor pressure vessel (RPV) failure, (iv) high reactor coolant system (RCS) pressure at time of failure, (v) shallow cavity at RPV failure, and (iv) dry cavity after RPV failure. The CSET 'P' includes successful isolation and failure of containment sprays in injection as well as failure of containment fans. The containment phenomena resulting from PDS3P leading to SLERF is containment failure due to pressure loading following high pressure melt ejection (HPME) of the core debris. The pressure loading can be due to direct containment heating (DCH) or hydrogen combustion following HPME. The licensee performed plant-specific thermal-hydraulics analysis to determine the split fraction for containment failure following HPME. The analysis was subjected to a focused-scope peer-review which resulted in no finding level F&Os.

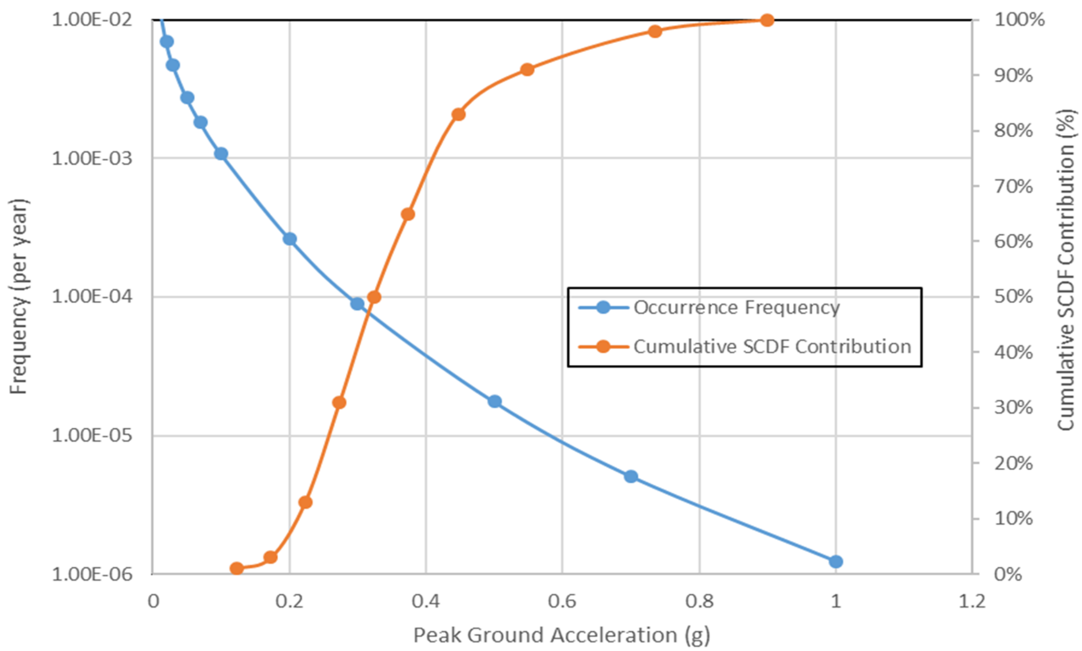

Figure 12 Cumulative contribution to SCDF and occurrence frequency for each 'bin' in the SPRA

