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

September 24, 2020

Mr. James Barstow
Vice President, Nuclear Regulatory
Affairs and Support Services
Tennessee Valley Authority
Browns Ferry Nuclear Plant
1101 Market Street, LP 4A-C
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT, 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-2019-JLD-023)

Dear Mr. Barstow:

The purpose of this letter is to document the staff's evaluation of the Browns Ferry Nuclear Plant, Units 1, 2, and 3 (Browns Ferry, BFN), 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 Browns Ferry.

By letter dated March 12, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12053A340), the NRC issued a request for information under Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.54(f) (hereafter referred to as the 50.54(f) letter). The request was issued as part of implementing lessons learned from the accident at the Fukushima Dai-ichi nuclear power plant. Enclosure 1 to the 50.54(f) letter requested that licensees reevaluate seismic hazards at their sites using present-day methodologies and guidance. Enclosure 1, Item (8), of the 50.54(f) letter requested that certain licensees complete 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 17, 2019 (ADAMS Accession No. ML19351E391), Tennessee Valley Authority (TVA, the licensee), provided its SPRA submittal in response to Enclosure 1, Item (8) of the 50.54(f) letter, for Browns Ferry. 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 NRC staff's assessment of the Browns Ferry SPRA submittal is contained in Enclosure 1 to this letter. As described below, the NRC staff has concluded that the Browns Ferry 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. ML14098A478), TVA submitted the reevaluated seismic hazard information for Browns Ferry. The NRC performed a staff assessment of the submittal and issued a response letter on April 21, 2015 (ADAMS Accession No. ML15090A745). The NRC's assessment concluded that TVA 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 Browns Ferry.

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, Browns Ferry was expected to complete an SPRA with an estimated completion date of December 31, 2019, which would also assess high frequency ground motion effects. In addition, TVA was expected to perform a limited-scope evaluation for the Browns Ferry spent fuel pool (SFP). This SFP limited-scope evaluation was submitted by letter dated December 21, 2016 (ADAMS Accession No. ML16356A596). The staff provided its assessment of the Browns Ferry SFP evaluation by letter dated January 27, 2017 (ADAMS Accession No. ML17024A164).

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

In its letter dated December 17, 2019, TVA provided the SPRA submittal that initiated the NRC's Phase 2 decisionmaking process for Browns Ferry. The NRC described this Phase 2 decisionmaking process in a guidance memorandum from the Director of the Division of Operating Reactor Licensing to the Director of the Office of Nuclear Reactor Regulation (NRR) dated March 2, 2020 (ADAMS Accession No. ML20043D958). This memorandum describes a Senior Management Review Panel (SMRP) consisting of three 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 screening recommendations 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 Browns Ferry is described below. Based on its evaluation, the staff recommended to the SMRP that Browns Ferry 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 17, 2019, a technical team of NRC staff members performed a completeness review to determine if the necessary information to support Phase 2 decisionmaking had been included in the licensee's submittal. The technical team performing the review consisted of staff experts in the fields of seismic hazards, fragilities evaluations, and plant response/risk analysis. On February 19, 2020 (ADAMS Accession No. ML20052D788), the technical team determined that sufficient information was available to perform the detailed technical review in support of the Phase 2 decisionmaking.

As described in the 50.54(f) letter, the staff's detailed review focused on verifying the technical adequacy of the licensee's SPRA such that an appropriate level of confidence could be placed in the results and risk insights of the SPRA to support regulatory decisionmaking associated with the 50.54(f) letter. As stated in its submittal dated December 17, 2019, 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. In addition, the licensee also performed a close-out independent assessment of the resolution of the finding level facts and observations (F&Os) from the full-scope peer review following the process accepted by the NRC (ADAMS Accession No. ML17079A427). The close-out independent assessment resulted in the closure of all finding level F&Os for the Browns Ferry SPRA. Appendix A of the licensee's submittal provided a summary of the full-scope and close-out independent assessment peer reviews, including excerpts from the corresponding peer review reports.

By letter dated July 6, 2017 (ADAMS Accession No. ML17177A446), the NRC issued a generic audit plan and entered into the audit process described in Office Instruction LIC-111, "Regulatory Audits," dated December 29, 2008 (ADAMS Accession No. ML082900195), to assist in the timely and efficient closure of activities associated with the 50.54(f) letter. The list of applicable licensees in Enclosure 1 of the July 6, 2017, letter included TVA as the licensee for Browns Ferry. 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 May 13, 2020, (ADAMS Accession No. ML20254A303), 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 17, 2019, 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. The NRC staff confirmed that the licensee's dispositions to these findings were appropriately incorporated into the SPRA model and did not identify any modeling issues that could impact the conclusions of the SPRA 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 decisionmaking 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 Browns Ferry's submittal. As documented in the checklist, the staff concluded that the Browns Ferry 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 Browns Ferry 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 guide its review and screening recommendation 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. The Browns Ferry SPRA submittal demonstrated that the plant SCDF and SLERF for all three units were not below the initial screening values in the staff memorandum dated August 29, 2017. Based on the SCDF and SLERF results, the NRC staff utilized the Browns Ferry SPRA submittal and other available information in conjunction with the guidance in the staff memorandum dated August 29, 2017, to complete a detailed screening evaluation. The detailed screening concluded that Browns Ferry should be considered a Group 1 plant because:

- Sufficient reductions in SCDF and SLERF cannot be achieved by potential modifications considered in this evaluation, 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.

Based on the detailed screening evaluation and its review of the Browns Ferry SPRA submittal, the technical team determined that recommending Browns Ferry 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.

As a part of the Phase 2 decisionmaking process for SPRAs, the NRC formed the Technical Review Board (TRB), a board of senior-level NRC subject matter experts, to ensure consistency of review across the spectrum of plants that will be providing SPRA submittals. The technical review team provided the results of the Browns Ferry review to the TRB with the Phase 2 recommendation that Browns Ferry 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 Browns Ferry as a Group 1 plant.

Subsequently, the technical review team consulted with the SMRP and presented the results of the review including the recommendation for Browns Ferry 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 Browns Ferry 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 a licensee's SPRA submittal associated with their reevaluated seismic hazard information. The NRC staff's audit included a review of licensee documents through an electronic reading room. An audit summary document is included as Enclosure 3 to this letter.

CONCLUSION

Based on the staff's review of the Browns Ferry submittal against the endorsed SPID guidance, the NRC staff concludes that the licensee responded appropriately to Enclosure 1, Item (8) of the 50.54(f) letter. Additionally, the staff's review concluded that the SPRA is of sufficient technical adequacy to support Phase 2 regulatory decisionmaking in accordance with the intent of the 50.54(f) letter. Based on the results and risk insights of the SPRA submittal, the NRC staff also concludes that no further response or regulatory actions associated with NTTF Recommendation 2.1 “Seismic” are required.

Application of this review is limited to the review of the 10 CFR 50.54(f) response associated with NTTF Recommendation 2.1 “Seismic.” 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 Stephen Philpott at (301) 415-2365 or via e-mail at Stephen.Philpott@nrc.gov.

Sincerely,

/RA/

David J. Wrona, Acting Deputy Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-259, 50-260, and 50-296

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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SUBJECT: BROWNS FERRY NUCLEAR PLANT, 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-2019-JLD-0023) DATED SEPTEMBER 24, 2020

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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, 2018b, 2018c).

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

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

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

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

In general, the SPID allows 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, 2018b, 2018c), which supersedes the staff's November 12, 2012 (NRC, 2012), letter to NEI.

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

Tennessee Valley Authority (TVA, the licensee) updated its PSHA to incorporate updated seismicity rates using seismicity data collected since the publication of the Central and Eastern United States Seismic Source Characterization (CEUS-SSC). This update, combined with updates to the site geologic profiles used in the site response analysis resulted in minimal changes to the overall hazard at the site.

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

Consequence(s): N/A</td></tr>
<tr><td>The NRC staff concludes that:</td><td></td></tr>
<tr><td>• 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.</td><td>Yes</td></tr>
<tr><td>• although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.</td><td>N/A</td></tr>
<tr><td>• the guidance in the SPID was followed for developing the probabilistic seismic hazard for the site.</td><td>Yes</td></tr>
<tr><td>• an alternate approach was used and is acceptable on a justified basis.</td><td>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.	Yes
Notes from staff reviewer: The licensee updated its site response analysis to incorporate updated site geologic and geophysical information collected after the March 31, 2014 seismic hazard screening report (SHSR) submittal (ADAMS Accession No. ML14098A478). This updated site response, combined with PSHA updates, resulted in minimal changes to the site seismic hazard results and better represents the at-site geologic and geophysical conditions. 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 all supporting requirements (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.	No
B) If the SSE control point is not defined in the FSAR, one of three criteria in the SPID (Section 2.4.1) should be used.	Yes
C) An alternative method has been used for this site.	N/A
The control point used as input for the SPRA is identical to the control point used to establish the GMRS and previously accepted by the staff.	Yes
If yes, the control point can be used in the SPRA and the NRC staff's earlier acceptance governs.	
If no, the NRC staff's previous reviews might not apply. The staff's review of the control point used in the SPRA is acceptable.	N/A
Notes from staff reviewer: None Deviation(s) or deficiency(ies) and Resolution: 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 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: According to Section 4.3.3.2 of the submittal, new three-dimensional finite element models (FEMs) were developed for the Reactor Building, Intake Pump Structure, and the Diesel Generator Building. An existing lumped-mass stick model (LMSM) of the Nuclear Steam Supply System (NSSS) was updated to meet the seven SPID criteria and incorporated into the Reactor Building FEM. No finding-level facts and observations (F&Os) were developed for SFR-B3 by the SPRA peer review team, which relates to the use of mathematical models to represent the three-dimensional characteristics of the building structures for seismic response calculations. 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.	N/A
• Although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's 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-Base Dynamic Seismic Analysis of Structures for Sites Previously Defined as “Rock” (SPID Section 6.3.3)

