UNITED STATES
NUCLEAR REGULATORY COMMISSION
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June 10, 2019

Mr. Bryan C. Hanson
Senior Vice President
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President and Chief Nuclear Officer
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SUBJECT: PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – STAFF REVIEW OF SEISMIC PROBABILISTIC RISK ASSESSMENT ASSOCIATED WITH REEVALUATED SEISMIC HAZARD IMPLEMENTATION OF THE NEAR-TERM TASK FORCE RECOMMENDATION 2.1: SEISMIC (EPID NO. L-2018-JLD-0010)

Dear Mr. Hanson:

The purpose of this letter is to document the staff's evaluation of the Peach Bottom Atomic Power Station, Units 2 and 3 (Peach Bottom), 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 Peach Bottom.

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

By letter dated August 28, 2018 (ADAMS Accession No. ML18240A065), Exelon Generation Company, LLC (Exelon, the licensee), provided its SPRA submittal in response to Enclosure 1, Item (8) of the 50.54(f) letter, for Peach Bottom. 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), as endorsed by NRC letter dated February 15, 2013 (ADAMS Accession No. ML12319A074), through the completion of the reviewer checklist in Enclosure 1 to this letter. As described below, the NRC has concluded that the Peach Bottom 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. ML14090A247), Exelon submitted the reevaluated seismic hazard information for Peach Bottom. The NRC performed a staff assessment of the submittal and issued a response letter on April 20, 2015 (ADAMS Accession No. ML15051A262). The NRC's assessment concluded that the licensee 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.

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, Peach Bottom was expected to complete an SPRA, which would also assess high frequency ground motion effects, and a limited-scope evaluation for the spent fuel pool. The limited-scope evaluation for the spent fuel pool was submitted by letter dated December 15, 2017 (ADAMS Accession No. ML17349A096). The staff provided its assessment of this evaluation in a letter dated July 10, 2018 (ADAMS Accession No. ML18187A403). The Peach Bottom SPRA submittal was expected to be submitted to the NRC by March 31, 2018. Subsequently in a letter dated March 15, 2018 (ADAMS Accession No. ML18074A303), the licensee requested an extension of the submittal date for the SPRA until September 28, 2018. In a letter dated April 24, 2018 (ADAMS Accession No. ML18093B511), the staff deferred the SPRA submittal required response date until September 28, 2018.

The completion of the April 20, 2015, NRC staff assessment for the reevaluated seismic hazard and the scheduling of Peach Bottom SPRA submittal described in the NRC's October 27, 2015, letter marked the fulfillment of the Phase 1 process for Peach Bottom.

In its August 28, 2018, letter, Exelon provided the SPRA submittal that initiated the NRC's Phase 2 decisionmaking process for Peach Bottom. The NRC described this Phase 2 decisionmaking process in a guidance memorandum from the Director of the Japan Lessons-Learned Division to the Director of the Office of Nuclear Reactor Regulation (NRR) on

September 21, 2016 (ADAMS Accession No. ML16237A103). This memorandum details 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 the recommendation for the screening decisions for consideration by the panel. In presenting recommendations to the SMRP, the supporting technical staff is expected to recommend placement of each SPRA plant into one of three groups:

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

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

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

The evaluation performed to provide the basis for the staff's grouping recommendation to the SMRP for Peach Bottom is described below. Based on its evaluation, the staff recommended to the SMRP that Peach Bottom be classified as a Group 1 plant and therefore, no further regulatory action was warranted.

EVALUATION

Upon receipt of the licensee's August 28, 2018, SPRA submittal, a technical team of staff 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 October 2, 2018, 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 August 28, 2018, submittal, the licensee developed and documented the SPRA in accordance with the SPID guidance, including performing a full-scope peer review against Part 5 of Addendum B to the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS), "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," (RA-Sb-2013). Appendix A of the licensee's submittal provided a summary of the full-scope peer review including, excerpts from the corresponding peer review report. Appendix A included the open SPRA finding level facts and observations (F&Os) along with licensee's dispositions which were reviewed by NRC staff in the context of the regulatory decisionmaking associated with the 50.54(f) letter.

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

The staff developed questions to verify information in the licensee's submittal and to gain understanding of non-docketed information that supports the docketed SPRA submittal. The staff's clarification questions dated February 6, 2019, and February 11, 2019 (ADAMS Accession Nos. ML19037A483, and ML19044A356, respectively), were sent to the licensee to support the audit. The licensee subsequently provided answers to the questions in the eportal, which the staff reviewed.

The staff determined that the answers to the questions provided in the eportal served to verify statements that the licensee made in its August 28, 2018, SPRA submittal. The findings from the licensee's internal events PRA were not provided in the submittal. However, the internal events PRA was reviewed by the staff to support the Peach Bottom license amendment to adopt Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, "Risk-Informed Categorization and Treatment of Structures, System and Components for Nuclear Power Plants." The staff's review of the internal events PRA that supported this license amendment can be found in a safety evaluation dated October 25, 2018 (ADAMS Accession No. ML18263A232). The safety evaluation dated October 25, 2018, identified a few commitments to update the internal events PRA (which provides the foundation for the SPRA plant response model) before implementing the risk-informed categorization process. As part of the audit, the NRC staff requested information about modelling updates that appeared to NRC staff to have the potential to impact the SPRA model results. In response, the licensee provided the results of a sensitivity study showing that incorporation of those updates would not change 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 SPID in response to the 50.54(f) letter. The SPRA Checklist also focuses on areas where the SPID contains differing guidance from standard industry SPRA guidance. Enclosure 1 contains the staff's application of the SPRA checklist to Peach Bottom's submittal. As documented in the checklist, the staff concluded that the Peach Bottom SPRA met the intent of the SPID. The staff further concluded that the peer review findings have been closed-out in accordance with the ASME/ANS Standard RA-Sb-2013 process.

