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

November 29, 2019

Mr. J. Ed Burchfield, Jr.
Site Vice President, Oconee Nuclear Station
Duke Energy Carolinas, LLC
7800 Rochester Highway
Seneca, SC 29672-0752

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 – 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-2018-JLD-0173)

Dear Mr. Burchfield:

The purpose of this letter is to document the staff's evaluation of the Oconee Nuclear Station, Units 1, 2, and 3 (Oconee), seismic probabilistic risk assessment (SPRA) which was submitted in response to Near-Term Task Force (NTTF) Recommendation 2.1 "Seismic." The U.S. Nuclear Regulatory Commission (NRC) has concluded that no further response or regulatory actions associated with NTTF Recommendation 2.1 "Seismic" are required for Oconee.

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* 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 an 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 21, 2018 (ADAMS Accession No. ML19004A127), Duke Energy Carolinas, LLC (Duke, the licensee), provided its SPRA submittal in response to Enclosure 1, Item (8) of the 50.54(f) letter, for Oconee. 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 ASME (American Society of Mechanical Engineers)/ANS (American Nuclear Society) 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 Oconee SPRA assessment is contained in Enclosure 1 to this letter. As described below, the NRC has concluded that the Oconee SPRA submittal meets the intent of the SPID guidance and that the results and risk insights provided by the SPRA support the NRC's determination that no further response or regulatory actions associated with NTTF Recommendation 2.1 "Seismic" are required.

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. ML14092A024), Duke submitted the reevaluated seismic hazard information for Oconee. The NRC performed a staff assessment of the submittal and issued a response letter on July 22, 2015 (ADAMS Accession No. ML15201A008). 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 Oconee.

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, Oconee was expected to complete an SPRA with an estimated completion date of December 31, 2018, which would also assess high frequency ground motion effects. In addition, Duke was expected to perform a limited-scope evaluation for the Oconee spent fuel pools (SFPs). This SFP limited-scope evaluation was submitted by letter dated December 4, 2017 (ADAMS Accession No. ML17348A075). The staff provided its assessment of the Oconee SFP evaluation by letter dated July 25, 2018 (ADAMS Accession No. ML18197A201).

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

In its letter dated December 21, 2018, Duke provided the SPRA submittal that initiated the NRC's Phase 2 decision-making process for Oconee. The NRC described this Phase 2 decision-making process in a guidance memorandum from the Director of the Japan Lessons-Learned Division to the Director of the Office of Nuclear Reactor Regulation (NRR) dated September 21, 2016 (ADAMS Accession No. ML16237A103). 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 Oconee is described below. Based on its evaluation, the staff recommended to the SMRP that Oconee be classified as a Group 1 plant and therefore, no further regulatory action was warranted.

EVALUATION

Upon receipt of the licensee's SPRA submittal dated December 21, 2018, a technical team of NRC staff members performed a completeness review to determine if the necessary information to support Phase 2 decision-making 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. On February 25, 2019, the technical team determined that sufficient information was available to perform the detailed technical review in support of the Phase 2 decision-making.

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 decision-making associated with the 50.54(f) letter. As stated in its submittal dated December 21, 2018, 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. Appendix A of the licensee's submittal also included the open SPRA finding level facts and observations (F&Os) along with licensee's dispositions. These elements were reviewed by NRC staff in the context of the regulatory decision-making associated with the 50.54(f) 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 50.54(f) letter. By letter dated July 11, 2017 (ADAMS Accession No. ML17192A168), the NRC staff confirmed that the audit process for the seismic hazard reevaluations applies to the Oconee 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 dated February 25, 2019, April 22, 2019, May 2, 2019, and May 22, 2019 (ADAMS Accession Nos. ML19056A455, ML19113A168, ML19122A256, and ML19142A158 (non-public), respectively), were sent to the licensee to support the audit. The licensee subsequently provided answers to the 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 December 21, 2018, SPRA submittal.

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. Except for one F&O regarding the use of a Unit 1 lead unit concept, the NRC staff identified no issues with the licensee's dispositions to these findings with respect to the SPRA submittal. For this one exception, the NRC staff evaluated the licensee's F&O resolution and concluded that a more detailed resolution would not impact the conclusions of the SPRA review and thus the use of the lead unit concept was acceptable for this submittal.

Based on the staff's review of the licensee's submittal, including the resolution of the peer review findings as described above, the NRC staff concluded that the technical adequacy of the licensee's SPRA submittal was sufficient to support regulatory decision-making associated with Phase 2 of the 50.54(f) letter.

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 licensees' 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 Oconee's submittal. As documented in the checklist, the staff concluded that the Oconee SPRA meets the intent of the SPID guidance, including the documentation requirements of the Code Case Standard.

Following the staff's conclusion on the SPRA's technical adequacy, the staff reviewed the risk and safety insights contained in the Oconee SPRA submittal. The staff also 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 Oconee 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. As part of this review the staff considered the modifications described in the Oconee SPRA which will be implemented in future refueling outages at each unit. By letter dated September 18, 2019 (ADAMS Accession No. ML19261D148), Duke provided regulatory commitments to perform certain plant modifications described in the SPRA submittal at the three Oconee units. These modifications are specifically targeted to reduce the SLERF value. The Oconee SPRA submittal demonstrated that the plant SCDF for all three units was not below the initial screening value in the staff memorandum dated August 29, 2017. Based on the SCDF results, the NRC staff utilized the Oconee SPRA submittal and other available information in conjunction with the guidance in the staff memorandum dated August 29, 2017, to complete a detailed screening with respect to SCDF for Oconee. The SCDF detailed screening concluded that Oconee should be considered a Group 1 plant because:

- Sufficient reductions in SCDF and SLERF cannot be achieved by potential modifications considered in this evaluation, other than those identified by the licensee and included as a regulatory commitment, to constitute substantial safety improvements based upon importance measures, available information, and engineering judgement;

- Additional consideration of containment performance, as described in NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," does not identify a modification that would result in a substantial safety improvement; and

- The staff did not identify any potential modifications that would be appropriate to consider necessary for adequate protection or compliance with existing requirements.

A discussion of the detailed screening evaluation completed by the NRC staff is provided in Enclosure 2 to this letter. With the planned modifications in place, the SLERF value was expected to show a substantial safety improvement, thus detailed screening was not performed for SLERF.

Based on the detailed screening evaluation and its review of the Oconee SPRA submittal, the technical team determined that recommending Oconee to be classified as a Group 1 plant was appropriate and additional review and/or analysis to pursue a plant-specific backfit was not warranted. This determination assumes that the modifications described in the licensee's submittal are implemented as planned.

As a part of the Phase 2 decision-making 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 Oconee review to the TRB with the Phase 2 recommendation that Oconee be categorized as a Group 1 plant, meaning that no further response or regulatory actions are required. The TRB members assessed the information presented by the technical team and agreed with the team's recommendation for classification of Oconee as a Group 1 plant.

Subsequently, the technical review team met with the SMRP and presented the results of the review including the recommendation for Oconee to be categorized as a Group 1 plant. The SMRP members asked questions about the review, as well as the risk insights and provided input to the technical team. The SMRP approved the staff's recommendation that Oconee should be classified as a Group 1 plant, meaning that no further response or regulatory action is required.

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 Oconee audit included a review of licensee documents through an electronic reading room. An audit summary document is included as Enclosure 3 to this letter.

CONCLUSION

Based on the staff's review of the Oconee 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 decision-making in accordance with the intent of the 50.54(f) letter. Based on the results and risk insights of the SPRA submittal, the NRC staff also concludes that no further response or regulatory actions associated with NTTF Recommendation 2.1 "Seismic" are required. The staff notes that this conclusion is dependent on the completion of the planned modifications, as described in the SPRA submittal.

Application of this review 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 Peter Bamford at (301) 415-2833 or via e-mail at Peter.Bamford@nrc.gov.

Sincerely,

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

Docket Nos. 50-269, 50-270, and 50-287

Enclosures:
1. NRC Staff SPRA Submittal Technical Review Checklist
2. NRC Staff SPRA Submittal Detailed Screening Evaluation
3. NRC Staff Audit Summary

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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 ASME-ANS [American Society of Mechanical Engineers-American Nuclear Society] 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 also 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-along 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 departures or differs 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]/Hybrid 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)

<table>
<tr><td>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.</td><td>No</td></tr>
<tr><td colspan="2">Notes from staff reviewer:

Deviation(s) or deficiency(ies) and Resolution: Findings related to SHA-B4 and SHA-B5 (20-1 and 20-2) were left unresolved by the licensee because NRC guidance, for the purposes of this review, allow the use of a standard seismic source model without update. The NRC's 50.54(f) letter specifically requests that licensees not update the source model to expedite staff review of seismic hazard.