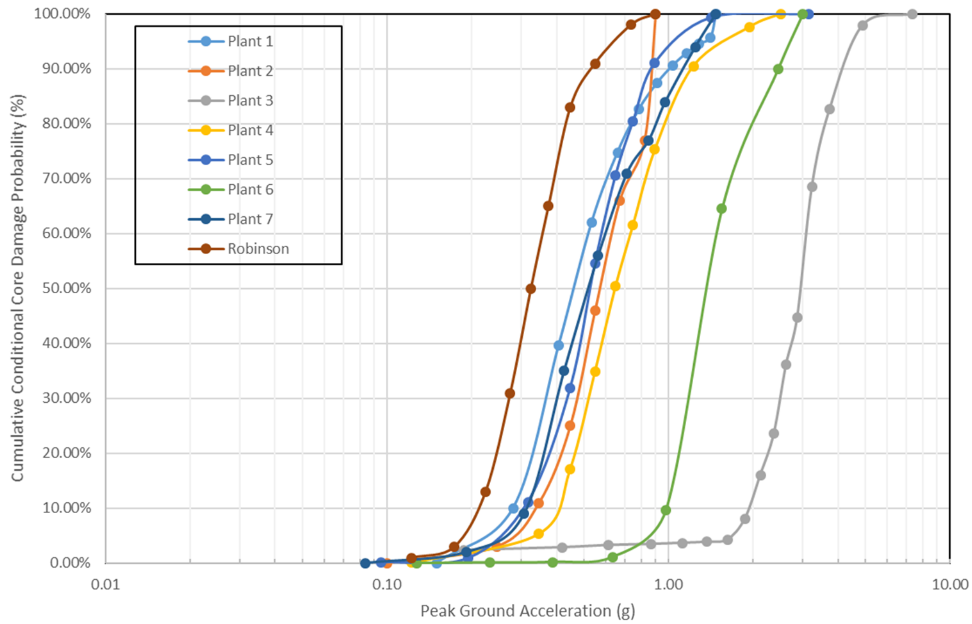

Figure 13 Comparison of conditional core damage probability as a function of seismic acceleration for Robinson against several other operating plants

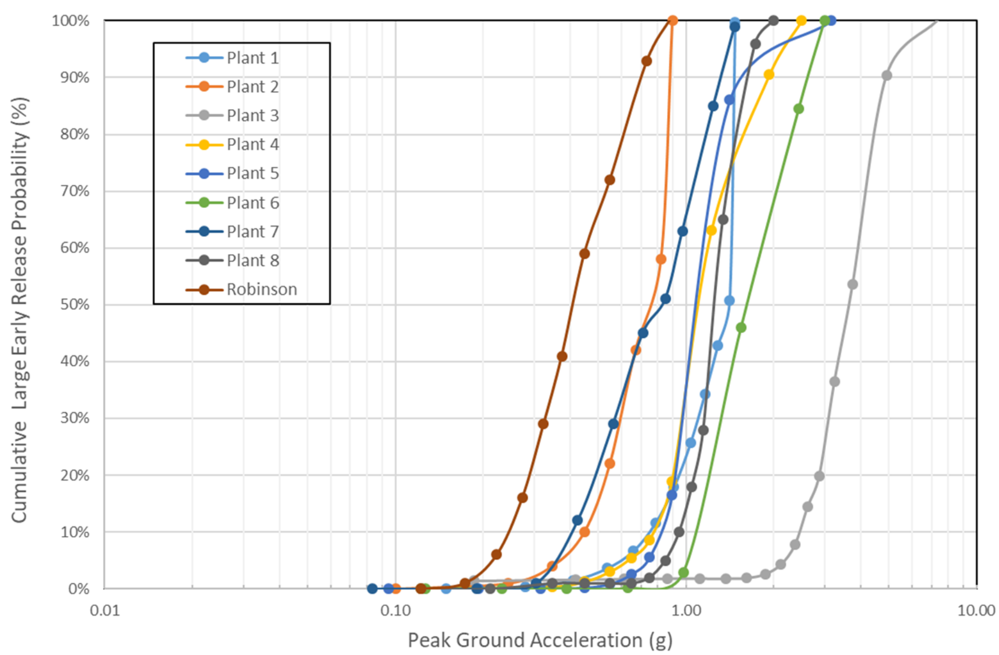

Figure 14 Comparison of conditional large early release probability as a function of seismic acceleration for Robinson against several other operating plants

The licensee performed a seismic margin analysis (SMA) as part of its IPEEE (i.e., response to GL 88-20, Supplement 4). The SMA does not provide a risk quantification or insights into dominant risk contributors. It results in two so-called success paths for safe shutdown following a seismic event; one for reactor coolant pressure boundary intact after a seismic event and the other for seismically-induced small loss-of-coolant accidents (LOCAs). The SPRA demonstrates the dominant risk contributors cause non-trivial failure probabilities of both success paths at even low seismic accelerations. Generic Issue (GI)-199, “Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants,” determined that the seismic risk at Robinson, using the hazard information available at the time and less sophisticated tools than the licensee’s SPRA, was approximately 1.5E-5 per year. The licensee’s SPRA demonstrates that the risk, using present day information and more sophisticated tools, is an order of magnitude higher and more consequential then previously known.

2.5.1. Uncertainty Analysis

Uncertainty analysis is an important part of development and quantification of SPRAs. Three types of uncertainty are evaluated in SPRAs: parametric, model, and completeness. The licensee evaluated the parametric uncertainty using state-of-practice techniques such as propagation of uncertainties using Monte-Carlo sampling. The mean SCDF and SLERF are derived from the results of the parametric uncertainty evaluation.