Following the staff's conclusion on the SPRA's technical adequacy, the staff reviewed the risk and safety insights contained in the Peach Bottom SPRA submittal. The staff also used the screening criteria described in the August 29, 2017 (ADAMS Accession No. ML17146A200), staff memorandum titled, "Guidance for Determination of Appropriate Regulatory Action Based on Seismic Probabilistic Risk Assessment Submittals in Response to Near Term Task Force Recommendation 2.1: Seismic" to assist in determining the group in which the technical team would recommend placing Peach Bottom 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 and Information Collection," dated October 9, 2013 (ADAMS Accession No. ML12059A460), to the SPRA submittal review. The Peach Bottom SPRA submittal demonstrated that the plant SCDF and SLERF for both units were not below the initial screening values in the August 29, 2017, staff memorandum. As a result, the NRC staff utilized the Peach Bottom SPRA submittal and other available information in conjunction with the guidance in the August 29, 2017, memorandum to complete a detailed screening with respect to SCDF and SLERF for Peach Bottom. The detailed screening concluded that Peach Bottom should be considered a Group 1 plant because:

- Sufficient reductions in SCDF and/or 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, 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 Peach Bottom SPRA submittal, the technical team determined that recommending Peach Bottom 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 Peach Bottom review to the TRB with the Phase 2 recommendation that Peach Bottom 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 Peach Bottom as a Group 1 plant.

Subsequently, the technical review team met with the SMRP and presented the results of the review including the recommendation for Peach Bottom 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

Peach Bottom should be classified as a Group 1 plant, meaning that no further response or regulatory action is required.

AUDIT REPORT

The July 6, 2017, generic audit plan describes the NRC staff's intention to issue an audit report that summarizes and documents the NRC's regulatory audit of licensee's SPRA submittals associated with their reevaluated seismic hazard information. The NRC staff's Peach Bottom 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 Peach Bottom 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" review. The staff notes that assessment of the SPRA for use in other licensing applications, would warrant review of the SPRA for its intended application. The NRC may use insights from this SPRA assessment in its regulatory activities as appropriate.

If you have any questions, please contact Joseph Sebrosky at (301) 415-1132 or via e-mail at Joseph.Sebrosky@nrc.gov.

Sincerely,

Louise Lund

Louise Lund, Director
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Office of Nuclear Reactor Regulation

Docket Nos. 50-277 and 50-278

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

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NRC Staff SPRA Submittal Technical Review Checklist

Several nuclear power plant licensees are performing seismic probabilistic risk assessments (SPRAs) as part of their required submittals to satisfy Near-Term Task Force (NTTF) Recommendation 2.1: Seismic. These submittals are 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 staff for this purpose (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12319A074). The SPRA peer reviews are also expected to follow the guidance in NEI 12-13 (NEI, 2012).

The SPID indicates that an SPRA submitted to satisfy NTTF Recommendation 2.1: Seismic must meet the requirements in the ASME-ANS Probabilistic Risk Assessment (PRA) Methodology Standard (the ASME/ANS Standard). 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.

Tables 6-4, 6-5, and 6-6 of the SPID also provide a comparison of each of the Supporting Requirements (SRs) of the ASME/ANS Standard to the relevant guidance in the SPID. For most SRs, the SPID guidance does not differ from the requirement in the ASME/ANS Standard. However, because the guidance of the SPID and the criteria of the ASME/ANS Standard differ in some areas, or the SPID does not explicitly address an SR, the staff developed this checklist, in part, to help staff members to address and evaluate the differences.

In general, the SPID allowed departures or differed 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 ASME/ANS 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 ASME/ANS Standard.

(iii) The SPID has some requirements that are not in the ASME/ANS Standard.

The technical positions in the SPID have been endorsed by the U.S. Nuclear Regulatory Commission (NRC) staff, subject to certain conditions concerning peer review outlined in the staff's November 12, 2012, letter to NEI (NRC, 2012).

The following checklist is comprised of the 16 "Topics" that require additional staff guidance because the SPID contains specific guidance that differs from the ASME/ANS Standard or expands on it. Each is covered below under its own heading, "Topic 1," "2," etc. The checklist was discussed during a public meeting held on December 7, 2016 (ADAMS Accession No. ML16350A181).

- 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 Safe Shutdown Earthquake (SSE) to Ground Motion Response Spectrum (GMRS) 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, In-Structure Response Spectra (ISRS), and Fragilities
- Topic 8: Screening by Capacity to Select Structures, Systems, and Components (SSCs) for Seismic Fragility Analysis (SPID Section 6.4.3)
- Topic 9: Use of the Conservative Deterministic Failure Margin (CDFM)/HybridMethodology 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 Seismic Large Early Release Frequency (SLERF) (SPID Section 6.5.1)
- Topic 14: Peer Review of the SPRA, Accounting for NEI 12-13 (SPID Section 6.7)
- Topic 15: Documentation of the SPRA (SPID Section 6.8)
- Topic 16: Review of Plant Modifications and Licensee Actions

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

The site under review has updated/revised its Probabilistic Seismic Hazard Analysis (PSHA) from what was submitted to NRC in response to the NTTF Recommendation 2.1: Seismic 50.54(f) letter.	No
Notes from staff reviewer: Minor changes to the PSHA that supported the SPRA were made from that provided in response to NTTF Recommendation 2.1. These minor changes are described in Section 3.1 of the SPRA report and include development of additional elements required for the Seismic PRA such as foundation input response spectra, hazard-consistent strain-compatible properties, and vertical ground motions. 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 peer review findings referred to relate to the Probabilistic Seismic Hazards Analysis (SHA) requirements in the ASME/ANS Standard, as well as to the requirements in the SPID.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The guidance in the SPID was followed for developing the probabilistic seismic hazard for the site.	Yes
• An alternate approach was used and is acceptable on a justified basis.	N/A

TOPIC 2: Site Seismic Response (SPID Section 2.4)

The site under review has updated/revised its site response analysis from what was submitted to NRC in response to the NTTF Recommendation 2.1: Seismic 50.54(f) letter.	No
Notes from staff reviewer: See notes in Topic 1. 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 peer review findings referred to relate to the SRs SHA-E1 and E2 in the ASME/ANS Standard, as well as to the requirements in the SPID.	Yes
• Although some peer review findings 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 V_s velocity 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 criteria 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.	N/A
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.	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 staff for the purposes of this evaluation. The peer review findings referred to relate to the requirements in the SPID. No requirements in the ASME/ANS Standard specifically address this topic.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's definition of the control point for site response analysis adequately meets the intent of the SPID guidance.	Yes
• The licensee's definition of the control point for site response analysis does not meet the intent of the SPID guidance, but is acceptable on another justified basis.	N/A

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

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

Notes from staff reviewer:

1. Existing structural models – SPRA Section 4.3.3 – Existing models were not used for any structures.

2. Enhancement of existing models – SPRA Section 4.3.3 –

    a. Existing lumped-mass-stick model (LMSM) for the Diesel Generator Building was enhanced by adding oscillators to capture floor response and outriggers to capture response at the building corners.

    b. Existing LMSM for Pump Structure was enhanced by adding oscillators to capture floor response, outriggers to capture response at the building corners, and additional discretization of the LMSM. The Pump Structure model was enhanced by connecting it to a flat foundation finite element slab model.