Fixed-base dynamic seismic analysis of structures was used, for sites previously defined as “rock.” If no, this issue is moot.	No
If yes, on which structure(s)? Structure #1 name:	N/A
Structure #1: If used, is V_S > about 5,000 feet (ft.)/second (sec.)?	N/A
If 3,500 ft./sec. < V_S < 5000, was peak-broadening or peak shifting used?	N/A
Potential Staff Finding: The demonstration of the appropriateness of using this approach is adequate.	N/A
Notes from staff reviewer: According to Section 4.3.1 of the submittal, BFN is a firm rock site and soil-structure interaction (SSI) was performed for the fragility development for each of the major structures analyzed for the SPRA (i.e., Reactor Building, Diesel Generator Building, and Intake Pump Structure) that was used in the SPRA. Fixed-base analyses were only performed as a verification step in the development of these SSI models. 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.	N/A

• Although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis	N/A
• The licensee's use of fixed-based dynamic analysis of structures for a site previously defined as "rock" adequately meets the intent of the SPID guidance.	N/A
• 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)?	No
Structure #1: Scaling based on: Previously developed ISRS Shapes of previous uniform hazard spectrum/review-level earthquake (UHS/RLE) Shapes of new UHS/RLE Structural natural frequencies, mode shapes, participation factors	N/A
Potential Staff Findings: If a new UHS or RLE is used, the shape is approximately similar to the spectral shape previously used for ISRS generation.	N/A
If the shape is not similar, the justification for seismic response scaling is adequate.	N/A
Consideration of non-linear effects is adequate.	N/A
Notes from staff reviewer: During the audit the NRC staff reviewed the SPRA Full Scope Peer Review report wherein the peer reviewer concluded that SR SFR-B2 was not applicable to the BFN SPRA, thus, scaling of existing response analysis was not performed. Deviation(s) or deficiency(ies) and Resolution: None Consequence(s): N/A	

The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to the SR requirement SFR-B2 in the Code Case Standard, as well as to the requirements in the SPID.	N/A
• Although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's use of seismic response scaling adequately meets the intent of the SPID guidance.	N/A
• The licensee's use of seismic response scaling does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

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

The SPID does not provide specific guidance on performing new response analysis for use in developing ISRS and fragilities. The new response analysis is generally conducted when the criteria for use of existing models are not met or more realistic estimates are deemed necessary. The requirements for new analysis are included in the standard. See all of the SRs 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 SSI analysis (for soil sites), the analysis of how the seismic energy enters the base level of a given building, and the ISRS analysis. Said another way, an acceptable SPRA must use these analysis pieces together in a consistent way. The following are high-level key elements that should have been considered:	
1. Foundation Input Response Spectra (FIRS) site response developed with appropriate building specific soil velocity profiles. Structure #1 name: Reactor Building Structure #2 name: Diesel Generator Building Structure #3 name: Intake Pumping Station Structure #4 name: Yard Equipment (Ground Surface) Are all structures appropriately considered?	Yes Yes
2. Are models adequate to provide realistic structural loads and response spectra for use in the SPRA? 1. Is the SSI analysis capable of capturing uncertainties and realistic? 2. Is the probabilistic response analysis capable of providing the full distribution of the responses?	Yes Yes Yes

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

During the audit the NRC staff reviewed the SPRA Full Scope Peer Review report, wherein the peer reviewers concluded that the SSI modeling and structural models are realistic.

According to Table A-2 of the submittal, finding-level F&O 23-4, was developed against SR SFR-B5. This F&O identified a concern that while a single representative model was created for the Reactor Building SSI (soil column on the north side of the Reactor Building structure), the soil properties on the south side of the Reactor Building structure were different. To address the F&O, the licensee performed an assessment of potential differences in the soil properties on the north and south sides of the Reactor Building. The results show that the properties of the soils surrounding the reactor buildings are similar in shear wave velocity, and the differences are not large enough to significantly affect the computed SSI response of the buildings. Based on this assessment, the F&O was considered resolved and closed by the Independent Assessment Team (IAT). The staff review did not find any challenges to the IAT conclusion.

Section 3.2 of the submittal compares the GMRS and soil velocity information provided in the previous NTTF 2.1 submittal reviewed by the NRC staff with that used in the SPRA. Regarding the updated soil velocity information, the results of sensitivity analyses showed that the differences have an insignificant impact on the PSHA results. Regarding the GMRS, the comparison shows the overall shapes of the spectra to be similar with some small differences, with the most significant differences being at higher frequencies (greater than 20 Hertz) where the GMRS used in the SPRA is somewhat lower. The lower GMRS is explained as being due to 1) site-specific measurements that reduced the uncertainty in the response spectra and 2) slightly softer (slower velocity) subsurface conditions in the top 25 feet.

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

Consequence(s): N/A
</td></tr>
<tr><td>
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.

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

• The licensee’s FIRS modeling is consistent with the prior NRC review of the GMRS and soil velocity information.
</td><td>
Yes

N/A

Yes
</td></tr>
</table>

• 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.	N/A
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.	N/A
C) The approach for screening out certain SSCs from the model based on their inherent seismic ruggedness follows the recommendations in Section 6.4.3 of the SPID and has been appropriately justified.	Yes
D) The Standard has been followed.	N/A
E) An alternative method has been used and its use has been appropriately justified.	N/A

Notes from staff reviewer:

According to Section 4.4.2 of the submittal, the only SSCs screened out of the SPRA based on capacity were those determined to be inherently rugged.

According to Sections 4.4.1 and A.2.3.2 of the submittal, high capacity items were judged to have a high confidence of low probability of failure (HCLPF) capacity of at least 2g. Consistent with SPID guidance, a fragility curve with 2g HCLPF capacity was convolved with the hazard curve to yield a point estimate SCDF of 1.4E-8/rx-yr. This result was used to screen out SSCs from the need to develop more detailed fragilities. Consistent with the SPID guidance, these SSCs are retained in the SPRA with the screening level fragility assigned.

There were no finding-level F&Os identified related to SFR-C1, SFR-C2, and SPR-B5. According to the SPRA Full Scope Peer Review report reviewed by the NRC staff during

the audit, the peer review concluded that SR SPR-B5 was not applicable to the BFN seismic PRA, thus, screening of SSCs based on fragility analysis was not performed. Deviation(s) or deficiency(ies) and Resolution: None Consequence(s): N/A	
The NRC staff concludes:	
• The peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to the SR requirements SFR-C1, SFR-C2, and SPR-B5 in the Code Case Standard, as well as to the requirements in the SPID.	N/A
• Although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's use of 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 Conservation Deterministic Failure Margin (CDFM)/Hybrid method was used for seismic fragility analysis. If no, See item C) below and next issue. If yes: Potential Staff Findings:	Yes
A) The recommendations in Section 6.4.1 of the SPID were followed appropriately for developing the CDFM High Confidence Low Probability of Failure capacities.	Yes
B) The Hybrid methodology in Section 6.4.1 and Table 6-2 of the SPID was used appropriately for developing the full seismic fragility curves.	Yes
C) An alternative method has been used appropriately for developing full seismic fragility curves.	N/A

Notes from staff reviewer:

Section 4.4.2 of the SPRA submittal explains that fragilities for SSCs requiring fragility analysis were initially assigned site-specific representative fragility values developed from previous seismic analyses. After a first quantification of the SPRA, fragilities for top risk contributors were refined using the CDFM or Hybrid methodology. Following a second quantification of the SPRA, fragilities for the dominant risk contributors (SSCs with Fussell-Vesely (F-V) importance values greater than 0.005) were further refined using a combination of the CDFM and separation of variables (SOV) approaches. Further refinements to fragilities were made, using the CDFM and SOV approaches, in response to peer review finding-level F&Os. Overall, fragilities for 3 structures, 124 electrical and mechanical components, 29 block walls, and all relays were refined following the Hybrid method, and 11 electrical components, 23 block walls, and 20 relays were refined following the SOV approach. Representative fragilities were retained for the SSCs, block walls, and relays that were determined to be non-dominant risk contributors.

Tables 5.4-4, 5.4-5, and 5.4-6 of the submittal, for Units 1, 2, and 3, respectively, show that the majority of fragilities for the dominant risk contributors to SCDF were developed using the CDFM method, with several developed using the SOV approach and a couple that were assigned representative values. Similar results are reported in Tables 5.5-4, 5.5-5, and 5.5-6 for Units 1, 2, and 3 for the dominant SLERF contributors. See Topic

<table>
<tr><td colspan="2">#12 for the NRC staff evaluation of the licensee's fragility refinement process.

Section A.2.3.2 of the submittal states that the peer review team found the fragilities developed for the risk-significant SSCs to be acceptable.

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

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

• The peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to the requirements in the SPID. No requirements in the Code Case Standard specifically address this Topic.

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

• The licensee's use of the CDFM/Hybrid method for seismic fragility analysis meets the intent of the SPID guidance.