Model uncertainties are assumptions made for modeling convenience or where a choice is made between different alternatives about the modeling approach. The licensee performed several sensitivity studies to evaluate model uncertainties. Modeling assumptions that can impact a decision are termed key assumptions. Potential key assumptions in the licensee’s SPRA include the fraction used for interaction of the failed Class III TB with the Class I TB (0.5), the fraction used for the failure of the CST upon failure of the TB Gantry Crane (0.75), and the assumption that the failure of the CST due to interaction with the failed Class III TB is captured by the failure of the SDAFW. The fraction used for interaction of the failed Class III TB with the Class I TB (0.5) was also identified by the peer-review team as requiring an evaluation to determine its impact. The impacts of these potential key assumptions on the results, especially the dominant risk contributors and risk profile, will be evaluated by the staff based on information provided by the licensee. As an example, a sensitivity study performed by the licensee which assumed that the failed Class III TB always failed the Class I TB resulted in an increase in SCDF and SLERF by 14 percent and 9 percent, respectively.

In summary, the staff, based on the quantification from the SPRA that the licensee considers to be refined and “high fidelity,” concludes that (1) the risk from seismic events at Robinson is high, by itself results in an exceedance of the 1E-4 per year guideline, and is the largest contributor to the plant risk (the risk from seismic events is approximately 3 and 4 times the internal fire risk for CDF and LERF, respectively), (2) 83 percent of the SCDF and 61 percent of the SLERF is from low seismic accelerations of 0.12g to 0.45g (‘bins’ %G01 through %G07), which have occurrence frequencies in the E-4 and E-5 per year range, and (3) the contribution from seismic accelerations at and below the plant’s SSE (‘bins’ %G01 through %03) is approximately 4 percent to SCDF (i.e., a SCDF of approximately 15.6E-6 per year). Therefore, the staff concludes that increased attention from NRC management is justified based on the fourth principle of integrated decisionmaking.

2.6. Principle 5 (Performance Monitoring)

In the context of RG 1.174, Revision 3, the primary goal of this principle is to ensure that no unexpected adverse safety degradation occurs because of the changes to the licensing basis.

Since the licensee's SPRA represents the base risk of the plant from seismic events, performance monitoring related to a specific change does not apply.

The guidance in LIC-504, Revision 5, states that for a decision on what action NRC should take in response to an emergent issue, the staff should consider whether performance monitoring strategies should be implemented. Such performance monitoring includes interim actions to manage the risk demonstrated by the SPRA.

2.7. Qualitative Factors

SECY-14-0087 and the corresponding SRM supports the consideration of qualitative factors in decisions for regulatory action such as backfitting. The staff has identified qualitative factors based on the review of Robinson's SPRA which should be considered in decisionmaking. Qualitative factors that are discussed are consistent with SECY-14-0087. The identified qualitative factors do not have a commonly accepted quantitative measure, there is a lack of methodologies to accurately quantify the factor, and there is a lack of data to apply to a given quantification methodology. The identified qualitative factors have the potential to increase the safety significance of the insights from the SPRA. It should be noted that the insights discussed in the previous sections remain valid regardless of the impact of the qualitative considerations.

The following qualitative factors are supplemental to the quantitative insights that have been discussed in previous section:

- The dominant failure modes are such that the potential for fatalities and the need for retrieval of personnel cannot be overlooked. In response to an audit question from the NRC staff, the licensee stated that operator fatalities were possible if operators were working in the Class III TB when the structure fails. The licensee's discussions with the operations staff indicated that auxiliary operators are typically stationed outside of the TB. The licensee stated that there is not a guaranteed failure of actions requiring passage through the TB as fatalities and/or injuries may not occur and additional staff may be available. However, the potential for fatalities and the need for retrieval of personnel resulting in unforeseen complications exists.

- The failure of the dam will result in flooding downstream of the site and significantly challenge the implementation of the plant's emergency plan. The impact of flooding due to dam failure on the local population and any evacuation due to the challenges at the plant can impact the emergency planning efforts.