3. Entirely new models – SPRA Section 4.3.3 –

    a. A new 3D finite element method (FEM) analyses were used for Reactor Building complex that included the Reactor Building, Turbine Building, Radwaste Building, and Main Control Room in a single model. Cracked and uncracked concrete models were used.

    b. Emergency Cooling Tower is a redundant structure required if the Conowingo Dam fails and therefore considered risk-significant. A new 3D FEM analyses were used for Emergency Cooling Tower to reduce potential conservatisms in structural fragilities.

4. Building response was not evaluated for FLEX Storage Building, which is founded on piles. The foundation level earthquake was used directly to assess capacity/demand for the non-operation FLEX equipment that is stored in this building. Use of foundation level earthquake is appropriate for equipment stored and not mounted to the floor of this building.

5. Provisions in Criteria 1-7: SPID Section 4.3.3 have been met. SPID Section 6.3.1 Criteria 1 through 7:

(i) The LMSM and FEM structural models are capable of capturing overall structural responses for both vertical and horizontal components of ground motion. (ii) For all soil-structure interaction (SSI) analyses, ground motion in three spatial directions were considered simultaneously (SPRA Section 4.3.2). (iii) LMSM and FEM structural models include structural mass and rotational inertia. (iv) The cutoff frequency for SSI was 50 hertz (SPRA Section 4.3.2) (v) 3D models consider torsional effects including out-of-plane response and in-plane diaphragm effects. (vi) "One-Stick" model was not used. (vii) In plane floor flexibility was used. Based on information provided in Table A-2 in the SPRA submittal, the review findings on SFR-C1 (F&O 5-15) were adequately addressed by using both cracked and uncracked concrete 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 staff for the purposes of this evaluation. The peer review findings referred to relate to the SRs Seismic Fragility Analysis (SFR)-C1 through C6 in the ASME/ANS Standard, as well as to the requirements in the SPID.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's structural model meets the intent of the SPID guidance.	Yes
• The licensee's structural model does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

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

Fixed-based dynamic seismic analysis of structures was used, for sites previously defined as "rock." If no, this issue is moot.	No
Structure #1: If used, is shear velocity (V_S)> about 5000 feet (ft.)/second (sec.)?	N/A
If 3500 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: Based on SPRA Section 4.3.1 and Table 4.3-1, fixed-base analysis was used only for verification of 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 staff for the purposes of this evaluation. The peer review findings referred to relate to the requirements in the SPID. No requirements in the ASME/ANS Standard specifically address this topic.	N/A
• Although some peer review findings 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.	No
Potential Staff Findings: If a new uniform hazard spectra or review level earthquake is used, the shape is approximately similar to the spectral shape previously used for ISRS generation. If the shape is not similar, the justification for seismic response scaling is adequate. Consideration of non-linear effects is adequate.	N/A N/A N/A
Notes from staff reviewer: Seismic Response Scaling of ISRS was not used. Structural response to obtain ISRS is discussed in SPRA Section 4.3.3. 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 peer review findings referred to relate to the SR SFR-C3 in the ASME/ANS Standard, as well as to the requirements in the SPID. • Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis. • The licensee's use of seismic response scaling adequately meets the intent of the SPID guidance. • The licensee's use of seismic response scaling does not meet the intent of the SPID guidance but is acceptable on another justified basis.	 N/A N/A N/A N/A

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

The SPID does not provide specific guidance on performing new response analysis for use in developing ISRS and fragilities. The new response analysis is generally conducted when the criteria for use of existing models are not met or more realistic estimates are deemed necessary. The requirements for new analysis are included in the ASME/ANS Standard. See SRs SFR-C2, C4, C5, and C6. 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 in-structure-response-spectrum analysis. Said another way, an acceptable SPRA must use these analysis pieces together in a consistent way. The following are high-level key elements that should have been considered:	
1. Foundation Input Response Spectra (FIRS) site response developed with appropriate building specific soil velocity profiles. Structure #1: Reactor Building Complex Structure #2: Diesel Generator Building Structure #3: Emergency Cooling Tower Structure #4: Pump Structure Are all structures appropriately considered?	 Yes Yes Yes Yes Yes
2. Are models adequate to provide realistic structural loads and response spectra for use in the SPRA? • Is the SSI analysis capable of capturing uncertainties and realistic? • Is the probabilistic response analysis capable of providing the full distribution of the responses?	Yes Yes N/A

Notes from staff reviewer:

1. Reactor Building complex (Reactor Building, Turbine Building, Radwaste, and Main Control Room) – founded on rock; SSI consists of incoherency, three structural property variation cases (Best Estimate (BE), Lower Bound (LB), and Upper Bound (UB)), and five time histories.
2. Diesel Generator Building – foundation consists of shear walls and bearing piles supported on rock; SSI consists of incoherency and three soil property variation cases (BE, LB, and UB).

3. Emergency Cooling Tower - founded on rock; SSI consists of incoherency, three structure cases and five time histories.
4. Pump Storage - founded on rock; SSI consists of incoherency, three structure cases and five time histories; included uncertainties for embedment conditions.
5. Buildings founded on rock – uncertainties are addressed by considering three structure cases and five time histories (find details). Rock properties were not varied.
6. Building found on load bearing piles – three cases of soil and three cases for structures.

Based on information provided in Table A-2 in the SPRA submittal, the review finding on SFR-C5 (F&O 5-11) is associated with the pounding (impact) between buildings. The pounding between the buildings in the Reactor Building Complex is limited because the buildings are on a common base mat. Pounding in locations near relay cabinets was addressed because the cabinet fragilities were lower than the building fragility required to produce pounding.