Consequence(s): The licensee's approach to seismicity is consistent with NRC guidance and the licensee's seismic hazard was approved by staff for the purposes of the 50.54(f) response in a staff assessment dated July 22, 2015 (ADAMS Accession No. ML15201A008).</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 the SHA requirements 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 guidance in the SPID was followed for developing the probabilistic seismic hazard for the site.

• an alternate approach was used and is acceptable on a justified basis.</td><td>Yes

Yes

Yes

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

TOPIC 2: Site Seismic Response (SPID Section 2.4)

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.	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 all SRs under HLR-SHA-E 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 development of PSHA inputs and base rock hazard curves meets the intent of the SPID guidance or another acceptable approach.	Yes
• 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.	Yes
• 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.	N/A

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. 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. C) An alternative method has been used for this site. 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. If <u>yes</u>, the control point can be used in the SPRA and the NRC staff's earlier acceptance governs. If <u>no</u>, the NRC staff's previous reviews might not apply. The staff's review of the control point used in the SPRA is acceptable.	Yes
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	N/A

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.	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	
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:

Section 4.3 of the Oconee SPRA report describes the analysis of structures which support the safety-related components and systems. Table 4-2 of the report provides a summary of the structural modeling, the consideration of soil structure interaction and ground motion incoherence effects, structural damping, and parameters that were varied for response analyses for each of the buildings. New Finite Element Method (FEM) models were developed for the Auxiliary Building (Units 1, 2, and 3), Reactor Building (Unit 1), West Penetration Room (Unit 3), Standby Shutdown Facility, Turbine Building (Unit 1), Switchgear Enclosure (Blockhouses, Units 1 and 2), and Intake Structure (Unit 3). The Oconee SPRA report explains that for the Reactor Building, FEM models were used for internal structures and the Nuclear Steam Supply System, while the existing Lumped Mass Stick Model (LMSM) of the containment shell was revised to meet the SPID requirements. Soil structure interaction analysis was performed for all the buildings except for the West Penetration Room where fixed-base dynamic response analysis was performed. The Oconee submittal indicates that an existing model was enhanced for the Protected Service Water Building.

In addition to the buildings listed in Table 4-2, the Oconee SPRA report also identifies other buildings that support Seismic Equipment List (SEL) equipment. A review of supporting documents conducted during the NRC audit process shows that new FEM models were also developed for the Switchyard Relay House and the Keowee Intake Structure.

For computing ISRS, Section 4.3.1 of the Oconee SPRA report identifies that un-cracked and cracked concrete model analyses were used in the building response analyses to identify the local areas likely to crack and affect the input motion for the SEL components. For cracked concrete models, the structural stiffness was varied based on Section 3.3 of ASCE [American Society of Civil Engineers]/SEI [Structural Engineering Institute] Standard 4-16, "Seismic Analysis of Safety-Related Nuclear Structures."

The NRC staff used the audit process to confirm that the FEM and LMSM structural models are capable of capturing the overall structural responses for both vertical and horizontal components of ground motion. The staff also notes that torsional effects were considered, including out-of-plane response and in-plane diaphragm effects. Additionally, the soil-structure interaction analysis considered ground motion in all three spatial directions and the FEM models include modes greater than 20 Hertz (Hz), consistent with the SPID criteria. Based on its review, the staff concludes that provisions in Criteria 1-7: SPID Section 4.3.3, have been met.

Facts and observations (F&O) 24-3 (SFR-B3) recommended that potential effects of the vertical ground motion on seismic demand for SEL components in the Keowee Powerhouse be evaluated. In response to this F&O, the licensee generated vertical ISRS accounting for the variability in ground motion and uncertainties in structural stiffness. Fragilities developed for relays with vertical ISRS had negligible impact on seismic core damage frequency (SCDF) and seismic LERF (SLERF) values. Based on the licensee's SPRA report, supplemented by the review performed during the NRC staff audit process, the staff concludes that resolution to the F&O is acceptable and meets the SFR-B3 requirements.

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 requirement 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: West Penetration Room (Unit 3)	Yes
Structure #1: If used, is V_S > about 5,000 feet (ft.)/second (sec.)? Review of the Oconee SPRA report shows that the mean shear wave velocity of the rock in the area where the West Penetration Room is located is greater than 5,000 ft./sec.	Yes
If 3,500 ft./sec. < V_S < 5,000, was peak-broadening or peak shifting used?	N/A
Potential Staff Finding: The demonstration of the appropriateness of using this approach is adequate.	Yes

Notes from staff reviewer: The fixed-based dynamic analysis for the Unit 3 West Penetration Room is discussed in Section 4.3.2 of the Oconee SPRA report. The ISRS was developed considering: (i) structural models using FEM; (ii) lower bound (LB), upper bound (UB), and best estimate (BE) structural stiffness properties; and (iii) five earthquake input accelerations. Results were used to evaluate 84 percent non-exceedance probability ISRS for estimating fragility of components and systems in the building. The NRC staff used the audit process to confirm that the variation of structural stiffness satisfies the guidance in ASCE 4-16. Based on this approach for fixed-base dynamic analysis, the staff concludes that the response uncertainties were appropriately included in the ISRS. 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.	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 fixed-based dynamic analysis of structures for a site previously defined as "rock" adequately meets the intent of the SPID guidance.	Yes
• 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

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)? Potential Staff Findings: If a new UHS [uniform hazard spectra] or RLE [review level earthquake] is used, the shape is approximately similar to the spectral shape previously used for ISRS generation. If the shape is not similar, the justification for seismic response scaling is adequate. Consideration of non-linear effects is adequate.	No N/A N/A N/A N/A
Notes from staff reviewer: As explained in Section 4.3 of the Oconee SPRA submittal, new FEM models were developed for several facility structures including the Reactor Building, Auxiliary Building, and the Turbine Building, and soil-structure-interaction (SSI) analyses were performed for these structures considering variability in stiffness and soil parameters. No scaling was necessary to develop the ISRS. There were no peer review comments on SFR-B2.	
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 requirement SFR-B2 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 use of seismic response scaling adequately meets the intent of the SPID guidance. • 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 N/A N/A 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 requirements 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: Auxiliary Building (Units 1, 2, and 3) Structure #2: Reactor Building (Unit 1) Structure #3: West Penetration Room (Unit 3) Structure #4: Standby Shutdown Facility Structure #5: Turbine Building (Unit 1) Structure #6: 4kV Switch Gear Enclosure Unit 1 and 2 Blockhouse Structure #7: Protected Service Water Building Structure #8: Oconee Intake Structure (Unit 3)	Yes
Are all structures appropriately considered? 2. Are models adequate to provide realistic structural loads and response spectra for use in the SPRA?	Yes Yes

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 N/A

Notes from staff reviewer:

The Oconee SPRA report explains the response analysis of structures in Section 4.3. The structural response analysis to develop the FIRS and ISRS were based on SSI analysis for all buildings in Table 4-2, with the exception of the West Penetration Room where a fixed-base analysis was used. The response of fixed-base analysis is discussed in Topic #5 of this checklist. Uncertainties for the SSI analysis were addressed using three structural stiffness properties (LB, UB, and BE), three soil parameters (LB, UB, and BE) and five time histories. The report states that when the structural response is dominated by structural stiffness, the structural properties were varied and BE soil properties were used; on the other hand, when SSI effects dominated the response, soil properties were varied and BE structural properties were used. Reponses for four structures (Auxiliary Building, Reactor Building, West Penetration Room, and Turbine Building) had negligible effects from SSI. The response of the Standby Shutdown Facility, Switchgear Enclosure and Blockhouse, Protected Service Water Building, and Oconee Intake Structures were dominated by SSI effects. The NRC staff confirmed that the approach taken by the licensee appropriately considers uncertainties in the SSI analysis.

There are four peer review findings related to SRs under SFR-B.