• The licensee's use of the CDFM/Hybrid method for seismic fragility analysis does not meet the intent of the SPID guidance, but is acceptable on another justified basis</td><td>Yes

N/A

Yes

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

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.	Yes
The flow chart in Figure 6-7 of the SPID was followed.	Yes
The flow chart was not followed but the analysis is acceptable on another justified basis.	N/A
Notes from staff reviewer: Section 4.1.2 of the submittal provides a brief description of the relay/breaker chatter evaluation and identifies the electrical components that are sensitive to high frequencies. The chatter-sensitive devices requiring fragility evaluation are identified in Table 4.1-2. The SPRA peer review concluded that the chatter evaluation was well performed and developed no finding-level F&Os against SR 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.	N/A
• Although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.	N/A

• The licensee's fragility analysis of SSCs sensitive to high frequency seismic motion meets the intent of the SPID guidance.	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) <u>Circuit analysis</u>: The seismic relay-chatter analysis of some relays relies on circuit analysis to assure that safety is maintained. (A) If <u>no</u>, then (B) is moot. (B) If <u>yes:</u> <u>Potential Staff Finding</u>: The approach to circuit analysis for maintaining safety after seismic relay chatter is acceptable.	Yes Yes
ii) <u>Operator actions</u>: The relay-chatter analysis of some relays relies on operator actions to assure that safety is maintained. (A) If <u>no</u>, then (B) is moot. (B) If <u>yes:</u> <u>Potential Staff Finding</u>: The approach to analyzing operator actions for maintaining safety after seismic relay chatter is acceptable.	Yes Yes
Notes from staff reviewer: During the audit the licensee explained that circuit analyses were performed for all SSCs on the seismic equipment list (SEL) for which chatter is possible and that the potential	

causes of chatter considered for fragility evaluation were 1) seismic shaking, 2) building to building impact, and 3) spatial interactions between cabinets containing chatter-sensitive devices. The chatter-sensitive devices were grouped into about 77 fragility groups based on make, model, and functions. Over 60 of these groups were evaluated using the Hybrid method, about 8 groups were evaluated using the SOV method, and the remainder were either evaluated using representative fragilities or were determined not to impact the corresponding function upon chatter. The SPRA Full Scope Peer Review report, reviewed by the NRC staff during the audit, noted that the SOV method was not performed for many of the top risk-significant relay groups. In response to this observation, a sensitivity study was performed that demonstrated that important risk insights are not masked by using the Hybrid method for those relay groups. During the audit the licensee explained that the SPRA credits operator actions to recover from two failures due to seismic-induced chatter by resetting the associated relays to restore power: 1) failure of diesel generators and 2) failure of the 4kV shutdown boards. All finding-level F&Os against HLR-SPR-D were assessed by the IAT to be resolved and closed. The staff review did not find any challenges to the IAT conclusions. Table A-2 of the submittal identifies that the SPRA peer review developed finding-level F&O 25-2 against SR SPR-B6, which identifies issues with the grouping of chatter-sensitive devices. The IAT concluded that this F&O was resolved in the SPRA and closed. The staff review did not find any challenges to the IAT conclusion. Table A-2 also identified F&O 25-6 against SR SPR-F1, which identifies issues with the documentation of the process accounting for all unscreened relays. The IAT concluded that this F&O was resolved in the SPRA and closed. The staff review did not find any challenges to the IAT conclusion. 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. • although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis. • the licensee's analysis of seismic relay-chatter effects meets the intent of the SPID guidance.	Yes N/A 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)

The CDFM methodology has been used in the SPRA for analysis of the bulk of the SSCs requiring seismic fragility analysis.	Yes
If no, 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.	N/A
If yes, 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."	Yes
Potential Staff Findings: 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.	Yes
B) The recommendations in Section 6.4.1 were not followed, but the analysis is acceptable on another justified basis.	N/A
Notes from staff reviewer: Section 4.4.2 of the SPRA submittal explains that fragilities for SSCs requiring fragility analysis were initially assigned site-specific representative fragility values developed from previous seismic analyses. After a first quantification of the SPRA, fragilities for top risk contributors were refined using the CDFM or Hybrid methodology. Following a second quantification of the SPRA, fragilities for the dominant risk contributors (SSCs with F-V values greater than 0.005) were further refined using a combination of the CDFM and SOV approaches. In response to peer review F&Os, further refinements to fragilities were made using the CDFM and SOV approaches. Overall, fragilities for 3 structures, 124 electrical and mechanical components, 29 block walls, and all relays were refined following the Hybrid method, and 11 electrical components, 23 block walls, and 20 relays were refined following the SOV approach. Representative fragilities are retained for the SSCs, block walls, and relays that are determined to be non-dominant risk contributors. Tables 5.4-4, 5.4-5, and 5.4-6, for Units 1, 2, and 3, respectively, show the majority of fragilities for the dominant risk contributors to SCDF were developed using the CDFM method, with several developed using the SOV approach and a couple that were assigned representative values. Similar results are reported in Tables 5.5-4, 5.5-5, and 5.5-6 for Units 1, 2, and 3 for the dominant SLERF contributors. Sensitivity Case #7	

<table>
<tr><td colspan="2">provided in Section 5.7 of the submittal addressed the sensitivity of the seismic risk results to four representative fragilities (SEIS_2-1-1, SEIS_5-8, SEIS_5-4, and SEIS_4-5) that show up in the lists of top risk contributors for both SCDF and SLERF for each of the three Units. Each of these representative fragilities was justified by the licensee during the audit as being conservative. The study showed that a significant increase in the fragilities for each (25-50% as determined during the audit) resulted in a relatively small decrease in SCDF and SLERF (less than 3% for SCDF and less than 15% for SLERF). Based on these results, the NRC staff concludes that use of these conservative fragilities does not change the decision for this submittal. Furthermore, because the risk importance of fragilities developed using CDFM are similar to or smaller than those for these representative fragilities (the fragilities for seismically-induced loss of offsite power (SEIS_LOOP) and failure of accident initiation signal relays and panels (SEIS_5-2B) are exceptions and are addressed in the Detailed Screening Evaluation), the NRC staff concludes that further refinement of the fragilities will not change the decision for this submittal.

The peer review team did not identify any finding-level F&Os against SR SFR-E3.

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

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

• the peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to SFR-E3 in the Code Case Standard and the requirements in the SPID.

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

• 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.

• 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.</td><td>

N/A

N/A

Yes

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

TOPIC 13: Evaluation of LERF (SPID Section 6.5.1)

<table>
<tr><td>The NRC staff review of the SPRA's analysis of LERF finds an acceptable demonstration of its adequacy.

<u>Potential Staff Findings</u>:
A) The analysis follows each of the elements of guidance for LERF analysis in Section 6.5.1 of the SPID, including in Table 6-3.

B) The LERF analysis does not follow the guidance in Table 6-3 but the analysis is acceptable on another justified basis.</td><td>Yes

Yes

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

Section 4.1.1 of the submittal describes the development of a seismic equipment list (SEL) for all three BFN units, including identification of SSCs associated with containment isolation and integrity. Section 5.1.5 further states that the SPRA LERF model is based on the internal events PRA (IEPRA) with seismic-induced failures incorporated. Lastly, Appendix A of the submittal explains that both the SPRA and the IEPRA were peer reviewed and all finding-level F&Os against SR HLR-SPR-E for the SPRA and all but one finding-level F&O (F&O 1-33) against LERF SRs of the IEPRA were closed using the NEI 12-13, Appendix X closure process accepted by the NRC. The NRC staff reviewed the licensee's disposition to F&O 1-33 against SR LE-F2 and concluded the finding does not impact the SPRA. Topic #14 provides the NRC staff's evaluation of the technical acceptability of the SPRA and IEPRA for supporting the staff's decision regarding this submittal.

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

Consequence(s): N/A</td></tr>
<tr><td>The NRC staff concludes that:
the peer review findings have been addressed and the analysis approach has been accepted by the 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.
although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.</td><td>Yes

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

• 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
<u>Potential Staff Findings</u>: 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. If <u>no</u>, go to (F). If <u>yes</u>, the "in process" peer review approach is acceptable. Go to (G).	N/A
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).	Yes

If no, go to (F). If yes, the "end-of-process" peer review approach is acceptable. Go to (G).	
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:

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 May 2019 against the capability category II (CC-II) SRs of the Code Case Standard (ASME/ANS RA-S Case 1, 2017), using the peer review process defined in NEI 12-13 (NEI, 2012). The Code Case Standard has been accepted by the NRC for use in regulatory applications, subject to NRC staff comments (NRC, 2018a). During the audit the licensee confirmed that these NRC staff comments were considered during the SPRA peer review. Guidance document NEI 12-13 has been accepted by the NRC, subject to certain NRC staff comments (NRC, 2018b, 2018c). The submittal confirmed that these NRC staff comments were considered during the SPRA peer review.

The SPRA submittal provides the qualifications for each of the peer review team members and states that the peer reviewers were independent of the BFN SPRA development. Concurrence on the assignment of capability categories to each SR was based on a consensus process involving all members of the review team. The two members focusing on review of the fragility analysis, one of which participated in the plant walkdown, were stated to have completed the SQUG training course or equivalent. The resumes for each of these peer reviewers, which were reviewed by the NRC staff during the audit, were shown to demonstrate significant PRA experience, and that the fragility reviewer walkdown participant is SQUG certified.

All elements of the SPRA were peer reviewed against the CC-II requirements. The submittal states that all finding-level F&Os have been closed using the NEI 12-13, Appendix X closure process accepted by the NRC, and that all SRs, with the exception of those determined to not be applicable to BFN by the peer review team, have been determined to meet the CC-II requirements. The licensee confirmed that the NRC's accepted process for closure of F&Os (NRC, 2017a, 2017b) was used, which included a self-assessment by the licensee as to whether each F&O disposition was a PRA maintenance or upgrade, and an assessment by the F&O closure team of concurrence or disagreement with this determination. The closure team assessment concluded that all finding-level F&O dispositions were PRA maintenance and that none incorporated use of a new methodology.