Based on information provided in Table A-2 in the SPRA submittal, the review findings on SFR-F1 (F&O 5-21 and 5-22) associated with the fragility of distributed piping have been properly addressed.

Based on information provided in Table A-2 in the SPRA submittal, the review finding on SFR-G2 (F&O 5-8) is associated with building fragilities. Additional review of supporting documents showed standard practice was followed for development of both demand and capacity for buildings.

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 peer review findings referred to relate to the SRs SFR-C2, C4, C5, and C6 in the ASME/ANS Standard, as well as to the requirements in the SPID.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's FIRS modeling is consistent with the prior NRC review of the GMRS and soil velocity information.	Yes
• The licensee's structural model meets the intent of the SPID guidance and the ASME/ANS Standard's requirements.	Yes

• The response analysis accounts for uncertainties in accordance with the SPID guidance and the ASME/ANS 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 ASME/ANS 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 <u>no</u>, see items D and E. If <u>yes, see items A, B, and C.</u> <u>Potential Staff Findings</u>:	Yes
A) The recommendations in Section 6.4.3 of the SPID were followed for the screening aspect of the analysis, using the screening criteria therein.	Yes
B) The approach for retaining certain SSCs in the model with a screening-level seismic capacity follows the recommendations in Section 6.4.3 of the SPID and has been appropriately justified.	Yes
C) The approach for screening out certain SSCs from the model based on their inherent seismic ruggedness follows the recommendations in Section 6.4.3 of the SPID and has been appropriately justified.	Yes
D) The ASME/ANS 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:

Screening of risk significant SSCs is based on three quantification stages. At each stage, a sensitivity analysis was performed with an SPRA model to address screening levels. After each stage fragilities were refined:

1. Representative fragilities for all items in the seismic equipment list (SEL).
2. Enhanced fragilities using detailed CDFM calculations for top contributors for SCDF and SLERF.
3. Fragilities using Separation of variable (SOV) for dominant contributors to risk.
4. Licensee provided documentation on fragility evaluation for Reactor Building and relays demonstrating use of the quantification process.

Based on information provided in Table A-2 of the SPRA submittal and the review finding on SFR-B1 (F&O 5-23), cable trays were assigned a 1.8g peak spectral in-structure high confidence low probability of failure (HCLPF) capacity. Subsequent

analysis showed cable trays had a higher capacity than associated equipment and the Fragility Report was updated. 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 peer review findings referred to relate to the SR SFR-B1 in the ASME/ANS Standard, as well as to the requirements in the SPID.	Yes
• Although some peer review findings 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 Conservative 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 HCLPF 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: The licensee stated in Section 4.4.2.2 of the SPRA submittal that generic aleatory variability and epistemic uncertainty were based on the SPID. The review of limited fragilities in supporting documents shows that the values used for variability parameters (β_U, β_R, and β_C) are either same as SPID Table 6-2 or more conservative. Conowingo Dam – Fragility of Conowingo dam was initially considered to be same as a loss of offsite power in the SPRA model. Subsequent to peer review comment (F&O 5-16) suggesting a more refined and realistic fragility, the licensee developed a structural fragility for the dam. Other failure modes were screened out based on expert judgment. 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 peer review findings referred to relate to the requirements in the SPID. No requirements in the ASME/ANS Standard specifically address this Topic.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's use of the CDFM/Hybrid method for seismic fragility analysis meets the intent of the SPID guidance.	Yes
• The licensee's use of the CDFM/Hybrid method for seismic fragility analysis does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

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

The SPID requires that certain SSCs that are sensitive to high-frequency seismic motion must be analyzed in the SPRA for their seismic fragility using a methodology described in Section 6.4.2 of the SPID.	
Potential Staff Findings: The NRC staff review of the SPRA's fragility analysis of SSCs sensitive to high frequency seismic motion finds that the analysis is acceptable.	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: The licensee stated in Section 4.1.2 of the SPRA submittal, that the evaluation of relays including circuit breakers and motor starters is based on the guidance in Section 6.4.2 of the SPID. Relay chatter scenarios were screened initially based on assessment of impact on component functions. Based on information provided in Table A-2 in the SPRA the review finding on SFR-D2 (F&O 5-25) associated with the anchorage evaluations, the licensee showed that inclusion of equipment high frequency modes had negligible impact. Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): None.	
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 peer review findings referred to relate to the SR SFR-F3 in the ASME/ANS Standard, as well as to the requirements in the SPID.	Yes

• Although some peer review findings 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 ASME/ANS Standard is generally acceptable for the fragility analysis of these components, the SPID (Section 6.4.2) contains additional guidance when either circuit analysis or operator-action analysis is used as part of the SPRA to understand a given relay's role in plant safety. When one or both of these are used, the NRC reviewer should use the following elements of the checklist.	
i) Circuit analysis: The seismic relay-chatter analysis of some relays relies on circuit analysis to assure that safety is maintained. (A) If no, then (B) is moot. (B) If yes: Potential Staff Finding: The approach to circuit analysis for maintaining safety after seismic relay chatter is acceptable.	Yes Yes
ii) Operator actions: The relay-chatter analysis of some relays relies on operator actions to assure that safety is maintained. (A) If no, then (B) is moot. (B) If yes: Potential Staff Finding: The approach to analyzing operator actions for maintaining safety after seismic relay chatter is acceptable.	Yes Yes
Notes from staff reviewer: Use of circuit analysis for relay chatter to screen relays is stated in supporting documents. The licensee also stated the circuit analysis was performed in accordance with the requirements in the ASME/ANS SPRA Standard and that it meets the SPID.	