F&O 22-1 (U2/3) SFR-B1 and F&O 22-4 (U3) SFR-B1 are associated with the use of the lead unit approach and possible conservatism associated with demand on SSCs. In response to F&O 22-1, it is stated in the disposition that the Turbine Building structural configuration was reviewed for all three units and the Turbine Building of Unit 1 was selected as lead unit because the dynamic response is expected to envelop the response of other two Turbine Building units. The staff observes that the conservatism in the seismic response and the demand on SSCs in the Turbine Building for Units 2 and 3 is consistent with the conservatisms associated with the lead unit concept utilized in the Oconee SPRA. Based on the description in the SPRA report, supplemented by the audit review, the staff agrees with the disposition that seismic response of Turbine Building Unit 1 meets the Category II (realistic seismic response) and Turbine Building Units 2 and 3 meet Category I (approximate seismic response) for SFR-B1.

F&O 22-4 was concerned that using the response of Block House for Units 1 and 2 (selected as lead unit) for estimating demand for the SSCs in the Blockhouse for Unit 3 is likely to be non-conservative because the Unit 3 Blockhouse structure is stiffer than the Unit 1 and 2 Blockhouse structure. The fundamental SSI frequency of the Unit 3 Blockhouse is about 15 Hz closer to the frequency of the peak input spectrum, whereas the SSI fundamental frequency of the Blockhouse for Units 1 and 2 is about 7 to 9 Hz. Thus, the ISRS in the Unit 3 Blockhouse will be potentially higher than in the Units 1 and 2 Blockhouse. Following the suggestion in the F&O, the licensee performed a sensitivity analysis where a scaling factor was developed, taking into

account the SSI effects of the soil strata to decrease the fragility of the components. The sensitivity studies showed marginal changes in the SCDF and SLERF values (Oconee SPRA Table 5.7.1) and no changes in the overall risk insights. Based on the SPRA submittal, supplemented by the audit review, the staff concludes that the seismic response of the Blockhouse for all three units meets Capability Category II for SFR-B1. F&O 22-2 SFR-B2 concerns SSI effects between massive buildings and smaller buildings (Turbine Building and Blockhouse; Auxiliary Building and Unit 3 West Penetration Room). Following the suggestion in the F&O, Duke used a fragility scaling factor and applied it to all SSCs in the Blockhouses and performed a sensitivity analysis. The results of the sensitivity analysis given in Oconee SPRA Table 5.7.1 shows an increase in overall SCDF and LERF; however, the licensee disposition stated that SSI effects did not change top risk contributors. Thus, no additional analysis was required. Based on the SPRA submittal, supplemented by the audit review, the staff concludes that the licensee's disposition is acceptable. F&O 24-2 SFR-B1 recommended evaluation of the potential for concrete cracking and building-to-building impacts at ground motion levels above the GMRS and to assess any impact on the top contributors in the Oconee SPRA. The concern was raised because the top contributors in the Oconee SPRA risk profile are dominated within the hazard range between 0.3 to 0.8 times the force of gravity (0.3g to 0.8g), whereas seismic response was evaluated at the GMRS (0.396g) ground motion level. The cracking in concrete buildings was addressed in the F&O disposition where it is stated that the building models with fixed base (rock foundation) are likely to fail under 0.8g horizontal ground motion while ground motion in the vertical direction will have negligible effects. Concrete cracking will cause increased damping in the building and lower the structural frequency, resulting in reduced ISRS demand on the components. Oconee determined that building-to-building interaction (pounding) would occur at the 0.8g level but SSCs sensitive to high frequency response from the pounding are not likely affected. Further, a sensitivity study performed for the engineered safeguards terminal cabinet (ESTC), a dominant contributor to SLERF, showed no significant change in the SCDF and a reduction in SLERF. This sensitivity study did not identify a change in the top risk contributors. Based on the licensee's SPRA submittal, supplemented by the audit review, the staff concludes that the F&O is acceptable. 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.	N/A

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: Section 4.4.1 of the Oconee SPRA report describes the methodology for screening high capacity SSCs in the SPRA evaluation. Capacity-based screening was used in accordance with SPID guidance in Section 6.4.3. The screening level High Confidence Low Probability of Failure (HCLPF) for rugged SSCs was determined to be 1.4g. The rugged components were identified based on analysis or engineering judgement including observations during walkdowns. The Oconee SPRA states that the rugged SSCs were not screened and their failures were retained in the SPRA. The screening level fragility was assigned to these SSCs in the SPRA model. The capacity-based screening process in Oconee SPRA is consistent with Section 6.4.3 of the SPID. There are no F&Os related to SFR-C1, SFR-C2, and SPR-B5. Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): None	
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	N/A

SR requirements SFR-C1, SFR-C2, and SPR-B5 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.	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. 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. C) An alternative method has been used appropriately for developing full seismic fragility curves.	Yes Yes Yes No
Notes from staff reviewer: The Oconee SPRA report explains in Section 4.4.1 that CDFM/hybrid procedures were used to update and refine initial conservatively biased representative fragilities. The report states that during iterative quantification analysis the fragilities of dominant contributors of risk were refined either by CDFM or separation of variable (SOV) approaches. Fragility analysis of the top 10 dominant contributors of risk in Tables 5.4-2 and 5.5-2 of the Oconee SPRA were based on either representative or SOV methods. The Oconee SPRA report, Section 4.4.2.2 states that SPID Table 6-2 was used in development of fragility by the hybrid method along with the use of the SPID representative values. During the audit process the NRC staff reviewed supporting documents to confirm this. The staff confirmed that the CDFM/Hybrid methodology followed EPRI TR-1002988, "Seismic Fragility Applications Guide," and EPRI NP 6041-SL, "A Methodology for Assessment of Nuclear Plant Seismic Margin." In addition, the staff reviewed the resolution of F&O 20-9, Supporting Requirement SFR-E4, with regards to the seismic fragility evaluation of Jocassee Dam. The NRC staff concluded that the licensee's F&O disposition was 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 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 use of the CDFM/Hybrid method for seismic fragility analysis meets the intent of the SPID guidance.	Yes
• 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	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. Potential Staff Findings: 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: Relay chatter caused by high frequency ground motions are addressed in SPRA Section 4.1.2. In the disposition of F&O 25-13 SPR-B6, the licensee stated that relays were screened based on circuit analysis and operator action. The relays were screened consistent with methodologies previously used and in accordance with the ASME/ANS standard. A sensitivity analysis was performed for operator action with no new risk insights derived. The Oconee SPRA also states that the fragility of relays not screened out was evaluated using the SOV method. During the NRC staff audit, the licensee clarified that additional screening and circuit analysis was performed, resulting in no new risk insights. Based on the licensee's submittal, supplemented by the audit review, the staff concludes that devices potentially sensitive to high-frequency were evaluated per EPRI Phase 2 testing and in accordance with SPID Section 6.4.2. There are no F&Os related to SFR-E5. 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 requirement SFR-E5 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 fragility analysis of SSCs sensitive to high frequency seismic motion meets the intent of the SPID guidance.	 N/A N/A Yes

• 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.	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 of these are used, the NRC reviewer should use the following elements of the checklist.	
i) Circuit analysis: The seismic relay-chatter analysis of some relays relies on circuit analysis to assure that safety is maintained. (A) If no, then (B) is moot. (B) If yes: Potential Staff Finding: The approach to circuit analysis for maintaining safety after seismic relay chatter is acceptable.	Yes Yes
ii) Operator actions: The relay-chatter analysis of some relays relies on operator actions to assure that safety is maintained. (A) If no, then (B) is moot. (B) If yes: Potential Staff Finding: The approach to analyzing operator actions for maintaining safety after seismic relay chatter is acceptable.	Yes Yes
Notes from staff reviewer: In the disposition of F&O 25-13 SPR-B6, the licensee stated circuit analysis and operator actions were used to screen out relays. Through the audit process, staff reviewed selected relay fragility analysis using the SOV method and found it to be consistent with the standard approach. 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 requirement 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 separation of variables (SOV) fragility calculations would make a significant difference in the SPRA results have been selected for SOV calculations."

<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.