2.8. Integrated Decisionmaking

The review of the insights from the licensee's SPRA using the December 12, 2019, submittal and information available to the staff as part of its audit demonstrate that based on the re-evaluated hazard:

- Defense-in-depth, the second principle of risk-informed decisionmaking, is degraded due to non-negligible failure probabilities of the dominant risk contributors at low seismic accelerations, including that at plant's SSE. This includes an approximately 12 percent failure probability for the Robinson dam at the acceleration corresponding to the plant's SSE and a 15 percent failure probability at 0.3g acceleration. If the dominant risk contributors fail, they impact of redundant, diverse, and independent mitigation SSCs.

- Safety margins in the SSCs at the plant that are necessary for mitigation of seismic events are degraded, even at low seismic accelerations, including that at plant's SSE of 0.2g. Structural failure and soil liquefaction failure modes are dominant and have non-negligible failure probabilities even at the acceleration corresponding to the plant's SSE. If these failures occur, they result in exceedance of the design margins of several SSCs because the resulting loading is not included during the design of the SSCs. Further, the insights from the licensee's SPRA submitted in response to the 50.54(f) letter challenges the evaluations in the licensee's UFSAR and IPEEE submittal.

- The licensee's SPRA demonstrates that, consistent with guidance in RG 1.174, Revision 3, increased attention from NRC management is justified. Seismic risk is the largest contributor to the plant risk (the risk from seismic events is approximately 3 and 4 times the internal fire risk for CDF and LERF, respectively). The risk from seismic events at Robinson results in an exceedance of the 1E-4 and 1E-5 per year guidelines for CDF and LERF, respectively. Further, 83 percent of the SCDF and 61 percent of the SLERF is from low seismic accelerations of 0.12g to 0.45g ('bins' %G01 through %G07), which have occurrence frequencies in the E-4 and E-5 per year range. The contribution from seismic accelerations at and below the plant's SSE ('bins' %G01 through %03) is approximately 4 percent to SCDF (i.e., a SCDF of approximately 5.6E-6 per year).

- Qualitative factors exist that have the potential to increase the safety significance of the insights.

Therefore, based on integrated decisionmaking using the results and insights from Robinson's SPRA, the staff determined the need for increased management attention (1) to determine whether a condition of undue risk to public health and safety exists, (2) to pursue plant improvements (modifications) that provide substantial safety enhancement to address the risk demonstrated by the SPRA, and (3) to pursue appropriate interim actions to manage the seismic risk for the duration that permanent plant improvements are implemented.

3. Conclusions

Based on its review of the information on Robinson SPRA available to the staff, the NRC staff recommended:

- Increased management attention (1) to determine whether a condition of undue risk to public health and safety exists, (2) to pursue plant improvements (modifications) that provide substantial safety enhancement to address the risk demonstrated by the SPRA, and (3) to pursue appropriate interim actions to manage the seismic risk for the duration that permanent plant improvements are implemented.

- Communication of the Robinson Dam related information to the NRC's dam safety officer and, via the dam safety office, to any other federal and/or local agencies, as appropriate.

The technical review team met with the SMRP on multiple occasions to present the results of the review, including information provided by the licensee as part of the audit, and its recommendations. The SMRP members sought detailed information about the review and provided input to the technical team.

Based on the evaluation and recommendations discussed above, and supported by the increased attention from the SMRP, the staff considered potential modifications, including those proposed by the licensee, to address the impacts demonstrated by the SPRA. The SPRA submittal proposed plant modifications that could potentially provide substantial safety enhancement and reduce the risk associated with the reevaluated seismic hazard. The staff and SMRP received details of the modifications as part of the audit process. In its letter dated June 19, 2020, the licensee proposed regulatory commitments to complete four (4) permanent plant modifications. The letter also identified interim actions that will be taken by the licensee for the duration that the proposed permanent modifications are completed.

Enclosure 2 discusses the information provided by the licensee on the proposed permanent and interim modifications. As discussed in Enclosure 2, the proposed permanent modifications address the insights demonstrated by the SPRA and evaluated in this enclosure.

SUBJECT: H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 – STAFF REVIEW OF SEISMIC PROBABILISTIC RISK ASSESSMENT ASSOCIATED WITH REEVALUATED SEISMIC HAZARD IMPLEMENTATION OF THE NEAR TERM TASK FORCE RECOMMENDATION 2.1: SEISMIC (EPID NO. L-2019-JLD-0022) DATED SEPTEMBER 22, 2020

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