Section 5.1 of the submittal states the IEPRA model-of-record as of February 2018 was used as the basis for the development of the SPRA model. Section A.7 of the submittal states the IEPRA (excluding internal flooding) was peer reviewed in May 2009 against the CC-II requirements of the PRA standard (ASME/ANS Addendum A, 2009) and Regulatory Guide (RG) 1.200, Revision 2, utilizing the peer review process in NEI 05-04 (NEI, 2008). This peer review developed 95 finding-level F&Os. The licensee's dispositions to these F&Os were subjected to a focused-scope peer review in July 2015, which concluded that 47 of the F&Os were resolved and closed, while 48 of the F&Os remained open. During the audit the licensee confirmed that this focused-scope peer review was performed in accordance with the CC-II requirements of the PRA standard (ASME/ANS Addendum A, 2009) and Regulatory Guide (RG) 1.200, Revision 2, utilizing the peer review process in NEI 05-04 (NEI, 2008). The submittal states that all but 10 finding-level F&Os have been closed using an NRC-accepted process, that 8 SRs remain Not Met, and that 1 SR remains CC-I. During the audit the licensee confirmed that the NRC's accepted process for closure of F&Os (NRC, 2017a, 2017b) was used, which included a self-assessment by the licensee as to whether each finding-level F&O disposition was a PRA maintenance or upgrade, and an assessment by the F&O closure team of concurrence with this determination. The closure team assessment concluded that all closed finding-level F&Os were PRA maintenance and that none incorporated use of a new methodology. The NRC staff reviewed the licensee's dispositions of the 10 finding-level F&Os in the IEPRA that were kept open by the closure team, which are provided in Table A-6 of the submittal.

With regard to F&O 4-25, the IEPRA F&O closure team identified two issues with the disposition of this F&O: 1) the human failure events (HFEs) for which timing information was clarified are not those identified in the F&O, nor is there any discussion of the basis for the HFEs selected for clarification of timing, and 2) certain screened HFEs are assumed to have a delay time of 24 hours, which is inconsistent with some of the event descriptions (some screened events list times of 15 minutes or less). The licensee clarified during the audit that all of the HFEs that were the subject of this F&O were included in the SPRA and assumed to fail (i.e., HFEs set to 1.0). Correspondingly, all SPRA cutsets that have multiple screened HFEs are assigned a joint human error probability (HEP) of 1.0. The licensee further explained that even with these conservative assumptions none of the HFEs were determined to be significant to the SPRA results.

With regard to F&O 6-30, the IEPRA F&O closure team 1) noted that the dependency analyses were completely redone, 2) could not determine how the dependency analysis was performed, 3) noted that there were discrepancies in how HFEs with screening HEPs were treated, and 4) could not determine how dependent HFEs were identified for the sensitivity analysis. The IAT concluded that there is a potential that the revised dependency analysis may constitute an upgrade. The licensee explained during the audit that the HFE dependency analysis performed for the IEPRA was completely redone in the SPRA, that the dependency analysis performed for the SPRA was peer reviewed, and that all finding-level F&Os from the SPRA peer review were subsequently closed by the SPRA F&O closure team with none of the resolutions identified as an upgrade. Based on this, the licensee concluded that the disposition to F&O 6-30 would not impact the SPRA.

Based on the licensee's dispositions, as clarified during the audit, the NRC staff finds that the resolutions to the open IEPRA finding-level F&Os are unlikely to impact the

staff's decision for this SPRA submittal. The licensee used NRC-accepted processes for performing the peer reviews and F&O closure. Further, the licensee's dispositions to the open finding-level IEPRA F&Os are unlikely to impact the staff's decision for this SPRA submittal. Therefore, the NRC staff concluded that the licensee's IEPRA is of sufficient technical acceptability to form the base for the development of the SPRA documented in this submittal. Because the licensee peer-reviewed its SPRA using an accepted Code Case and peer-review guidance, an NRC-accepted process was used to close the SPRA F&Os, and that the underlying IEPRA is of sufficient technical acceptability to form the base for the development of the SPRA, the NRC staff concludes that the licensee's SPRA is of sufficient technical adequacy for its decision on this SPRA submittal. Deviation(s) or deficiency(ies) and Resolution: None Consequence(s): N/A	
The NRC staff concludes:	
• the licensee's peer-review process meets the intent of the SPID guidance as supplemented by NRC staff comments in the NRC letter dated March 7, 2018 (NRC 2018a, 2018b).	Yes
• 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.	N/A

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.	Yes
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

Notes from staff reviewer:

Tables 2.0-1 and 2.0-2 of the SPRA report provide a cross-reference of information requested by 10 CFR 50.54(f) and specified in Section 6.8 of the SPID to the sections of the submittal where the information can be found. The level-of-detail of the information provided is generally consistent with that specified in Section 6.8 of the SPID. The SPID requires that there should be sufficient information to assess the results of all key aspects of the analysis. Sections 5.3.2, 5.6, and A.8 of the submittal identify and discuss key assumptions and sources of uncertainty for the SPRA, with sensitivity analyses on some of these parameters provided in Section 5.7. Sections 5.4 and 5.5 of the submittal provide the SPRA results.

Section 5.6 of the submittal presents the SPRA quantification uncertainty results for both SCDF and SLERF for each of the 3 Units (i.e., the 5th percentile, mean, median (i.e., 50th percentile), and the 95th percentile). These mean and 95th percentile values were used in the NRC staff's screening evaluation reported in Enclosure 2 of this document.

According to Section 5.4.5.3, Table A-2, and Appendix B of the SPRA submittal, Diverse and Flexible Coping Strategies (FLEX) are credited in the SPRA for operator actions to align backup nitrogen to drywell control air for the safety relief valves (SRVs), which includes credit for portable nitrogen bottles that can be recharged and replaced. The associated storage racks and associated tubing, valves and instrumentation are permanently installed equipment and are included in the SPRA. No other FLEX systems or actions were credited. The NRC memorandum dated May 30, 2017, "Assessment of the Nuclear Energy Institute 16-06, 'Crediting Mitigating Strategies in Risk-Informed Decision Making,' Guidance for Risk-Informed Changes to Plants Licensing Basis" (ADAMS Accession No. ML17031A269), provides the NRC staff's assessment of the challenges of incorporating FLEX coping strategies and equipment into a PRA model in support of risk-informed decisionmaking in accordance with the guidance of RG 1.200, Revision 2 (ADAMS Accession No. ML090410014). Section 5.7 of the SPRA submittal indicates that credit for FLEX is significant to the SCDF results (removal of the credit increases SCDF by over 40%). The NRC staff evaluated the impact of not crediting FLEX on the screening evaluation reported in Enclosure 2 of this document and determined this FLEX credit does not impact the staff's decision on this submittal. Because FLEX equipment and actions would not change the staff's decision for this submittal, the licensee's treatment of FLEX was not pursued by the NRC staff for its decision on this submittal.

Appendix A of the submittal explains that the SPRA was peer reviewed and all finding-level F&Os were closed using an NRC-accepted process. Topic #14 provides the NRC

staff's evaluation of the technical acceptability of the SPRA to support decisionmaking on this submittal. Deviation(s) or deficiency(ies) and Resolution: None Consequence(s): N/A	
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	No N/A N/A N/A
Plant will: • complete modifications by: • report completion of modifications by:	N/A N/A
Notes from the Reviewer: 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	N/A N/A

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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

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

NRC, 2018c: "US Nuclear Regulatory Commission Acceptance of Nuclear Energy Institute (NEI) Guidance NEI 12-13, "External Hazards PRA Peer Review Process Guidelines" (August 2012)," tabular compilation of NRC staff comments, appended to (NRC, 2018b) (ADAMS Accession No. ML18025C022)

NRC Staff SPRA Submittal Detailed Screening Evaluation

Introduction

The Browns Ferry Nuclear Plant, Units 1, 2, and 3 (BFN) Seismic Probabilistic Risk Assessment (SPRA) report (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19351E391) provides the mean seismic core damage frequency (SCDF) as 1.51E-05/reactor-year (/rx-yr), 1.57E-05/rx-yr, and 1.72E-05/rx-yr for Units 1, 2, and 3, respectively, and the mean seismic large early release frequency (SLERF) as 6.74E-06/rx-yr, 7.27E-06/rx-yr, and 8.04E-06/rx-yr for Units 1, 2, and 3, respectively. The NRC staff compared these mean values against the guidance in NRC staff memorandum dated August 29, 2017 (ADAMS Accession No. ML17146A200), titled, "Guidance for Determination of Appropriate Regulatory Action Based on Seismic Probabilistic Risk Assessment Submittals in Response to Near Term Task Force Recommendation 2.1: Seismic" (hereafter referred to as the SPRA Screening Guidance), which establishes a process the NRC staff uses to develop a recommendation on whether the plant should move forward as a Group 1, 2, or 3 plant.[1]

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

Because the mean SCDF and SLERF for each of the BFN units were above the initial screening values of 1.0E-5/rx-yr and 1.0E-6/rx-yr, respectively, the NRC staff performed a detailed screening following the SPRA Screening Guidance. The detailed screening shows that BFN should be considered a Group 1 plant because:

- Sufficient reductions in SCDF and SLERF cannot be achieved by potential modifications considered in this evaluation to constitute substantial safety improvements based upon importance measures, available information, and engineering judgement;

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[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).