Operator recovery actions are credited in the SPRA model in response to relay chatter. This is discussed in Section 4.1.2 of the submittal and supporting documents. The licensee stated that quantification of operator action in human reliability analysis is consistent with the ASME/ANS PRA standard. Based on information provided in Table A-2 in the SPRA regarding the review finding on SFR- G2 (F&O 5-8 Item 6) associated with relay capacities, the licensee showed that inclusion of equipment high frequency modes had negligible impact on the SPRA results. 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 peer review findings referred to relate to the SRs Seismic Plant Response Analysis (SPR)-B6 (Addendum A) or SPR-B4 (Addendum B) in the ASME/ANS Standard, as well as to the requirements in the SPID.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's analysis of seismic relay-chatter effects meets the intent of the SPID guidance.	Yes
• The licensee's analysis of seismic relay-chatter effects does not meet the intent of the SPID guidance, but is acceptable on another justified basis.	N/A

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

The CDFM methodology has been used in the SPRA for analysis of the bulk of the SSCs requiring seismic fragility analysis. 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. If yes, significant risk contributors for which use of SOV fragility calculations would make a significant difference in the SPRA results have been selected for SOV calculations.	No
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 SOV 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 4.4.1 of the SPRA submittal states that the first risk quantification for all equipment on the seismic equipment list (SEL) was performed using representative fragilities based on site-specific scaling and simplified analyses. The submittal explains that more enhanced fragilities were developed for the second quantification using a detailed CDFM approach. The second quantification was completed for important SSCs identified based on an Fussell-Vesely (F-V) importance analysis. For the third quantification, the licensee explained that detailed fragilities were developed using the SOV approach using the F-V importance analysis from the second quantification. Rationale for not refining the representative fragility analysis for a handful of exceptions was provided in Sections 5.4 and 5.5 of the submittal (i.e., the fragility of offsite power sources and SSCs in which significantly increasing the capacity factor would have only a minimal impact on SCDF and SLERF.) Accordingly, the results of the three-tiered approach achieved detailed fragility analyses for the dominant risk contributors.

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 peer review findings referred to relate to the requirements in the SPID. No requirements in the ASME/ANS Standard specifically address this Topic.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's method for selecting the "dominant risk contributors" for further seismic fragilities analysis using the SOV methodology meets the intent of the SPID guidance.	Yes
• The licensee's method for selecting the "dominant risk contributors" for further seismic fragilities analysis using the SOV methodology does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

TOPIC 13: Evaluation of SLERF (SPID Section 6.5.1)

The NRC staff review of the SPRA's analysis of SLERF finds an acceptable demonstration of its adequacy.	Yes
Potential Staff Findings:	
A) The analysis follows each of the elements of guidance for SLERF analysis in Section 6.5.1 of the SPID, including in Table 6-3.	Yes
B) The SLERF analysis does not follow the guidance in Table 6-3 but the analysis is acceptable on another justified basis.	NA

Notes from staff reviewer:

Section 4.1 of the SPRA submittal explains that the SEL for each unit includes SSCs that prevent or mitigate radioactivity release if core damage occurs and explains that the SSCs included in the SEL are included in the SPRA models. Table 4.1.1-1 of the submittal identifies LERF-related critical safety functions (i.e., Containment Pressure and Temperature Control, Vapor Suppression, and Containment Isolation) and the systems that support those functions. The LERF contributors listed in Table 6-3 of the SPID either had no significant seismic-induced impact (per Table 6-3); were determined by NRC staff not to apply to a BWR; or were judged by NRC staff to be addressed in Section 4.1 of the submittal.

Section 5.1 of the submittal describes the SPRA logic model including transfer of core damage sequences from the Level 1 event trees to the Level 2 Containment Event Trees (CETs). The submittal explains that the seismic CETs used the same LERF timing and radionuclide release categories as the internal events PRA. The submittal explains that SSCs with a potential impact on containment integrity (e.g., containment bypass scenarios) were also evaluated and modeled accordingly for the Level 2 LERF model.

Section 5.5 of the submittal presents importance values for LERF-significant SSC seismic fragility failure groups and operator failures.

No open F&Os associated with LERF are unresolved for this submittal. (See Topic 14 of the NRC staff review). The SPRA submittal does not discuss the impact of a seismic event on emergency plans, which is acceptable per the SPID for NTTF Recommendation 2.1.

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 peer review findings referred	Yes

to relate to SRs SFR-F4, SPR-E1, SPR-E2, and SPR-E6 (Addendum B only) in the ASME/ANS Standard, as well as to the requirements in the SPID.	
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's analysis of SLERF meets the intent of the SPID guidance.	Yes
• The licensee's analysis of SLERF 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. Potential Staff Findings: A) The analysis follows each of the elements of the peer review guidance in Section 6.7 of the SPID.	Yes Yes
B) The composition of the peer review team meets the SPID guidance.	Yes
C) The peer reviewers focusing on seismic response and fragility analysis have successfully completed the Seismic Qualifications Utility Group 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 submittal. 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 guidance in the SPID (Section 6.7), including the three "bullets" and the guidance related to NRC's additional input in the paragraph immediately following those three bullets. These three bullets are: • The SPRA findings should be based on a consensus process, and not based on a single peer review team member • A final review by the entire peer review team must occur after the completion of the SPRA project • An "in-process" peer review must assure that peer reviewers remain independent throughout the SPRA development activity.	 N/A
If no, go to (F).	

If yes, the "in process" peer review approach is acceptable. Go to (G).	
E) The "end-of-process" peer review process followed the peer review guidance in the SPID (Section 6.7). If no, go to (F). If yes, the "end-of-process" peer review approach is acceptable. Go to (G).	Yes
F) The peer-review process does not follow the guidance in the SPID but is acceptable on another justified basis.	N/A
G) The licensee peer-review findings were satisfactorily resolved or were determined not to be significant to the SPRA conclusions for this evaluation.	Yes

Notes from staff reviewer:

The Peach Bottom SPRA submittal follows the recommendations of Section 6.7 of the SPID. Section 5.2 and Appendix A of the SPRA submittal describe the peer review process used to establish the technical adequacy of the SPRA. All elements of the SPRA were peer reviewed.