B) The recommendations in Section 6.4.1 were not followed, but the analysis is acceptable on another justified basis.</td><td>No

No

Yes</td></tr>
<tr><td colspan="2">Notes from staff reviewer:

The fragility for the top ten contributors of SCDF (Oconee SPRA Table 5.4-2) and SLERF (Oconee SPRA Table 5.5-2) include a combination of representative and SOV fragilities. The SPRA submittal, including the licensee's dispositions to finding level F&Os 23-3 and 23-4 from the SPRA peer review, describes the licensee's process for selecting dominant risk contributors for more detailed fragility analysis. Section 4.4.1 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." Section 4.4.2 of the submittal further explains that the representative fragilities were developed based on a combination of design information and calculations, past Oconee seismic evaluations, judgements from Oconee seismic walkdowns, past fragility estimates and experience, recent Oconee assessments addressing NTTF recommendations, and the results of the Oconee seismic response analysis described in Section 4.3 of the SPRA submittal. For the top contributors that were "conservatively biased, more detailed fragilities were developed as SOV or CDFM fragilities." The SPRA was then quantified with the revised fragilities, and the iterative process of SPRA quantification and development of more detailed fragilities was repeated until "the top risk contributors were ultimately characterized with a realistic and plant-specific fragility." Section 4.2.2 of the SPRA submittal clarifies that the fragilities used for the results reported in the submittal were a combination of representative fragilities and more detailed CDFM and SOV fragilities, and that the majority of the detailed fragilities were performed using the SOV approach. During the audit, the NRC staff confirmed that almost all the detailed fragilities were calculated using the SOV approach, with a relatively few calculated using CDFM.</td></tr>
</table>

The NRC staff finds the licensee's methodology acceptable because it utilizes screening fragilities initially followed by detailed fragility analysis of SSCs that are significant contributors to seismic risk, and because sensitivity performed by the licensee for the impact of additional detailed fragilities in lieu of representative fragilities demonstrated that the NRC staff's decision for this submittal would not be changed as discussed in Enclosure 2.

Deviation(s) or deficiency(ies) and Resolution:

Tables 5.4-2 and 5.5-2 of the SPRA submittal show that about one-half of the top 10 contributors to both SCDF and SLERF are based on representative fragilities rather than detailed fragilities. The use of representative or generic fragilities was the topic of finding level F&Os 23-3 and 23-4 from the SPRA peer review. The licensee explained in the submittal that the representative fragilities were developed with a conservative bias. During the audit process, the NRC staff questioned whether the use of representative fragilities for some of the top contributors would mask the risk importance of other SSCs. For clarification, the licensee provided the results of a sensitivity study posted on the eportal that increased the fragility capacities of the top risk contributors for both SCDF and SLERF for which representative fragilities were used. The results of the sensitivity study showed generally an increase in risk significance of SSCs with fragilities developed using the SOV method and decrease in the risk significance of SSCs with representative fragilities, some of which dropped out of the top 10 contributor list. As a result, other SSCs took their place in the top 10 list of risk contributors. For SCDF, the SSCs that rose into the top 10 list were relay chatter failure events and control rod drive failure. For SLERF, the SSCs that rose into the top 10 list were control rod drive failure, main condenser failure, failure of the 4160V switchgear, failure of the 120 Volts-alternating current (120V-AC) power panel board, and a relay chatter failure event.

During the audit the licensee also provided the list of top 10 risk contributors for SCDF and SLERF for sensitivity case SY-1p reported in the submittal. This sensitivity case increased the fragilities for the reactor coolant pumps and reactor coolant loop piping, which were evaluated in the SPRA using representative fragilities. The results of the sensitivity showed that the reactor coolant pump and reactor coolant loop piping SSCs dropped out of the top 10 risk contributor lists for SCDF and SLERF. While these SSCs were replaced by other SSCs, the top 10 list of risk contributors did not change appreciably.

Review of the top 10 risk contributors from these sensitivity studies, which were evaluated in accordance with the NRC's detailed screening methodology discussed in Enclosure 2, determined that the NRC staff's assessment and conclusions for this submittal reported in Enclosure 2 were not impacted by the results of the sensitivity studies.

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 and the requirements in the SPID. No requirements in the Standard specifically address this Topic.	No

• although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.	Yes
• 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.	No
• 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. Potential Staff Findings: 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.	Yes
B) The LERF analysis does not follow the guidance in Table 6-3 but the analysis is acceptable on another justified basis.	No

Notes from staff reviewer:

Section 4.1 of the submittal describes the development of a SEL for Oconee, including identifying SSCs for achieving safe shutdown following a seismic event and those that mitigate radioactivity releases if core damage occurs. A single SPRA model was developed based on Unit 1 to represent all three units (referred to as the lead-unit approach). The SEL was developed for Unit 1 and was expanded to include SSCs from Units 2 and 3 that did not have a Unit 1 equivalent. For specific SSC failures, the lowest seismic fragility for each SSC across all three Oconee units was developed and utilized in the SPRA. Section 4.2 describes several walkdowns performed on all three units, including walk-bys of Unit 2 and 3 SSCs to assess their similarity to the Unit 1 (lead-unit) SSCs.

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 of the top 25 LERF cutsets identified in Table 5.5-1 are either containment isolation failure or containment bypass events due to seismically-induced failure of a single containment isolation valve.

As discussed under Topic 14, a peer review of the Oconee SPRA was performed. The licensee provided the LERF-related F&Os from this peer review and dispositions for each. For F&Os 25-3 and 25-4, regarding SR SPR-E6, the peer review assessed this SR to be CC-I because the Oconee SPRA uses a conservative method for evaluating LERF and does not quantify the LERF contributors identified in the PRA standard or ASME/ANS RA-S Case 1 (ASME/ANS RA-S Case 1, 2017). More specifically, the licensee used the simplified event tree and LERF analysis method described in NUREG/CR-6595, "Measures of Risk Importance and Their Applications," Revision 1 (ADAMS Accession No. ML071690031). Table B-4 of RG 1.200, in relation to SR LE-A1, states that NUREG-6595 is inadequate for CC-II and Section 6.6.1 of the EPRI SPID states that, where feasible, the evaluations are expected to meet CC-II in accordance with the ASME/ANS PRA standards. However, with regards to SPRA submittals in response to NTTF Recommendation 2.1, Section 6.6.1 of the SPID provides the following additional clarification: "For this application, which is aimed at developing an improved understanding of the impact of new seismic hazard estimates, screening approaches will be used to limit the scope of detailed analyses for some specific elements of the seismic PRA. Where more detailed analyses are essential to achieve an adequate level of understanding (e.g., with respect to "realism"), these analyses will be performed or alternative measures will be taken (such as making plant changes to address the impacts)."

During the audit, the licensee identified 11 LERF-related SRs that are CC-I, of which 9 had finding level F&Os written against them. The licensee provided the rationale for the lack of impact of meeting the LERF-related SRs at CC-I. Based on information provided by the licensee during the audit, the NRC staff concludes that the CC-I designation for each of these SRs will not impact the staff's decisions on the submittal because the plant-specific evaluations for SLERF will not impact the dominant contributors for this submittal. SLERF is dominated by the seismically-induced failure of a single isolation letdown valve at each unit to close (or failure of the associated control cabinets), resulting in containment isolation failure or containment bypass (nearly 70 percent of the SLERF and all of the top 25 SLERF contributing cutsets are related to direct containment bypass). The licensee submitted a regulatory commitment to implement plant modifications to provide a means of alternate letdown isolation in each unit (ADAMS Accession No. ML19261D148). According to the licensee's commitment letter, the modification results in a significant reduction in SLERF. During the audit, the NRC staff reviewed the changes made to the SPRA to model this modification and identified no significant issues with the modeling. Based on the limited, if any, impact on the dominant SLERF contributors, the context of the decision for this submittal, and the commitment by the licensee to implement the modifications to provide an alternate means of letdown isolation in each unit, the NRC staff finds the licensee's use of the simplified SLERF model acceptable for this 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 SR requirements 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 SPRA submittal follows the recommendations of Section 6.7 of the SPID. Section 5.2 and Appendix A of the submittal describe the peer review process used to establish the technical adequacy of the SPRA. The SPRA peer review was conducted in June 2018 against the CC-II supporting requirements of PRA Standard ASME/ANS RA-S Case 1, and associated NRC clarifications, and in accordance with the peer review characteristics and attributes described in RG 1.200, Revision 2. ASME/ANS RA-S Case 1 has been accepted by the NRC for use in regulatory applications, subject to certain qualifications and clarifications. The peer review team utilized the peer review process defined in NEI 12-13.

The SPRA submittal states that the peer review team members met the required qualifications for peer reviewers in accordance with Section 1-6.2.2 of the ASME/ANS PRA Standard (ASME/ANS Addendum A 2009 and ASME/ANS Addendum B 2013) and the guidelines of NEI 12-13, and that the peer reviewers were independent of the Oconee PRA development. During the audit, the NRC staff noted that two of the three reviewers focused on review of the fragility analysis (SFR technical element) had SQUG experience.