- 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

Tennessee Valley Authority (TVA, the licensee for BFN), in performing its seismic analysis in response to the Near-Term Task Force Recommendation 2.1, and the NRC staff 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 BFN license, or the rules and orders of the Commission. For these reasons, the licensee and the staff did not identify any potential modifications necessary for adequate protection or compliance with existing requirements.

The detailed screening uses information provided in the BFN SPRA report, particularly the importance measures, SCDF, SLERF, as well as other information described below, to establish threshold and target values used to identify potential cost-justified substantial safety improvements. The detailed screening process makes several simplifying assumptions, similar to a Phase 1 SAMA analysis (NEI 05-01, ADAMS Accession No. ML060530203) used for license renewal applications. The detailed screening process uses risk importance values as defined in NUREG/CR-3385, "Measures of Risk Importance and Their Applications" (ADAMS Accession No. ML071690031). The NUREG/CR-3385 states that the risk reduction worth (RRW) importance value is useful for prioritizing feature improvements that can most reduce the risk. The BFN 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 License Renewal Application, Browns Ferry Nuclear Plant, dated December 2003 (ADAMS Accession No. ML040060361), and associated supplements were used to calculate the RRW threshold. For this analysis, the NRC staff determined the RRW threshold from the SCDF-based MACR to be 1.326 for Unit 1, 1.287 for Unit 2, and 1.217 for Unit 3. The MACR calculation includes estimation of offsite exposures and offsite property damage, which captures the impact of SLERF. Therefore, separate SLERF-based MACR calculations were not performed. The target RRW values based on the mean and 95th percentile SCDF and SLERF were also calculated by the NRC staff to be between 1.09 and 2.95 for Unit 1, 1.09 and 2.75 for Unit 2, and 1.08 and 2.40 for Unit 3.

Section 5 of the BFN SPRA report includes tables listing and describing the structures, systems, and components (SSCs) that are the most significant failure contributors to SCDF and SLERF for each BFN unit. Similar tables were also provided for the most significant contributors to SCDF and SLERF due to failure of operator actions. The descriptions of the significant contributors included the F-V for each. The NRC staff utilized the F-V values to calculate the RRW and the contribution to SCDF and SLERF of each contributor. The results are provided in Tables 1 and 2 for Unit 1, Tables 3 and 4 for Unit 2, and Tables 5 and 6 for Unit 3. The listed seismic-induced failures and human failure events that contribute to SCDF and SLERF have an RRW greater than 1.005. These tables provide the following information by column: (1) Description of the component, (2) Failure Mode, (3) RRW, and, (4) maximum SCDF reduction (MCR) or SLERF reduction (MLR) from eliminating the failure. The same single SPRA model element or contributor exceeded the mean target RRW for SCDF for each of the three units and the same three SPRA model elements or contributors exceeded the mean target RRW for SLERF for each of the three units.

The NRC staff considered both single and combinations of basic events in accordance with the SPRA Screening Guidance. It is not the intent of that aspect of the guidance to aggregate several disparate basic events that individually have RRW values close to the mean target RRW. A review of these model elements in Tables 1 through 6 of this enclosure reveals that most modifications or sets of modifications to achieve a SCDF reduction of at least 1.0E-05/rx-yr or a SLERF reduction of at least 1.0E-06/rx-yr will have to mitigate or prevent multiple failure types (e.g., seismically-induced failures, random failures, 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).

The highest contributor to SCDF and SLERF for all three units was seismically-induced loss of offsite power (SEIS_LOOP), which exceeded the mean target RRW for both SCDF and SLERF. The submittal did not provide tables of the top SCDF and SLERF cutsets. During the audit, the NRC staff reviewed tables containing the top 100 SCDF and SLERF cutsets for each unit. According to these tables, this basic event is a contributor to 47 of the top 50 SCDF and SLERF cutsets for each unit. Section 5.1.3 of the submittal explained that SEIS_LOOP represents seismic-induced loss of offsite power from both the plant switchyard and from the offsite power grid and that the fragility used in the SPRA is a single representative fragility representing both. Consideration of improvements to the offsite power grid is beyond the scope of this screening evaluation. The dominant sequences with LOOP that contribute to the SCDF and SLERF involve failure of mitigation SSCs due to relay chatter, failure of buried piping, and structural failures. The SPRA models human actions for recovery from relay chatter. The potential risk reduction achievable from eliminating failures of SSCs mitigating the LOOP sequences (random and seismic) and related human actions is insufficient to provide substantial safety enhancement. Therefore, while the mitigation systems were reviewed, potential modifications to those systems would not be substantial safety enhancements and therefore, were not pursued by the NRC staff. Because this event involves seismic-induced failures outside of the plant boundary, and because the risk reduction from potential plant improvements to mitigate the consequences of LOOP events are below the threshold for consideration of substantial safety enhancements, the NRC staff did not pursue potential improvements related to SEIS_LOOP.

Excluding the contribution from SEIS_LOOP, the total contribution to SCDF from the remainder of the seismically-failed elements listed in Tables 1, 2, and 3 for Units 1, 2, and 3, respectively, is less than 1.0E-05/rx-yr. The NRC staff therefore did not pursue potential improvements to these elements to reduce SCDF.

The second highest contributor to SLERF for all three units is seismically-induced failure of accident initiation signal relays and panels (SEIS_5-2B), which fails standby coolant injection, low-pressure core injection (LPCI), core spray, and shutdown cooling. According to information provided during the audit, this basic event is a contributor to over 30 of the top 50 SLERF cutsets for each unit. Potential modifications to substantially increase the fragility of the SEIS_5-2B panels was evaluated in Appendix B of the submittal. Two modifications would be required because the controlling failure mode is failure of the panel anchorage followed closely by failure of the panel welds. The licensee determined that modifications to improve the panel fragility substantially were impractical due to interferences with other equipment, accessibility issues, the number of panels in the room, and that the only potential support point for bracing are block walls that are seismically vulnerable. Furthermore, the NRC staff experience from SAMA analyses for modifications that require moving or relocating safety-related SSCs, in this case multiple electrical panels, is that the implementation cost would likely exceed the calculated MACR for this detailed screening. The NRC staff therefore did not pursue further potential improvements to SEIS_5-2B.

The third highest contributor to SLERF for all three units is seismically-induced failure of the unit batteries (SEIS_2-1-1), which exceeded the mean target RRW for SLERF for all three units. According to information provided during the audit, this basic event is a contributor to at least 6 of the top 50 SLERF cutsets for each unit. This seismic-induced failure is modeled using a representative fragility. Section 5.7 of the submittal provides the results of a sensitivity study, using the Unit 1 SPRA, of the four top SLERF contributors that were modeled using representative fragilities, including SEIS_2-1-1. During the audit the licensee explained that the representative fragility for SEIS_2-1-1 is conservative because the anchorage structural evaluation used BFN design documentation that includes design margin of at least 40 percent, which is how much the capacity of the unit batteries was increased in the sensitivity study. This study showed that the SLERF would be reduced by about 14 percent if the conservatisms in the fragilities for the four contributors were reduced. Furthermore, due to there being overlapping failures in the cutsets associated with SEIS_2-1-1 (for example, SEIS_LOOP is in every one of the associated top 100 cutsets), a plant modification to increase the fragility of SEIS_2-1-1 would reduce the SLERF significantly less than the MLR shown in Table 2. As a result, the NRC staff concluded that even if seismically-induced failure of the unit batteries was completely eliminated, the reduction in SLERF would be significantly less than 1.0E-6/rx-yr. The same conclusion is made for the other units. The NRC staff therefore did not pursue further potential improvements to SEIS_2-1-1.

Excluding the contributions from SEIS_LOOP, SEIS_5-2B, and SEIS_2-1-1, the total contribution to SLERF from the remainder of the seismically-failed elements listed in Table 2 is 2.3E-06/rx-yr. Given that Unit 1 plant modifications would need to address at least three of these elements to achieve a risk reduction of at least 1.0E-06/yr (or at least three of the remainder of the seismically-failed elements listed in Tables 4 and 6 for Units 2 and 3, respectively), that no single modification was identified to address at least three of these elements, and that the implementation cost to address multiple elements is expected to exceed the monetary value of any beneficial improvements, the NRC staff did not pursue potential improvements to these elements.

Based on the analysis described above, the NRC staff concludes that no modifications are warranted in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.109 to reduce SLERF because a potential cost-justified substantial safety improvement was not identified.

In accordance with Section 3.3.2 of NUREG/BR-0058, Revision 4, the NRC staff further evaluated BFN accident sequences and cutsets impacting the conditional probability of early containment failure or bypass (CPCFB) for seismic events to determine if any substantial safety improvements would reduce the SLERF of those sequences. All the dominant failures are already evaluated, as described above.

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 considered insights from the individual plant examination of external events (IPEEE) and SAMA analyses previously completed for BFN to understand previous work done to identify substantial safety improvements and to further inform this review. Based on previous evaluations and based on the detailed screening completed as part of this review, no potential improvements were found.

Conclusion

Based on the analysis of the submittal and supplemental information, the NRC staff concludes that no modifications 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 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 a modification that would result in a substantial safety improvement.