A full-scope peer review of the SPRA was conducted in March 2017 in accordance with: 1) NEI 12-13, "External Hazard PRA Peer Review Process Guidelines," Revision 0, dated August 2012 (ADAMS Accession No. ML122400044); 2) Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, dated March 2009 (ADAMS Accession No. ML090410014); and 3) Capability Category II requirements of PRA Standard ASME/ANS RA Sb-2013, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," dated September 30, 2013, which is endorsed in the SPID for response to the 50.54(f) letter. Appendix A of the submittal described the qualifications of each of the eight peer review members. The combined experience of the eight reviewers spanned the three technical elements of the SPRA: hazards analysis, fragility analysis, and plant response. One team member was assigned the lead for each of the three areas and one member was designated as the overall team leader. The submittal states that seismic walkdowns were performed by two members with the appropriate Seismic Qualification Users Group (SQUG) training and an additional member with expertise in the Seismic Plant Response (SPR) technical element.

All elements of the SPRA were peer reviewed, including those identified in Section 6.7 of the SPID, and the 29 Finding-level Facts and Observations (F&Os) resulting from the peer review were provided in Table A-2 of the submittal. This documentation includes the description of the Finding, the basis for the Finding, and the resolutions suggested by the peer review appear along with dispositions by the licensee to each Finding. The NRC staff reviewed these F&Os, as well as their corresponding dispositions and the licensee's responses to staff audit questions on certain dispositions. Based on its review, the staff concluded that the F&Os are sufficiently dispositioned for this submittal.

Section 5.2 and Appendix A of the submittal describe the peer review process used to establish the technical adequacy of the SPRA and internal events PRA. The internal events PRA (including internal flooding) which is the foundation for the SPRA, was peer reviewed in November 2010 by the Boiling Water Reactor Owners Group against the CC-II supporting requirements of the ASME/ANS PRA Standard RA-Sa-2009 and in accordance with Regulatory Guide (RG) 1.200, Revision 2. No open F&Os were presented in the submittal. The licensee states in Section A.7 of the submittal that "[a]ll of the internal events and internal flooding PRA peer review findings that may affect the SPRA model have been addressed." Additionally, the internal events PRA was reviewed by the staff to support the Peach Bottom license amendment to adopt Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, "Risk-Informed Categorization and Treatment of Structures, System and Components for Nuclear Power Plants." The staff's review of the internal events PRA that supported this license amendment can be found in a safety evaluation dated October 25, 2018 (ADAMS Accession No. ML18263A232). The safety evaluation dated October 25, 2018, identified a few commitments to update the internal events PRA before implementing the risk-informed categorization process. As part of the audit, the NRC staff requested information about the impact of modelling updates to the internal events PRA that appeared to NRC staff to have the potential to impact the SPRA model results. The staff's review of the results of a sensitivity study performed by the licensee that incorporated those updates concluded that the updates would not change the conclusions of the SPRA submittal.

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

Finding F&O 1-5 cites concern about eliminating failure modes for cases where fragilities for different failure modes of equipment that are "close together". The disposition of the F&O states that the fragilities for different failure modes for components evaluated using SOV were either "not closely spaced" or were correlated. The resolution suggested by the peer review was to define and justify the term "close together" that was used as a criterion for eliminating failure modes. During the audit review, the licensee explained that if the difference between the fragilities for the two failure modes is greater than 20% then using only the dominant failure modes in the SPRA produces essentially the same results as including both failure modes. Accordingly, the licensee revisited all its SOV calculations in light of this criteria. It identified only two SOV calculations which contained fragilities for failure modes that were less than 20% apart, but in both cases the failure modes were determined to be correlated. In all other SOV calculations, the difference between the fragilities of different failure modes was over 20% so only the dominant failure was modelled. The NRC staff concluded that the licensee's disposition is sufficient for this submittal because the approach for determining whether failure modes are "close together" is consistent with the state-of-practice and the licensee reviewed applicable calculations.

The disposition to two F&Os (F&O 1-1 and F&O 1-2) presented in the submittal state that modeling was added to the SPRA to credit alignment of FLEX generators to Unit 2

and 3 load centers and to credit alignment of diesel-power FLEX pumps to reactor pressure vessel (RPV) make-up. The submittal does not describe this major update in the modeling though this modeling appears to impact significant accident sequences and therefore could be considered a PRA upgrade requiring a focused-scope peer review. (Failure of operators to align FLEX diesel generators was identified in the submittal as a dominant failure). Furthermore, no sensitivity study addressing this modelling uncertainty was presented in the submittal. However, as part of the audit the licensee provided the results of a sensitivity study of FLEX modeling. The results of the sensitivity study indicate that not crediting FLEX leads in an increase of about 5% in the SCDF and SLERF for each unit. Based on this sensitivity study, the NRC staff concludes that no further information is needed, given that credit for FLEX modeling in the SPRA will not change the conclusions of the submittal. Consequence(s): N/A	
The NRC staff concludes that:	
• The licensee's peer-review process meets the intent of the SPID guidance.	Yes
• The licensee's peer-review process does not meet the intent of the SPID guidance 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:

The SPRA submittal follows the recommendations of Section 6.8 of the SPID. Tables 2-1 and 2-2 of the submittal provide a cross-reference of information required by the 50.54(f) letter and specified in Section 6.8 of the SPID to the sections of the submittal where the information can be found. The level-of-detail of the information provided appears to be generally consistent with that specified in Section 6.8 of the SPID. It is noted, however, that not all the information identified in Section 6.8 of the SPID (with regard to what was submitted for the Individual Plant Examination of External Events (IPEEE) program) is included in the submittal (e.g., all functional/systemic event trees). However, the SPID only identifies this IPEEE information as guidance for consideration in the 50.54(f) response.

There were no F&Os related to SPRA documentation (e.g., HLR-SHA-J, HLR-SPR-G, and HLR-SFR-F) with the exception of F&O 6-8 concerning SRs SPR-F1 and SPR-F2 which were resolved by the licensee by updating the SPRA documentation to include the information cited as missing or incomplete (see Topic #14).

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

Section 6.8 of the SPID, states that SPRA submittals should provide the level of detail needed to determine the validity of the SPRA models "to assess the sensitivity of the results to all key aspects of the analysis to make necessary decisions as part of NTTF Phase 2 activities."