All elements of the seismic PRA were peer reviewed, including those identified in Section 6.7 of the SPID. Appendix A of the SPRA submittal clarified that the peer review team performed the review against a modified version of the SFR-E3 for CC-I. The modified version of SFR-E3 for CC-I was provided in the SPRA submittal as well as the corresponding inquiry number with the JCNRM, which maintains the PRA Standard. A total of 47 finding-level SPRA F&Os generated by the peer review are documented in Table A-2 of the SPRA submittal along with the licensee's dispositions for the submittal.

The dispositions to F&Os 25-14 (SR C-SPR-B1) and 26-2 (SR C-SPR-A1), regarding applicability of the accident sequences, system logic, and initiating events in the internal events PRA model to the SPRA, state that the majority of internal event assumptions required no changes to the SPRA model but that "the remainder were considered for inclusion." It is unclear from this disposition if assumptions potentially important to the SPRA were excluded from the SPRA model. During the audit, the licensee stated that a systematic and thorough review of these assumptions, or potential sources of uncertainty, was conducted for impact on the SPRA model and provided a table listing those uncertainties and their dispositions. The NRC staff did not identify any issues with the dispositions for F&Os 25-14 and 26-2 for this SPRA submittal.

F&O 25-10 concerns the identification and screening of seismically-induced fires. During the audit, the NRC staff reviewed Appendix J, Section 1.32 of OSC-11576, "Oconee SPRA F&O Resolution Notebook," which identified four seismically-induced fire scenarios. As part of the audit, the licensee clarified that all seismically-induced fires were screened from the SPRA model. Based on information provided during the audit, the four scenarios of specific concern were cable fires caused by sparking due to differential movement of associated cable trays. These fire scenarios do not result in additional SPRA scenarios because functional failure of the SSCs associated with these cables are already modeled in the SPRA with detailed fragilities developed for the cable tray supports and conservatively assuming that failure of the supports and associated SSCs leads directly to core damage. With this additional information as confirmation, the NRC staff concludes that seismically-induced fires have been appropriately considered in the Oconee SPRA.

The internal events PRA (IEPRA) model 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. The NRC staff's comments on NEI 12-13 stated that the SPRA peer review team is to review all of the IEPRA F&Os and determine whether the resolutions were appropriate and in accordance with the PRA standard. 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 and the IEPRA F&O closure report. Regarding SR SPR-B2, the SPRA peer review report states that "all internal events findings not related to LERF have been closed out via the closure review process in Appendix X to NEI 05/04/07-12/12-13." The IEPRA F&O closure report is dated May 2016. Since the closure review was conducted prior to the approved guidance provided in the NRC letter dated May 3, 2017, regarding the updated NEI process detailed in their letter dated February 21, 2017, the NRC staff reviewed the IEPRA F&Os, and the associated dispositions by the licensee, during the audit for potential impact on the SPRA submittal. With the exception of F&O QU-A1-01, discussed in the next section, the NRC staff identified no issues with the licensee's dispositions of the IEPRA findings with respect to the SPRA submittal.

Deviation(s) or deficiency(ies) and Resolution:

The disposition to SPRA F&O 25-1 states the required truncation value based on convergence was not achieved in accordance with SR QU-B3 of the ASME/ANS PRA standard. The disposition states that the truncation value used to quantify seismic risk could only achieve a difference of 5.4 and 9 percent for CDF and LERF, respectively, compared to 5 percent as recommended by SR QU-B3. The NRC staff questioned this during the audit process, inquiring whether the risk results using truncation values that do not achieve convergence could be unstable (i.e., inaccurate), potentially impacting the dominant risk contributors and corresponding importance measures. In response to this question, the licensee provided the results of a sensitivity study on the eportal. This study compared the Fussell-Vesely (F-V) risk importance for the dominant risk contributors between the baseline truncation results provided in the SPRA submittal and the "lowest achievable" truncation results that represent the 5.4 and 9 percent decade iteration change for CDF and LERF respectively (hence, the SPRA results provided in the submittal are not for the truncation value that produced the 5.4 and 9 percent convergence, but are for a decade higher truncation value). The 15 dominant risk contributors for SCDF and SLERF did not change significantly for the quantification with the "lowest achievable" truncation compared to those reported in the submittal (14 of the top 15 for SCDF were the same and 11 of the top 15 were the same for SLERF). Thus, the NRC staff determined that the changes to the importance measures for the dominant risk contributors in the sensitivity study did not impact its decisions for this SPRA submittal and that the SPRA quantification that did not achieve the recommended convergence criteria in SR QU-B3 was acceptable for this application.

SPRA F&O 25-13 concerns the criteria used to screen 124 relays from the SPRA model. The F&O disposition in the submittal states that an updated analysis was performed that determined 44 relays could be screened based on circuit analysis. The remaining 80 relays were screened based on highly reliable operator recovery actions. It was unclear to the NRC staff if those operator actions were incorporated into the SPRA model used for the submittal. During the audit the licensee provided an updated analysis of these relays that included circuit analysis, which provided the basis to qualitatively screen 105 of the original 124 relays. The qualitative screening criteria included screening based on the configuration which would require two channels to chatter simultaneously to result in an undesired state for an SSC. For the remaining 19 relays the licensee performed a sensitivity study that included operator actions to reset these relays after seismically-induced failure due to relay chatter. The results of the sensitivity study showed an increase in SCDF and SLERF of 2 percent and 0.5 percent respectively. In addition, the top 10 most risk-significant SSCs did not change from those provided in the submittal, and changes to their F-V values were small. The staff review of the sensitivity case concludes that any additional relays incorporated in the SPRA following a detailed review of the basis and implementation of the qualitative screening criteria would not likely significantly impact the decision-making factors for this review. Therefore, based on the SPRA description, as clarified in the audit process, the NRC staff concludes that any refinements to the licensee's treatment of unscreened relays in the SPRA model would not impact the staff conclusions in evaluating the submittal.

The closure review team noted for IEPRA F&O QU-A1-01, regarding the use of Unit 3 as the 'lead unit' for the IEPRA model, that there are differences between the units for both the reactor coolant pump (RCP) seal modeling and internal flooding. However, Section 4.1.1 of the SPRA submittal states that the SPRA model uses a Unit 1 lead unit concept. During the audit, the licensee clarified that for the internal events PRA, unit-specific models were developed starting in 2011 and Unit 3 was no longer the lead unit. However, both the internal flooding and fire PRA models reinstituted the lead unit concept and so it was decided to do the same for the SPRA model. For internal flooding scenarios, Units 2 and 3 were analyzed for additional impacts and reflected in the lead unit model. For specific SSC failures, the lowest seismic fragility for each SSC across all three Oconee units was developed and utilized in the SPRA. Section 4.2 of the submittal describes several walkdowns performed on all three units, including walk-bys of Unit 2 and 3 SSCs to assess their similarity to the Unit 1 (lead-unit) SSCs. All shared systems were individually analyzed for seismic impact and included in the lead unit SPRA model. Hence, the SPRA and results presented in the submittal does not

<table>
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specifically represent any of the three Oconee units, but rather represents an aggregation of the three units.
During the audit, the licensee also provided the results of a sensitivity study evaluating unit-specific differences in the configuration of SSCs. This study generally showed little impact on SCDF and SLERF for the unit-specific configurations, except for the ESTCs which were shown to have a large contribution to SLERF. The licensee provided a table of the top 19 most risk-significant SSCs for SLERF comparing the F-Vs for Units 1, 2, 3, and the Lead Unit used for the submittal. While the unit-specific F-V values were shown in many cases to be higher than the Lead Unit F-V values reported in the submittal, generally these differences were less than 20 percent. Several SSCs were shown to have unit-specific F-V values that were a factor of up to three greater than the Lead Unit F-V value. The NRC staff evaluated these unit-specific differences. The ESTCs continued to have a large contribution to the SLERF results even when unit-specific differences were included. The licensee submitted a regulatory commitment to implement plant modifications to provide a means of alternate letdown isolation in each unit. According to the licensee's commitment letter, the modification results in a significant reduction in SLERF. In the context of the decision for this submittal, the staff concludes that including unit-specific differences would not have identified any substantial safety improvements other than those identified by the licensee in the regulatory commitment. Therefore, the NRC staff concludes that use of a Lead Unit SPRA is acceptable for the purposes of decision-making for this submittal.
Consequence(s): N/A
</td></tr>
<tr><td>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).</td><td>Yes</td></tr>
<tr><td>the licensee's peer-review process does not meet the intent of the SPID guidance as supplemented by NRC staff comments in the NRC letter dated March 7, 2018 (NRC 2018a, 2018b), but is acceptable on another justified basis.</td><td>N/A</td></tr>
</table>

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 of 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 the 10 CFR 50.54(f) letter (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 is consistent with that specified in Section 6.8 of the SPID. The SPID requires that there be sufficient information to assess the results of all key aspects of the analysis. Section 5.3.2 of the submittal identifies and discusses the key assumptions and sources of uncertainty for seismic PRA. Sections 5.4 and 5.5 of the submittal present and discuss the results. The staff notes, 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 uses a lead unit approach as discussed under Topic #14. Section 5.6 of the submittal presents the SPRA quantification uncertainty results for SCDF and SLERF (i.e., the median (50 percent), and the 5^{th} and 95^{th} percentiles). However, according to the NRC staff memorandum dated August 29, 2017, 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 SCDF and SLERF as 5.7E-05 per year and 2.8E-05 per year, respectively. These values are referenced in the NRC staff's screening evaluation reported in Enclosure 2.