Table 1. Importance Analysis Results of Top Contributors to Unit 1 Seismic CDF

Description	Failure Mode	RRW	MCR (/yr)
Seismically-failed SSCs			
LOOP (Loss of Offsite Power)	Ceramic Insulators	8.696	1.34E-05
Residual Heat Removal Service Water (RHRSW) pumps based on pipe fragility (Pipe Calculation)	Soil (Buried Piping) Failure	1.059	8.47E-07
Emergency Equipment Cooling Water (ECCW) pumps based on pipe fragility calculation	Soil (Buried Piping) Failure	1.056	8.02E-07
Unit Batteries (Fragility Group 15-03)	Anchorage	1.049	7.11E-07
480v Board (BD) 219 A, B (U1, U2) (Fragility Group 01-05-01) Anchorage Failure	Anchorage	1.034	4.99E-07
3EA,3EC Shutdown Board (SDBD) Common Accident Signal Relays (CASA)	Functionality	1.022	3.33E-07
480v BD 219 A, B (U1, U2) (Fragility Group 01-05-01) Functional Failure	Functionality	1.022	3.33E-07
RHRSW pumps and EECW Alternate (Fragility Group 06-03)	Anchorage	1.017	2.57E-07
Initiation relays and panels	Anchorage	1.016	2.42E-07
Battery CH 248-1 (Fragility Group 16-06)	Functionality	1.015	2.27E-07
Relay Group 3 for group SEIS_11-1 (EECW Pp A3 UV device)	Functionality	1.013	1.97E-07
Relay Group 1-1 for group SEIS_11-1 (EECW Pp B3 UV device)	Functionality	1.011	1.66E-07
Intake Pumping Station	Structural Analysis	1.010	1.51E-07
Wall 28 falls towards 480v SD BD 1A and 1B	Block Wall Failure	1.010	1.51E-07
4160-480V Transformer, Control Bay (Fragility Group 4-12)	Functionality	1.008	1.21E-07
Battery and Wall 8 or 47 interface (Block Wall Group 5)	Block Wall Failure	1.007	1.06E-07
Relay Group 2 for group SEIS_11-1 (EECW Pp B3&D3 OC device)	Functionality	1.007	1.06E-07

Description	Failure Mode	RRW	MCR (/yr)
480V BD 1A, 2A, 3A, 1B, 2B, 3B (Fragility Group 02-01)	Functionality	1.007	1.06E-07
B,D SDBD 480 Transformer Trip relays (50G)	Functionality	1.007	1.06E-07
Relay Group 4 for Group SEIS_11-1 (EECW Pp A3 OC device)	Functionality	1.005	7.57E-08
3EA,3EC SDBD Lockout relays (86)	Functionality	1.005	7.57E-08
3EA,3EC SDBD Lockout relays (86)	Functionality	1.005	7.57E-08
Seismic failure of Main Control Room instrumentation	Functionality	1.005	7.57E-08
Human Failure Events			
Operator fails to align Nitrogen (N2) FLEX backup to Drywell Control Air (SRVs)	Not Applicable	1.261	3.13E-06
Operator fails to align Containment Air Dilution backup to Drywell Control Air	Not Applicable	1.222	2.75E-06
Failure to align Residual Heat Removal (RHR) for suppression pool cooling (Anticipated Transient Without Scram or ATWS) or Inadvertent Opening of One Relief Valve (IOORV))	Not Applicable	1.121	1.63E-06
Failure to align RHR for suppression pool cooling (non-ATWS/IOORV)	Not Applicable	1.071	9.97E-07
Operator fails to align backup EECW pump	Not Applicable	1.066	9.43E-07
Operator reset of 4kV Shutdown Board lockout relays (seismic)	Not Applicable	1.057	8.17E-07
Operator fails to isolate SD BD and align alternate DC	Not Applicable	1.057	8.09E-07
Failure to align alternate battery charger	Not Applicable	1.032	4.66E-07
Operator fails to align alternate feeder	Not Applicable	1.029	4.27E-07
Failure to open doors and install fans after Heating, Ventilation, and Air Conditioner (HVAC) failure	Not Applicable	1.022	3.19E-07
Failure to start standby Control Bay HVAC	Not Applicable	1.019	2.77E-07
Failure to transfer 480V shutdown board to alternate source	Not Applicable	1.015	2.25E-07

Description	Failure Mode	RRW	MCR (/yr)
Failure to transfer deenergized 480v board to alternate supply	Not Applicable	1.011	1.71E-07
Failure to initiate reactor-vessel depressurization (transient or ATWS)	Not Applicable	1.009	1.35E-07
Operator fails to manually control level with High Pressure Coolant Injection	Not Applicable	1.008	1.26E-07
Seismic failure of Main Control Room instrumentation	Not Available	1.005	7.72E-08

Table 2. Importance Analysis Results of Top Contributors to Unit 1 Seismic LERF

Description	Failure Mode	RRW	MLR (/yr)
Seismically-failed SSCs			
LOOP (Loss of Offsite Power)	Ceramic Insulators	5.556	5.53E-06
Initiation relays and panels	Anchorage	1.266	1.42E-06
Unit Batteries (Fragility Group 15-03)	Anchorage	1.224	1.23E-06
Reactor Protection & Nuclear Steam Supply (NSS) Panel (PNL) (18-02)	Functionality	1.059	3.77E-07
Emergency Equipment Cooling Water pumps based on pipe fragility calculation	Soil (Buried Piping) Failure	1.044	2.83E-07
Battery and Wall 8 or 47 interface (Block Wall Group 5)	Block Wall Failure	1.036	2.36E-07
480v BD 219 A, B (U1, U2) (Fragility Group 01-05-01) Anchorage Failure	Anchorage	1.035	2.29E-07
Intake Pumping Station	Structural Analysis	1.021	1.42E-07
480v BD 219 A, B (U1, U2) (Fragility Group 01-05-01) Functional Failure	Functionality	1.021	1.42E-07
RHRSW pumps and EECW Alternate (Fragility Group 06-03)	Anchorage	1.016	1.08E-07
480V BD 1A, 2A, 3A, 1B, 2B, 3B (Fragility Group 02-01)	Functionality	1.015	1.01E-07
RHRSW pumps based on pipe fragility (Pipe Calculation)	Soil (Buried Piping) Failure	1.012	8.09E-08
Relay Group 1-1 for group SEIS_11-1 (EECW Pp B3 UV device)	Functionality	1.011	7.42E-08
Relays for Group SEIS_14-1, Relay Group 1 (High Pressure Core Injection (HPCI)/Reactor Core Isolation Cooling (RCIC) Isolations)	Functionality	1.011	7.42E-08
Relay Group 3 for Group SEIS_11-1 (EECW Pp A3 UV device)	Functionality	1.011	7.42E-08
Wall 28 falls towards 480v SD BD 1A and 1B	Block Wall Failure	1.011	7.42E-08
Panel Group 4-1 Control Room Lower Fragility (Fragility Group 20-02 and 20-03)	Anchorage	1.010	6.74E-08

Description	Failure Mode	RRW	MLR (/yr)
Relay Group 2 for Group SEIS_11-1 (EECW Pp B3&D3 OC device)	Functionality	1.009	6.07E-08
3EA,3EC SDBD Common Accident Signal Relays (CASA)	Functionality	1.006	4.04E-08
Initiation relays and panels (20-08)	Anchorage	1.006	4.04E-08
Wall 28 falls toward 480v reactor motor-operated valve (RMOV) 1A and 4kv SD BD A	Block Wall Failure	1.006	4.04E-08
EECW pumps (Fragility Group 06-03-01)	Anchorage	1.005	3.37E-08
Human Failure Events			
Operator fails to align alternate feeder	Not Applicable	1.036	2.37E-07
Failure to initiate Reactor Pressure Vessel (RPV) depressurization (transient or ATWS)	Not Applicable	1.036	2.32E-07
Operator fails to manually control level with High Pressure Coolant Injection	Not Applicable	1.035	2.29E-07
Operator Fails to Initiate Depressurization (SLERF)	Not Applicable	1.027	1.80E-07
Operator fails to manually initiate injection into drywell after core damage	Not Applicable	1.027	1.80E-07
Operator reset of 4kV Shutdown Board lockout relays (seismic)	Not Applicable	1.009	6.27E-08
Operator fails to manually initiate injection for in-vessel recovery	Not Applicable	1.008	5.33E-08
Operator fails to manually initiate injection into drywell after core damage	Not Applicable	1.005	3.44E-08

Table 3. Importance Analysis Results of Top Contributors to Unit 2 Seismic CDF

Description	**Failure Mode**	**RRW**	**MCR (/yr)**
Seismically-failed SSCs			
LOOP (Loss of Offsite Power)	Ceramic Insulators	8.696	1.39E-05
RHRSW pumps based on pipe fragility (Pipe Calculation)	Soil (Buried Piping) Failure	1.059	8.81E-07
Emergency Equipment Cooling Water pumps based on pipe fragility calculation	Soil (Buried Piping) Failure	1.056	8.34E-07
Unit Batteries (Fragility Group 15-03)	Anchorage	1.049	7.39E-07
480v BD 219 A, B (U1, U2) (Fragility Group 01-05-01) Anchorage Failure	Anchorage	1.034	5.19E-07
480v BD 219 A, B (U1, U2) (Fragility Group 01-05-01) Functional Failure	Functionality	1.022	3.46E-07
3EA,3EC SDBD Common Accident Signal Relays (CASA)	Functionality	1.019	2.99E-07
RHRSW pumps and EECW Alternate (Fragility Group 06-03)	Anchorage	1.017	2.67E-07
Battery CH 248-1 (Fragility Group 16-06)	Functionality	1.015	2.36E-07
Relay Group 3 for Group SEIS_11-1 (EECW Pp A3 UV device)	Functionality	1.015	2.36E-07
250v DC Bus A interface with wall 64 (Block Wall Group 3)	Block Wall Failure	1.013	2.04E-07
Intake Pumping Station	Structural Analysis	1.010	1.57E-07
Wall 63 falls towards 480v SD BD 2A and 2B	Block Wall Failure	1.010	1.57E-07
Relay Group 1-1 for Group SEIS_11-1 (EECW Pp B3 UV device)	Functionality	1.009	1.42E-07
Initiation relays and panels	Anchorage	1.009	1.42E-07
Battery and Wall 8 or 47 interface (Block Wall Group 5)	Block Wall Failure	1.007	1.10E-07
480V BD 1A, 2A, 3A, 1B, 2B, 3B (Fragility Group 02-01)	Functionality	1.007	1.10E-07