In regard to the sensitivity of the SPRA results to inputs, the NRC's safety evaluation of Peach Bottom's request to adopt risk-informed categorization dated November 26, 2018, states that Peach Bottom committed to update the PRA model to account for the need for two Emergency Diesel Generator (EDG) cooling fans during periods when the outdoor temperature at the Peach Bottom are above the design temperature of 80 °F prior to implementation of their risk-informed program. The NRC staff notes that a seismic event results in the likely loss of offsite power which increases the importance of EDGs and associated cooling fan success which can have non-negligible contribution at low seismic accelerations. Also, in the NRC's safety evaluation of Peach Bottom's request to adopt risk-informed categorization, it states that Peach Bottom committed to removing credit for core melt arrest in-vessel at high RPV pressure conditions. It is not clear to the NRC staff whether this update has been performed or whether it can impact the SPRA results. During the audit, the licensee explained that the updated modelling committed to for adoption of the 10 CFR 50.69 risk categorization was not incorporated into the SPRA. However, the licensee provided the results of a sensitivity study which incorporated the committed updates. The EDG cooling fan success criteria were revised to account for ambient outdoor temperatures greater than 80 °F and credit for the core melt arrest in-vessel at high RPV pressure was removed. Based on the sensitivity case SPRA, the importance values for Unit 3 were recalculated and presented. The results of the sensitivity study show that even though certain importance values increased slightly, the SPRA importance values results, in general, did not change significantly. Refer to Enclosure 2 for detailed evaluation.

The sensitivity study results presented in Table 5.7-3 of the submittal appear to show significant sensitivity to truncation limits for seismic hazard initiating event bins referred to as %G6 and %G7. The ASME/ANS PRA Standard, as endorsed by RG 1.200, Revision 2 provides criteria for demonstrating truncation convergence (i.e., the change in CDF or LERF should be less than 5% for a decade change in truncation limit). It appeared to the NRC staff that sensitivity to the truncation limit could impact the staff's decision (i.e., identifying potential cost-justified substantial safety improvements using importance measures). During the audit, the licensee explained that truncation test results presented in Table 5.7-3 of the submittal were based on the change in the SLERF for the hazard interval rather than the change in the total SLERF for sequences associated with the hazard interval. The licensee presented a revised table showing the impact of decreasing the truncation limit on the total overall SLERF that clearly shows that the impact is less than 5% for all hazard bins when the truncation limit is lowered an addition decade. This is consistent with the suggested criteria in Supporting Requirement QU-B3 of the ASME/ANS PRA standard.

Consequence(s): N/A

The NRC staff concludes that:	
• The licensee's documentation meets the intent of the SPID guidance. The documentation requirements in the ASME/ANS Standard can be found in HLR-SHA-J, HLR-SFR-G, 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 No No No
Plant will: • complete modifications by: • report completion of modifications by:	 N/A N/A
Notes from the Reviewer: Section 6.0 of the Peach Bottom SPRA submittal states that the SPRA reflects the as-built, as-operated plant as of the February 2018 "freeze date." The submittal states that there are no significant plant changes that are not included in the model which would have an adverse impact on the results. The submittal concludes that, based on the insights from the SPRA results, no seismic hazard vulnerabilities were identified requiring plant actions (i.e., modifications). Refer to Enclosure 2 for detailed evaluation. Deviation(s) or Deficiency(ies), and Resolution: Sensitivity study Case 1d results presented in Table 5.7-1 of the submittal shows significant SLERF sensitivity (i.e., 16%) to refinement in hazard event bin %G8 was large in comparison to other bins. Section 5.3.2 of the SPRA submittal states that human error probabilities (HEPs) associated with FLEX actions were not set to 1.0 in bin %G8 as they were for the other bins, though FLEX is more likely to fail at higher acceleration hazard events. During the audit, the licensee provided the results of a combined sensitivity study on SLERF for Unit 2 and 3 in which bin %G8 (the highest acceleration bin and much wider than other bins) was refined and credit for FLEX was removed. Hazard bin %G8 was refined by dividing it into six hazard bins. The results of the sensitivity study show that although some importance values increased, and others decreased, the results do not change the conclusions of the submittal. No cost-justified substantial safety enhancements related to seismically-induced failures or operator errors or combination thereof were identified from the results of the sensitivity. Refer to Enclosure 2 for detailed evaluation. Consequences: N/A	

The NRC staff concludes that: • The licensee identified plant modifications necessary to achieve the appropriate risk profile.	No
• The licensee provided a schedule to implement the modifications (if any) with appropriate consideration of plant risk and outage scheduling.	No

REFERENCES

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

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

EPRI-SPID, 2012: "Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," Electric Power Research Institute, EPRI report 1025287, November 2012, ADAMS Accession No. ML12333A170

NEI, 2012: NEI 12-13 "External Hazards PRA Peer Review Process Guidelines," Nuclear Energy Institute, August 2012, ADAMS Accession No. ML12240A027

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

NRC Staff SPRA Submittal Detailed Screening Evaluation

Introduction

The Peach Bottom Atomic Power Station, Units 2 and 3 (Peach Bottom) Seismic Probabilistic Risk Assessment (SPRA) submittal (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18240A065) indicates that the point estimate of the seismic core damage frequency (SCDF) is 2.1×10^{-5} per reactor-year (/rx-yr) for Units 2 and 3 and the point estimate of the seismic large early release frequency (SLERF) is 4.0×10^{-6} /rx-yr for Unit 2 and 4.1×10^{-6} /rx-yr for Unit 3. The mean CDF and LERF values were not provided in the SPRA submittal but the 5%, 50%, and 95% values were provided. The staff estimated the mean SCDF and mean SLERF for each unit based on the information in the submittal and confirmed the same during the audit. The NRC staff compared these values against the guidance in NRC staff memorandum dated August 29, 2017, titled, "Guidance for Determination of Appropriate Regulatory Action Based on Seismic Probabilistic Risk Assessment Submittals in Response to Near Term Task Force Recommendation 2.1: Seismic" (ADAMS Accession No. ML17146A200; hereafter referred to as 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 (1.0×10^{-6} /rx-yr), or 0.1 times the threshold for the screening criterion for SCDF (1.0×10^{-5} /rx-yr).

The NRC staff found that because the SCDF and SLERF for Peach Bottom were above the initial screening values of 1.0×10^{-5}/rx-yr and 1.0×10^{-6}/rx-yr, respectively, a detailed screening following the SPRA Screening Guidance was performed. The detailed screening concluded that Peach Bottom should be considered a Group 1 plant because:

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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).