According to Section 5.1.2 of the SPRA submittal, Diverse and Flexible Coping Strategies (FLEX) is credited in the SPRA to provide primary injection for small-small loss of coolant accident (SSLOCA) mitigation. In addition, Section 5.3.2 of the submittal discusses the implicit credit in the SPRA that the Phase 3 FLEX equipment and supplies from the regional response center will be available within 24 hours to justify the assumption that actions to stabilize the plant would be successful to prevent reaching core damage after 36 hours for SSLOCA scenarios in which there is no depressurization (this issue was the topic of SPRA F&O 25-7). Sensitivity Case HR-2a shows that there is no change in either SCDF or SLERF from removing credit for FLEX. During the audit the licensee clarified that the sensitivity analysis removed all on-site and off-site FLEX credit (i.e., both quantitative and qualitative credit) by assigning the sequences with FLEX credit to core damage, thus confirming appropriate treatment of FLEX in the model.

Sections 5.4 and 5.5 of the submittal state that the point estimate SCDF and SLERF are 3.2E-05 per reactor-year and 1.3E-05 per reactor-year, respectively. However, the top 15 SCDF cutsets listed in Table 5.4-1 cumulatively add to 3.29E-05 per year (top 20 cutsets cumulatively equal 4.19E-05). Similarly, the top 8 SLERF cutsets listed in Table

5.5-1 add to 1.37E-05 (top 10 cutsets equal 1.65E-05 per year). During the audit the licensee clarified that the difference in the values was due to the quantification approach used, which was less refined for the cutset determination. The licensee also clarified that each seismic bin was quantified separately, which produced individual cutset files for each bin, and that the cutset files for all the bins were combined into one file and processed for importance measures. This clarification resolves this issue for the SPRA submittal. The results of Sensitivity Case IE-1d provided in Section 5.7 of the submittal showed a 98 percent increase in SCDF and 14 percent increase in SLERF when the number of hazard intervals is increased from 10 to 12. However, the submittal does not show how this change impacts the risk-significant SSCs and associated F-V values. The NRC questioned this during the audit process. The licensee explained that incorrect model files were used in developing the results for Sensitivity Case IE-1d. This sensitivity case was re-quantified with the correct files and updated results provided to the staff for review. The results showed a small decrease in SCDF and SLERF and confirmed to the staff that the results of this case do not impact the regulatory decision-making for this submittal. Table 5.7.1 of the SPRA submittal provides a summary of SPRA sensitivity cases with the associated changes in SCDF and SLERF. During the audit, the licensee clarified that the "% Delta CDF" was calculated by dividing the sensitivity case SCDF by the base case SCDF of 3.53E-05 per year (shown in the third row of the table) and multiplying the result by 100; SLERF was similarly calculated. These values do not account for the 0.9 availability factor used to develop the point estimate SCDF and SLERF values discussed above. The licensee's explanation clarifies the SPRA submittal and is acceptable to the staff. Deviation(s) or deficiency(ies) and Resolution: None Consequence(s): None	
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.	Yes
• The licensee's documentation does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

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 N/A
Plant will: • complete modifications by the Fall 2022 refueling outage at Oconee, Unit 1	
Notes from the Reviewer: The licensee submitted a regulatory commitment to implement plant modifications to provide a means of alternate letdown isolation in each unit (ADAMS Accession No. ML19261D148). According to the supplement the modification results in a substantial reduction in SLERF. The staff credited the proposed modifications being implemented in accordance with the licensee's regulatory commitment during its review and decision-making on this SPRA submittal. Refer to Enclosure 2 for the detailed screening evaluation. Deviation(s) or Deficiency(ies), and Resolution: None Consequence(s): N/A	
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, 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, 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 2102" 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. ML18025C022)

NRC Staff SPRA Submittal Detailed Screening Evaluation

Introduction

The Oconee Nuclear Station, Units 1, 2, and 3 (Oconee) Seismic Probabilistic Risk Assessment (SPRA) report (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19004A127) indicates that the mean seismic core damage frequency (SCDF) is 5.7E-05/reactor-year (/rx-yr) and the mean seismic large early release frequency (SLERF) is 2.8E-05/rx-yr for each of the three units. The mean SCDF and SLERF values are not provided in the SPRA report, however the 5 percent, 50 percent, and 95 percent values were provided. The staff estimated the mean SCDF and mean SLERF for each unit based on the information in the submittal and confirmed these values during the audit. The NRC staff compared these values against the guidance in NRC staff memorandum dated August 29, 2017, 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" (ADAMS Accession No. ML17146A200; hereafter SPRA Screening Guidance), which establishes the 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, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," Revision 4 (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. 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 Oconee 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 should be performed. The detailed screening shows that Oconee should be considered a Group 1 plant because:

- The modifications identified by the licensee and included in a regulatory commitment (ADAMS Accession No. ML19261D148) are the same substantial safety enhancement

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).

that would have been identified by the staff through its detailed evaluation. Additional potential modifications which could result in sufficient reductions in SCDF and SLERF to constitute substantial safety improvements were not identified based upon importance measures, available information, and engineering judgement;

- Additional consideration of containment performance, as described in NUREG/BR-0058, does not identify a modification that would result in a substantial safety improvement; and

- The staff did not identify any potential modifications that would be appropriate to consider necessary for adequate protection or compliance with existing requirements.

As such, additional refined screening, or further evaluation, was not required.

Detailed Screening

Duke Energy Carolinas, LLC (Duke, the licensee), in performing its seismic analysis in response to the Near-Term Task Force Recommendation 2.1, and the NRC in conducting its review, did not identify concerns that would require licensee action above and beyond existing regulations to maintain the level of protection necessary to avoid undue risk to public health and safety. In addition, there were no issues identified as non-compliances with the Oconee licenses, or the rules and orders of the Commission. For these reasons, the licensee and the staff did not identify a potential modification necessary for adequate protection or compliance with existing requirements. However, the licensee identified a plant modification at each unit to provide a means for alternate letdown isolation to mitigate the risk contribution from seismically-induced failure of certain associated cabinets that could fail the signal to close the letdown isolation valves. The licensee provided a regulatory commitment in a supplemental letter dated September 18, 2019 (ADAMS Accession No. ML19261D148), to implement these plant modifications. The NRC staff credited the licensee's self-identified modifications in its evaluation as explained in the following paragraphs.

The detailed screening uses information provided in the Oconee 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, like 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). 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 Oconee 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 *Application for Renewed Operating Licenses, Oconee Nuclear Station Units 1, 2, and 3*, dated June 1998[2], was used to calculate the RRW threshold. For this analysis, the NRC staff determined the RRW threshold from the SCDF-based MACR to be 1.07. The MACR calculation includes estimation of offsite exposures and offsite property damage, which captures the impact of SLERF. Therefore, separate SLERF-based MACR calculations

2. Available online at https://www.nrc.gov/reactors/operating/licensing/renewal/applications/oconee/exhibitd.pdf

were not performed. The target RRWs based on the mean and 95th percentile SCDF and SLERF were also calculated by the NRC staff and ranged between 1.02 and 1.21.