Description	Failure Mode	RRW	MCR (/yr)
Relay Group 2 for Group SEIS_11-1 (EECW Pp B3&D3 OC device)	Functionality	1.006	9.44E-08
Relay Group 4 for Group SEIS_11-1 (EECW Pp A3 OC device)	Functionality	1.006	9.44E-08
Wall 28 falls towards 480v SD BD 1A and 1B	Block Wall Failure	1.006	9.44E-08
Relays for Group SEIS_14-1, Relay Group 1 (HPCI/RCIC Isolations)	Functionality	1.006	9.44E-08
Seismic failure of Main Control Room instrumentation	Functionality	1.005	7.87E-08
Human Failure Events			
Operator fails to align N2 FLEX backup to Drywell Control Air (SRVs)	Not Applicable	1.294	3.57E-06
Operator fails to align Containment Air Dilution backup to Drywell Control Air	Not Applicable	1.232	2.96E-06
Failure to align RHR for suppression pool cooling (ATWS) or Inadvertent Opening of One Relief Valve (IOORV))	Not Applicable	1.126	1.76E-06
Failure to align RHR for suppression pool cooling (non-ATWS/IOORV)	Not Applicable	1.076	1.12E-06
Operator fails to align backup EECW pump	Not Applicable	1.062	9.17E-07
Operator fails to isolate SD BD and align alternate DC	Not Applicable	1.049	7.41E-07
Operator reset of 4kV Shutdown Board lockout relays (seismic)	Not Applicable	1.038	5.82E-07
Failure to align alternate battery charger	Not Applicable	1.031	4.75E-07
Operator fails to align alternate feeder	Not Applicable	1.028	4.29E-07
Failure to open doors and install fans after HVAC failure	Not Applicable	1.025	3.78E-07
Failure to transfer 480V shutdown board to alternate source	Not Applicable	1.019	2.89E-07
Failure to start standby Control Bay HVAC	Not Applicable	1.019	2.86E-07

Description	Failure Mode	RRW	MCR (/yr)
Operator fails to manually control level with High Pressure Coolant Injection	Not Applicable	1.018	2.77E-07
Failure to transfer deenergized 480v board to alternate supply	Not Applicable	1.013	1.95E-07
Failure to initiate reactor-vessel depressurization (transient or ATWS)	Not Applicable	1.011	1.64E-07
Failure to initiate reactor-vessel depressurization (transient or ATWS)	Not Applicable	1.009	1.35E-07
Seismic failure of Main Control Room instrumentation	Not Applicable	1.005	8.02E-08

Table 4. Importance Analysis Results of Top Contributors to Unit 2 Seismic LERF

Description	**Failure Mode**	**RRW**	**MLR (/yr)**
Seismically-failed SSCs			
LOOP (Loss of Offsite Power)	Ceramic Insulators	5.348	5.91E-06
Initiation relays and panels	Anchorage	1.233	1.37E-06
Unit Batteries (Fragility Group 15-03)	Anchorage	1.225	1.34E-06
Reactor Protection & NSS PNL (18-02)	Functionality	1.075	5.09E-07
Emergency Equipment Cooling Water pumps based on pipe frag calc	Soil (Buried Piping) Failure	1.046	3.20E-07
480v BD 219 A, B (U1, U2) (Fragility Group 01-05-01) Anchorage Failure	Anchorage	1.041	2.83E-07
Battery and Wall 8 or 47 interface (Block Wall Group 5)	Block Wall Failure	1.041	2.83E-07
Relays for Group SEIS_14-1, Relay Group 1 (HPCI/RCIC Isolations)	Functionality	1.037	2.62E-07
RHRSW pumps and EECW Alternate (Fragility Group 06-03)	Anchorage	1.034	2.40E-07
Initiation relays and panels (20-08)	Anchorage	1.034	2.25E-07
Relay Group 1-1 for Group SEIS_11-1 (EECW Pp B3 UV device)	Functionality	1.032	2.25E-07
Relay Group 2 for Group SEIS_11-1 (EECW Pp B3&D3 OC device)	Functionality	1.031	2.18E-07
Relay Group 3 for Group SEIS_11-1 (EECW Pp A3 UV device)	Functionality	1.031	2.18E-07
Intake Pumping Station	Structural Analysis	1.026	1.82E-07
3EA,3EC SDBD Common Accident Signal Relays (CASA)	Functionality	1.025	1.74E-07
Relay Group 4 for Group SEIS_11-1 (EECW Pp A3 OC device)	Functionality	1.024	1.67E-07
480v BD 219 A, B (U1, U2) (Fragility Group 01-05-01) Functional Failure	Functionality	1.022	1.60E-07
C SDBD 480 Transformer Trip relays (50G)	Functionality	1.021	1.53E-07

C DG BKR Trip Relays (CAR, OTX, RI, VRL, VRR)	Functionality	1.021	1.53E-07
3EA,3EC SDBD 480 Transformer Trip relays (50G)	Functionality	1.020	1.45E-07
3EA,3EC SDBD Lockout relays (86)	Functionality	1.020	1.45E-07
3EA,3EC SDBD Lockout relays (86)	Functionality	1.020	1.45E-07
Initiation relays and panels (20-07)	Functionality	1.020	1.45E-07
480v BD 219 3EA, 3EB (U3) (Fragility Group 01-05-02)	Anchorage	1.020	1.45E-07
U3 4kv SD BD EA and EC (Fragility Group 03-03)	Functionality	1.020	1.45E-07
250v DC Bus A interface with wall 64 (Block Wall Group 3)	Block Wall Failure	1.019	1.38E-07
EECW Pumps (Fragility Group 06-03-01)	Anchorage	1.018	1.31E-07
RHRSW Pumps based on pipe frag (Pipe Calc)	Soil (Buried Piping) Failure	1.018	1.31E-07
480V BD 1A, 2A, 3A, 1B, 2B, 3B (Fragility Group 02-01)	Functionality	1.017	1.24E-07
Seismic failure of Main Control Room instrumentation	Functionality	1.016	1.16E-07
C SDBD Common Accident Signal Relays (CASA)	Functionality	1.013	9.45E-08
B,D SDBD Common Accident Signal Relays (CASA)	Functionality	1.013	9.45E-08
Wall 63 falls towards 480v SD BD 2A and 2B	Block Wall Failure	1.012	8.72E-08
Panels Group 4-1 Control Room panels Lower Fragility (Fragility Group 20-02 and 20-03)	Functionality	1.011	7.99E-08
3A,3C DG BKR Trip relay (R3)	Functionality	1.008	5.81E-08
A SDBD 480 Transformer Trip relays (50G)	Functionality	1.008	5.81E-08
Wall 63 falls toward 480v RMOV 2A and 4kv SD BD C	Block Wall Failure	1.007	5.09E-08
CAD Nitrogen Storage Tank (084) (Fragility Group 21-06)	Anchorage	1.007	5.09E-08
Emergency Diesel Generator (EDG) (Fragility Group 17-01)	Functionality	1.006	4.36E-08
250v DC bus B (Fragility Group 01-02)	Functionality	1.006	4.36E-08

HPCI/RCIC PNLS (925-0058&63)	Anchorage	1.005	3.63E-08
Human Failure Events			
Operator fails to manually control level with High Pressure Coolant Injection	Not Applicable	1.051	3.53E-07
Failure to initiate RPV depressurization (transient or ATWS)	Not Applicable	1.045	3.15E-07
Operator fails to manually initiate injection into drywell after core damage	Not Applicable	1.034	2.38E-07
Operator Fails to Initiate Depressurization (SLERF)	Not Applicable	1.033	2.35E-07
Operator fails to align alternate feeder	Not Applicable	1.033	2.34E-07
Seismic failure of Main Control Room instrumentation	Not Applicable	1.017	1.20E-07
Operator fails to manually initiate injection for in-vessel recovery	Not Applicable	1.009	6.47E-08
Operator fails to manually initiate injection into drywell after core damage	Not Applicable	1.006	4.22E-08
Operator reset of 4kV Shutdown Board lockout relays(seismic)	Not Applicable	1.005	3.63E-08

Table 5. Importance Analysis Results of Top Contributors to Unit 3 Seismic CDF

Description	Failure Mode	RRW	MCR (/yr)
Seismically-failed SSCs			
LOSP (Loss of Offsite Power)	Ceramic Insulators	10.000	1.54E-05
480v BD 219 3EA, 3EB (U3) (Fragility Group 01-05-02)	Anchorage	1.052	8.40E-07
RHRSW pumps based on pipe frag (Pipe Calculation)	Soil (Buried Piping) Failure	1.048	7.89E-07
Emergency Equipment Cooling Water pumps based on pipe fragility calculation	Soil (Buried Piping) Failure	1.046	7.55E-07
Unit Batteries (Fragility Group 15-03)	Anchorage	1.037	6.17E-07
3EA,3EC SDBD Common Accident Signal Relays (CASA)	Functionality	1.033	5.49E-07
480v BD 219 A, B (U1, U2) (Fragility Group 01-05-01) Anchorage Failure	Anchorage	1.027	4.46E-07
3EA,3EC SDBD Common Accident Signal Relays (CASA)	Functionality	1.018	3.09E-07
U3 4kv SD BD EA and EC (Fragility Group 03-03)	Functionality	1.018	3.09E-07
RHRSW pumps and EECW Alternate (Fragility Group 06-03)	Anchorage	1.012	2.06E-07
Battery CH 248-1 (Fragility Group 16-06)	Functionality	1.011	1.89E-07
Relay Group 3 for Group SEIS_11-1 (EECW Pp A3 UV device)	Functionality	1.009	1.54E-07
Intake Pumping Station	Structural Analysis	1.009	1.54E-07
Initiation relays and panels	Anchorage	1.008	1.37E-07
Wall 28 falls towards 480v SD BD 1A and 1B	Block Wall Failure	1.008	1.37E-07
4160-480V Transformer, Control Bay (Fragility Group 4-12)	Block Wall Failure	1.008	1.37E-07
3EA,3EC SDBD Lockout relays (86)	Functionality	1.006	1.03E-07
3EA,3EC SDBD Lockout relays (86)	Functionality	1.006	1.03E-07
3EA,3EC SDBD 480 Transformer Trip relays (50G)	Functionality	1.005	8.58E-08

Description	Failure Mode	RRW	MCR (/yr)
Battery and Wall 8 or 47 interface (Block Wall Group 5)	Block Wall Failure	1.005	8.58E-08
Human Failure Events			
Operator fails to align N2 FLEX backup to Drywell Control Air (SRVs)	Not Applicable	1.242	3.34E-06
Operator fails to align Containment Air Dilution backup to Drywell Control Air	Not Applicable	1.181	2.62E-06
Failure to align RHR for suppression pool cooling (ATWS) or Inadvertent Opening of One Relief Valve (IOORV))	Not Applicable	1.111	1.72E-06
Failure to align RHR for suppression pool cooling (non-ATWS/IOORV)	Not Applicable	1.076	1.21E-06
Operator reset of 4kV Shutdown Board lockout relays (seismic)	Not Applicable	1.073	1.16E-06
Operator fails to isolate SD BD and align alternate DC	Not Applicable	1.050	8.20E-07
Failure to transfer 480V shutdown board to alternate source	Not Applicable	1.047	7.77E-07
Operator fails to align backup EECW pump	Not Applicable	1.042	6.88E-07
Failure to align alternate battery charger	Not Applicable	1.015	2.57E-07
Failure to transfer deenergized 480v board to alternate supply	Not Applicable	1.013	2.28E-07
Operator fails to manually control level with High Pressure Coolant Injection	Not Applicable	1.011	1.85E-07
Operator fails to align alternate feeder	Not Applicable	1.011	1.84E-07
Failure to initiate reactor-vessel depressurization (transient or ATWS)	Not Applicable	1.008	1.32E-07
Failure to open doors and install fans after HVAC failure	Not Applicable	1.006	1.08E-07
Failure to initiate reactor-vessel depressurization (transient or ATWS)	Not Applicable	1.006	9.78E-08
Failure to start standby Control Bay HVAC	Not Applicable	1.019	8.58E-08

Table 6. Importance Analysis Results of Top Contributors to Unit 3 Seismic LERF

Description	Failure Mode	RRW	MLR (/yr)
Seismically-failed SSCs			
LOOP (Loss of Offsite Power)	Ceramic Insulators	5.348	6.54E-06
Initiation relays and panels	Anchorage	1.242	1.57E-06
Unit Batteries (Fragility Group 15-03)	Anchorage	1.156	1.09E-06
480v BD 219 3EA, 3EB (U3) (Fragility Group 01-05-02)	Anchorage	1.071	5.31E-07
Reactor Protection & NSS PNL (18-02)	Functionality	1.057	4.34E-07
Emergency Equipment Cooling Water pumps based on pipe frag calc	Soil (Buried Piping) Failure	1.032	2.49E-07
480v BD 219 A, B (U1, U2) (Fragility Group 01-05-01) Anchorage Failure	Anchorage	1.030	2.33E-07
Battery and Wall 8 or 47 interface (Block Wall Group 5)	Block Wall Failure	1.026	2.01E-07
Relay Group 3 for Group SEIS_11-1 (EECW Pp A3 UV device)	Functionality	1.025	1.93E-07
U3 4kv SD BD EA and EC (Fragility Group 03-03)	Functionality	1.017	1.37E-07
480v BD 219 A, B (U1, U2) (Fragility Group 01-05-01) Functional Failure	Functionality	1.017	1.37E-07
Intake Pumping Station	Structural Analysis	1.016	1.29E-07
3EA,3EC SDBD Common Accident Signal Relays (CASA)	Functionality	1.015	1.21E-07
Relays for Group SEIS_14-1, Relay Group 1 (HPCI/RCIC Isolations)	Functionality	1.010	8.04E-08
Relay Group 4 for Group SEIS_11-1 (EECW Pp A3 OC device)	Functionality	1.010	8.04E-08
Initiation relays and panels (20-08)	Anchorage	1.010	8.04E-08
Wall 95 falls towards 480v RMOV 3A and 250vdc RMOV BD 3A	Block Wall Failure	1.010	8.04E-08
480V BD 1A, 2A, 3A, 1B, 2B, 3B (Fragility Group 02-01)	Functionality	1.008	6.43E-08

Description	Failure Mode	RRW	MLR (/yr)
Panel Group 4-1 Control Rm Lower Fragility (Fragility Group 20-02 and 20-03)	Anchorage	1.008	6.43E-08
RHRSW pumps based on pipe frag (Pipe Calc)	Soil (Buried Piping) Failure	1.007	5.63E-08
RHRSW pumps and EECW Alternate (Fragility Group 06-03)	Anchorage	1.006	4.83E-08
Human Failure Events			
Operator fails to manually control level with High Pressure Coolant Injection	Not Applicable	1.041	3.16E-07
Failure to initiate RPV depressurization (transient or ATWS)	Not Applicable	1.036	2.78E-07
Operator fails to align alternate feeder	Not Applicable	1.033	2.61E-07
Operator fails to manually initiate injection into drywell after core damage	Not Applicable	1.031	2.40E-07
Operator Fails to Initiate Depressurization (SLERF)	Not Applicable	1.029	2.24E-07
Operator reset of 4kV Shutdown Board lockout relays(seismic)	Not Applicable	1.026	2.03E-07
Failure to transfer 480V shutdown board to alternate source	Not Applicable	1.024	1.88E-07
Operator fails to manually initiate injection for in-vessel recovery	Not Applicable	1.008	6.03E-08
Operator fails to manually initiate injection into drywell after core damage	Not Applicable	1.005	4.02E-08

AUDIT SUMMARY BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO

BROWNS FERRY NUCLEAR PLANT, 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-2019-JLD-0023)

BACKGROUND AND AUDIT BASIS

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

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

By letter dated July 6, 2017 (ADAMS Accession No. ML17177A446), the NRC issued a generic audit plan and entered into the audit process described in Office Instruction LIC-111, "Regulatory Audits," dated December 29, 2008 (ADAMS Accession No. ML082900195), to assist in the timely and efficient closure of activities associated with the letter issued pursuant to 10 CFR Part 50, Section 50.54(f). The list of applicable licensees in Enclosure 1 to the July 6, 2017, letter included Tennessee Valley Authority (TVA) as the licensee for Browns Ferry Nuclear Plant, Units 1, 2, and 3 (Browns Ferry, BFN).

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 May 13, 2020 (ADAMS Accession No. ML20254A303), were sent to the licensee to support the audit.

The licensee provided clarifying information in the following areas:

- The peer review process and confirmation that the seismic PRA full-scope peer review considered the NRC staff's comments and proposed resolutions regarding the use of PRA Standard ASME/ANS RA-S Case 1 provided with its letter dated March 12, 2018 (ADAMS Accession No. ML18017A963).

- Dispositions for some open internal events PRA (IEPRA) Facts and Observations related to human failure events and their potential impact on the SPRA results.

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

DOCUMENTS AUDITED

- Jensen-Hughes Report 006069-RPT-01, Revision 0, "Brown Ferry Seismic PRA Fact and Observation Closure Independent Assessment," November 2019.

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- PRA Evaluation BFN-0-18-116, Revision 1, "BFN Seismic PRA Seismic-Fire Interaction," March 2019.

- BWR Owners Group Report "BROWNS FERRY UNITS 1,2,3 PRA Peer Review Report Using ASME PRA Standard Requirements," August 2009.

- ABS Consulting, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," October 2009.

- P3146-1000-01, Revision 0, “F&O Closeout by Independent Assessment Report for the Browns Ferry Nuclear Plant (BFN) Internal Events PRA Model Against the ASME PRA Standard Requirements and NEI 05-04 Appendix X,” November 2018.

- P0132150002-5175, Revision 0, “BFN PRA Focused Scope Peer Review Final Report,” August 31, 2015.

- Report CJC-BFN-C-001, Revision 1, “Updated Soil Failure and Fragility Analysis for the Browns Ferry Nuclear Plant (BFN),” September 2019.

- SC Solutions Report BFN-17-001, Revision 2, “Browns Ferry Nuclear Plant Seismic Probabilistic Risk Assessment: Structural Response Analysis,” September 2019.

- ENERCON Report TVAEBFN062-REPT-002, Revision 2, “Browns Ferry Nuclear Components and Structures Fragility Evaluation,” December 2019.

- PRA Evaluation BFN-0-19-115, Revision 0. “Updated Fragility Estimates for Selected Components,” November 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 Browns Ferry.