Section 5 of the Oconee SPRA report included tables listing and describing the structures, systems, and components (SSCs) 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 value for each. The NRC staff utilized the F-V values to calculate the RRW, the maximum monetary value of completely eliminating the failure, 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 seismically-induced failures that contribute to SCDF have an RRW greater than about 1.02 and those that contribute to SLERF have an RRW greater than about 1.01. 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. No single SPRA model element or contributor exceeded the mean target RRW for SCDF, whereas six seismically-induced failures exceeded the mean target RRW for SLERF.

The elements that contribute the greatest to SLERF are seismically-induced failure of the engineered safeguards terminal cabinet (ESTCs) for each unit (ESTC1, ESTC2, and ESTC3) due to either failure of the anchorage or functional failure which, in turn, fails the signal to close the letdown isolation valve (HP-5) resulting in containment bypass. All the top 25 SLERF cutsets reported in Table 5.5-1 of the submittal include failure of the ESTC cabinets. To mitigate this vulnerability, the licensee has submitted a regulatory commitment to implement plant modifications to provide a means of alternate letdown capability for each unit. During the audit process, the licensee explained that plans for the modification involve installation of a temperature transmitter device that will close the letdown isolation valve automatically on loss of power. The results of Sensitivity Case MOD reported in Table 5.7.1 of the submittal show that this modification would substantially reduce SLERF (based on the point estimate of the SLERF). The NRC staff notes that the mean SLERF is significantly higher than the point estimate SLERF but, because of the substantial SLERF reduction, the licensee's use of point estimates rather than mean values is not expected to impact the identification of the potential modifications by the licensee and the staff's conclusions for this SPRA submittal. The substantial reduction in SLERF was further confirmed by the licensee in the supplement documenting its regulatory commitment. The NRC staff's review of other SLERF basic events did not identify additional potential cost-justified substantial safety improvements besides the ones committed to by the licensee.

With regards to contributors to SCDF, the NRC staff considered 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 about 2.9E-05/yr. A review of these model elements reveals that any modification or set of modifications to achieve a SCDF reduction of at least 1.0E-05/rx-yr will have to mitigate or prevent multiple failure types (e.g., seismically-induced failures, random failures[3], and failure of operator actions) and failure modes (e.g., seismically-induced structural failures of multiple SSCs and seismically-induced functional failures of multiple SSCs).

3. The licensee provided information on random failures and operator actions that are not due to the seismic event in its submittal. The staff included this information as an aid to help identify potential modifications that could reduce the overall SCDF and/or SLERF.

The highest contributor to SCDF was seismically-induced loss of offsite power (SF-STY-OSP) due to failure of the 230 kilovolt (KV) switchyard or the transformer yard and, according to Table 5.4-1 of the submittal, is a contributor to 17 of the top 25 SCDF cutsets. During the audit, the licensee explained that SF-STY-OSP represents seismically-induced loss of offsite power from both the plant switchyard and from offsite power lines and that the fragility used in the SPRA is a single representative fragility representing both. As a result, the NRC staff did not pursue potential improvements to SF-STY-OSP. The second highest contributor to SCDF was seismically-induced small loss of coolant accidents (SF-SLOCA), which is a contributor to 11 of the top 25 SCDF cutsets. During the audit, the licensee explained that SF-SLOCA represents SLOCA events at many locations through the plant and that the fragility used in the SPRA is a single representative fragility used for all SLOCA failures. Because the implementation cost to address multiple sources of SLOCA failures is expected to far exceed the monetary value of any beneficial improvement, the NRC staff also did not pursue potential improvements to SF-SLOCA.

Excluding the contribution from SF-STY-OSP and SF-SLOCA, the total contribution to SCDF from the remainder of the seismically-failed elements listed in Table 1 is 1.6E-05/yr. Given that plant modifications would need to address at least four of these elements to achieve a risk reduction of at least 1.0E-05/yr, that no single modification was identified to address at least four of these elements, and that the implementation cost to address multiple elements is expected to far exceed the monetary value of any beneficial improvements, the NRC staff did not pursue potential improvements to these elements.

Furthermore, the results of Sensitivity Case SY-3b reported in Table 5.7.1 of the submittal shows that implementing operator actions to recover all relay chatter failure events would not achieve a reduction in SCDF of 1.0E-05/year. Based on this, the NRC staff did not pursue a potential improvement to implement operator actions to reset seismically-induced relay chatter failures.

Based on the analysis described above, the NRC staff concludes that the modifications identified by the licensee and included in its regulatory commitment are the same substantial safety enhancement that would have been identified by the staff through its detailed evaluation. Additional potential modifications which result in sufficient reductions in SCDF and SLERF to constitute substantial safety improvements were not identified.

In accordance with Section 3.3.2 of NUREG/BR-0058, Rev. 4, the NRC staff further evaluated Oconee accident sequences impacting the conditional probability of early containment failure or bypass (CPCFB) for seismic events to determine if any substantial safety improvements would reduce the SCDF and related SLERF of those sequences. All the dominant LERF sequences include either anchorage or functional failure of the ESTC cabinets, as discussed above. Seismically-induced failures of these cabinets will be mitigated by plant modifications to provide an alternate means of letdown isolation at each unit.

Based on the available information and engineering judgement, the NRC staff concluded that there were no further potential improvements to containment performance that would rise to the level of a substantial safety improvement or would warrant further regulatory analysis.

Additionally, the NRC staff reviewed the results of the Individual Plant Examination of External Events (IPEEE) and SAMA analyses previously completed for Oconee 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.

Conclusion

Based on the analysis of the submittal and supplemental information, the NRC staff concludes that no modifications, except for the plant modifications to provide an alternate means of letdown isolation in each unit identified by the licensee and included in a regulatory commitment, are warranted under 10 CFR Section 50.109 because:

- The staff did not identify a potential modification necessary for adequate protection or compliance with existing requirements;
- no additional potential cost-justified substantial safety improvement was identified based on the estimated achievable reduction in SCDF and/or SLERF; and
- additional consideration of containment performance, as described in NUREG/BR-0058 and assessed via SLERF, did not identify an additional modification that would result in a substantial safety improvement.

Table 1. Importance Analysis Results of Top Contributors to Seismic SCDF

Description	Failure Mode	RRW	MCR (/yr)
Seismically-failed SSCs			
Oconee 230KV Switchyard, Transformer Yard	Function	1.183	8.81E-06
Small LOCA	Structural/Anchorage	1.077	4.06E-06
RC PUMP	Structural/Anchorage	1.075	3.97E-06
Main Control Board	Structural/Anchorage	1.053	2.87E-06
Cable Trays - B103, B104, and B105 at Turbine Elevation 796 ft-6	Structural/Anchorage	1.041	2.22E-06
Unit 1 Reactor Coolant Loop Piping	Structural/Anchorage	1.040	2.17E-06
Relay Chatter - 1 SF-CA-AB822-EB_WESTINGHOUSE - CO-7 (1875253)	Relay/Function	1.038	2.10E-06
Relay Chatter - 1SF-SH-1T-SWGR_WESTINGHOUSE - AR (606B029A09)	Relay/Function	1.019	1.05E-06
600/208V XFMR 1XSF (30KVA/3PH/60HZ)	Function	1.018	9.80E-07
208V MCC 1XSFA	Function	1.018	9.80E-07
Randomly-failed SSCs			
Either PSV Fails to Close When Challenged by Liquid	Not Applicable	1.085	4.48E-06
RCP Seal LOCA Given CBO Not Isolated	Not Applicable	1.019	1.04E-06
Pressurizer PORV Fails to Reclose During a Transient	Not Applicable	1.014	7.70E-07
Either PSV Fails to Close During Steam Relief	Not Applicable	1.004	2.34E-07
Human Failure Events			
Operators Fail to Trip the RCPs In Time to Prevent RCP Seal Failure (Seismic-3)	Not Applicable	1.018	1.01E-06
Dependent HEP for TTRPRCPDHE-S3,UPSRCPSDHE-S3	Not Applicable	1.014	7.92E-07
Dependent HEP for NSF1ASWDHE-S3,RRCVENTDHE-S3	Not Applicable	1.011	6.33E-07
Dependent HEP for NSF1ASWDHE-S4,RRCVENTDHE-S4	Not Applicable	1.007	4.05E-07
Dependent HEP for FEFEFW2DHE-S3,RRCVENTDHE-S3	Not Applicable	1.007	3.82E-07
Dependent HEP for NSF1ASWDHES3,WHSEWSTDHE-S3,RRCVENTDHE-S3	Not Applicable	1.005	2.62E-07

Table 2. Importance Analysis Results of Top Contributors to Seismic LERF

Description	Failure Mode	RRW	MCR (/yr)
Seismically-failed SSCs			
FRAGILITY CORRELATION FACTOR: ESTC1, ESTC2, and ESTC3 cabinets - anchorage	Structural/Anchorage	2.125	1.48E-05
RC PUMP	Structural/Anchorage	1.365	7.48E-06
FRAGILITY CORRELATION FACTOR: ESTC1, ESTC2, and ESTC3 cabinets - function	Function	1.199	4.64E-06
Small LOCA	Structural/Anchorage	1.132	3.26E-06
Unit 1 Reactor Coolant Loop Piping	Structural/Anchorage	1.060	1.59E-06
Oconee 230KV Switchyard, Transformer Yard	Function	1.055	1.46E-06
Main Control Board	Structural/Anchorage	1.020	5.60E-07
Relay Chatter - 1 SF-SH-1TSWGR_WESTINGHOUSE -AR (6068029A09)	Relay	1.014	3.86E-07
125VDC I & C POWER PNL 1DID	Function	1.011	3.16E-07
Relay Chatter - 1 SF-CA-AB822EB_WESTINGHOUSE - CO-7 (1875253)	Relay	1.010	2.88E-07
Randomly-failed SSCs			
Either PSV Fails to Close When Challenged by Liquid	Not Applicable	1.043	1.15E-06
Conditional Probability of Early Containment Failure at Vessel Breach (Low)	Not Applicable	1.011	2.94E-07
RCP Seal LOCA Given CBO Not Isolated	Not Applicable	1.008	2.35E-07
One Week /yr factor used for in-service time of East Penetration Room LPSW pipe	Not Applicable	1.007	1.93E-07
Pressurizer PORV Fails to Reclose During a Transient	Not Applicable	1.006	1.57E-07
Either PSV Fails to Close During Steam Relief	Not Applicable	1.002	5.88E-08
Human Failure Events			
Dependent HEP for TTRPRCPDHE-S3, UPSRCPSDHE-S3	Not Applicable	1.009	2.52E-07
Operators Fail to Trip the RCPs In Time to Prevent RCP Seal Failure (Seismic-3)	Not Applicable	1.008	2.32E-07
Dependent HEP for NSF1ASWDHE-S3, WHSEWSTDHE-S3	Not Applicable	1.002	6.44E-08
Dependent HEP for TTRPRCPDHE-S3, WHSEWSTDHE-S3	Not Applicable	1.002	6.16E-08

AUDIT SUMMARY BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

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-2018-JLD-0173)

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). By letter dated July 11, 2017 (ADAMS Accession No. ML17192A168), the NRC staff confirmed that the audit process for the seismic hazard reevaluations applies to Oconee Nuclear Station, Units 1, 2, and 3 (Oconee).

REGULATORY AUDIT SCOPE AND METHODOLOGY

The areas of focus for the regulatory audit are the information contained in the SPRA submittal 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 submittal and to gain understanding of non-docketed information that supports the docketed SPRA submittal. The staff's clarification questions dated February 25, 2019, April 22, 2019, May 2, 2019, and

May 22, 2019 (ADAMS Accession Nos. ML19056A445, ML19113A168, ML19122A256, and ML19142A158 (non-public), respectively), were sent to the licensee to support the audit.

The licensee provided clarifying information in the following areas:

- Information describing the relationship between the Oconee Internal Events PRA and the Seismic PRA.
- Discussion of the use of representative fragilities for certain structures, systems and components (SSCs) in response to Fact and Observation (F&O) 23-3.
- Discussion of the technical basis and justification for the LERF evaluation.
- Discussion of the truncation level used to quantify seismic risk.
- Discussion of the inclusion of fire scenarios into the SPRA model.
- Discussion of relay chatter events within the SPRA model.
- Discussion of the lead unit concept and its applicability to the three Oconee units.
- Discussion of the mean values for SCDF and SLERF.
- Discussion of the use of Diverse and Flexible Coping Strategies (FLEX) equipment in the SPRA model.
- Discussion of structural fragility provided for the Jocassee Dam.
- Clarification of various portions of the SPRA submittal contained in Sections 5.4, 5.5, and 5.7

The licensee's response to the questions aided in the staff's understanding of the Oconee 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 Oconee's docketed SPRA submittal.

<u>DOCUMENTS AUDITED</u>

- PWROG-18011-P, "Peer Review of the Oconee Units 1, 2 & 3 Seismic Probabilistic Risk Assessment," Revision 0 (contained in OSC-11576, Revision 5, Appendix I)
- OSC -11788, "Oconee Seismic Probabilistic Risk Assessment Quantification Notebook," Revision 1
- OSC -11651, "Oconee Nuclear Plant Seismically-induced Flood and Fire Assessment," Revision 0
- OSC-11576, Appendix A, "Oconee Nuclear Station Internal Events at Power Probabilistic Risk Assessment (PRA) Peer Review," Revision 0

- OSC-11576, Appendix B, "AREVA Internal Events F&O Resolutions Review," Revision 0
- OSC-11576, Appendix C, "Original LERF Peer Review," Revision 0
- OSC -11783, "Oconee Nuclear Station (ONS) Seismic Human Reliability Analysis," Revision 1
- OSC -11782, "Oconee Seismic Probabilistic Risk Assessment Model Notebook," Revision 1
- OSC -11790, "Oconee Seismic Probabilistic Risk Assessment Uncertainty and Sensitivity Notebook," Revision 1
- "Oconee Seismic Probabilistic Risk Assessment 2018 Facts and Observation Notebook," Revision 0 (contained in OSC-11576, Revision 5, Appendix J)
- OSC-11670.01, Appendix J, "West Penetration Rooms (Units 1, 2 and 3)," Revision 0
- OSC-11671, "Fukushima NTTF 2.1 Seismic Vendor Support Documents for development of Fixed-Based Finite Element Models (FEM) for the Auxiliary Building (AB)," Revision 0
- OSC-11680, "Fukushima NTTF 2.1 Seismic Vendor Support Documents for Development of Seismic Response Spectra for the Auxiliary Building based on the Ground Motion Response Spectrum (GMRS)," Revision 0
- OSC-11681, "ONS Fukushima NTTF 2.1 Seismic Vendor Support Documents for Development of Seismic Response Spectra for the Reactor Building (RB) based on the Ground Motion Response Spectrum (GMRS)," Revision 0
- OSC-11684, "ONS Fukushima NTTF 2.1 Seismic Vendor Support Documents for Development of Seismic Response Spectra for the Turbine Building (TB) based on the Ground Motion Response Spectrum (GMRS)," Revision 0
- OSC-11685, "ONS Fukushima NTTF 2.1 Seismic Vendor Support Documents for Development of Seismic Response Spectra for the nit 1 & 2 Blockhouse (BH) based on the Ground Motion Response Spectrum (GMRS)," Revision 0
- OSC-11689, "ONS Fukushima NTTF 2.1 Seismic Vendor Support Documents for the Development of Representative Seismic Fragilities to Support the Oconee Nuclear Station (ONS) Seismic Probabilistic Risk Assessment (SPRA)," Revision 0
- OSC-11690, "Fukushima NTTF 2.1 Seismic Vendor Support Documents for the Development of Seismic fragilities for Essential Relays to Support the Oconee Nuclear Station (ONS) Seismic Probabilistic Risk Assessment (SPRA)," Revision 0
- OSC-11691.13, "Fukushima NTTF 2.1 Seismic Vendor Support Documents for the Main Control Board (1SF-BD-MCRBOARD) Fragility," Revision 0

- OSC-11691.28, "Fukushima NTTF 2.1 Seismic Vendor Support Documents for the SSLOCA Limit State Fragility," Revision 1
- OSC-11749, "Fukushima NTTF 2.1 Oconee Finite Element Model Notebook," Revision 0
- OSC-11750, "Fukushima NTTF 2.1 Oconee Instructure Response Analysis Notebook," Revision 0
- OSC-11751, "Fukushima NTTF 2.1 Oconee Fragility Notebook," Revision 0
- Memorandum from Tim Graf (Simpson, Gempertz & Heger) to Russell Childs (Duke), Subject: "Seismic Force Development for Reactor Coolant Pump Snubbers," dated March 26, 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 SPRA submittal, concludes the SPRA audit process for Oconee.

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 – 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-2018-JLD-0173) DATED NOVEMBER 29, 2019

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