- Sufficient reductions in SCDF and/or 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, 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.

The licensee, in performing its seismic analysis in response to the Near-Term Task Force Recommendation 2.1, and the NRC in conducting its review, did not identify concerns that would require licensee action above and beyond existing regulations to maintain the level of protection necessary to avoid undue risk to public health and safety. In addition, there were no issues identified as non-compliances with the Peach Bottom licenses, or with the rules and orders of the Commission. For these reasons, the licensee and the staff did not identify a potential modification necessary for adequate protection or compliance with existing regulations.

Detailed Screening

The detailed screening uses information provided in the Peach Bottom SPRA submittal, particularly the importance measures, SCDF, and SLERF, as well as other information described below, to establish threshold and target values to identify potential cost-justified substantial safety improvements. The detailed screening process makes several simplifying assumptions, similar to a Phase 1 Severe Accident Mitigation Alternatives (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 Peach Bottom SPRA submittal provides Fussell-Vesely (F-V) importance measures, which were converted to RRW values by the NRC staff for this screening evaluation using an established mathematical relationship (included in the SPRA Screening Guidance).

Data used to develop the maximum averted cost-risk (MACR) for the severe accident mitigation alternative (SAMA) analysis provided in the *Generic Environmental Impact Statement for License Renewal of Nuclear Power Plants Regarding Peach Bottom Atomic Power Station Units 2 and 3*, NUREG-1437, Supplement 10, dated January 2003 (ADAMS Accession Nos. ML030270026, ML030270038, and ML030270065), was used to calculate the RRW threshold. For this analysis, the NRC staff determined the RRW threshold from the SCDF-based MACR to be 1.056 for both Units. 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 corresponds to reduction in SCDF of $1.0x10^{-5}$ /rx-yr or reduction in SLERF of $1.0x10^{-6}$ /rx-yr. The target RRWs based on the mean and 95th percentile SCDF and SLERF were also calculated by the NRC staff and ranged between 1.63 and 1.96 for both units.

Section 5 of the Peach Bottom SPRA submittal included tables listing and describing the seismic structures, systems, and components (SSCs) failures that are the most significant contributors to SCDF and SLERF. Similar tables were also provided for the most significant contributors due to failure of operator actions. The descriptions of the significant contributors included the corresponding F-V importance measures. The NRC staff utilized the F-V values to calculate the RRW and the maximum risk reduction achievable by eliminating the failure. The results for both units are provided in Table 1 for the SCDF contributors and Table 2 for the SLERF contributors that have an RRW greater than about 1.005. These tables provide the following information by column: (1) Description of the component, (2) Failure mode of the component, if applicable, (3) RRW, and (4) maximum SCDF reduction (MCR) or SLERF reduction (MLR) from eliminating that failure.

A single SPRA seismic failure exceeded the target RRW for SCDF and two contributors exceeded the target RRW for SLERF for both units. The common contributor for both SCDF and SLERF was the seismically-induced loss of offsite power (OSP), which has an SCDF RRW of 52.6 and a maximum SCDF reduction potential of $2.6x10^{-5}$ /rx-yr for both units. According to Section 5 of the SPRA submittal, OSP is a contributor for all the top ten accident sequences for SCDF and SLERF. During the audit, the licensee explained that a representative fragility was used for modeling OSP that included the contribution of seismic-induced failure modes in the switch yard as well as seismic-induced failures outside the plant's boundary such as transmission line failure. The NRC staff notes that improvements only in the switch yard will likely not yield the target risk reduction. Also, the licensee stated that installation of a seismically-qualified power source in the plant switch yard to provide offsite power or hardening the existing offsite power supply would clearly exceed the maximum monetary value by a large amount and therefore would not be cost-justified. As a result, the NRC staff did not pursue potential improvements to OSP.

The second contributor that exceeded the RRW threshold for SLERF was structural failure of Reactor Pressure Vessel (RPV) internals (SCRAM). The NRC staff concludes that the cost of a plant modification to strengthen the RPV internals would far exceed the maximum monetary value. As a result, the NRC staff did not pursue potential improvements to RPV internals.

A few combinations of two failures would also exceed the target RRW for SCDF and SLERF. However, all but one of those combinations included one of the two failures discussed above and therefore, were not pursued further. For SLERF, the combination of eliminating operator failure to manually start Reactor Core Isolation Cooling (RCIC) (RHUBLKSTDXI3) with operator failure to valve-in the nitrogen bottle early or late (AHUBTL-RDXI3 or AHUBTL-RDXD3) would result in a SLERF reduction of $1.17x10^{-6}$ /rx-yr for Unit 3. However, the NRC staff's review of the submittal determined that these combinations would not result in substantial safety enhancements because high degree of uncertainty exists for the feasibility of such actions at higher seismic accelerations where such actions are currently not credited and that plant operational changes (e.g., procedure changes) cannot achieve all of the risk reduction reflected by the importance measures (i.e., make operator actions always successful). Further, the NRC staff concludes that physical plant modifications that would eliminate the need for the operator actions cited above would exceed the maximum monetary value.

To account for internal event PRA modeling updates that were part of the implementation items supporting the NRC staff's approval of the licensee's request to adopt risk-informed categorization of SSCs (ADAMS Accession No. ML18263A232), the licensee provided the results of a sensitivity study in which the Emergency Diesel Generator (EDG) cooling fan success criteria was revised for ambient outdoor temperatures greater than 80 degrees

Fahrenheit (°F) and credit for the core melt arrest in-vessel at high RPV pressure was removed in the sensitivity case. Based on the sensitivity case, the importance values for Unit 3 were recalculated and presented. The results of the sensitivity study show that some importance values increased slightly and that the importance measure for S-DGFN2- (EDG Supplemental Supply Fan 0(A—D)V91) increased enough to be identified with the list of important risk contributors, but, in general, the importance values did not change significantly.

The licensee provided the results of an aggregate sensitivity study on SLERF for Unit 2 and 3 in which bin %G8 (the highest acceleration bin and much wider than other bins) was refined by dividing that one bin into six hazard bins and removing credit for FLEX. The results of the sensitivity study show that although the importance values for some failures increased, the results did not change significantly. The sensitivity study showed that if operator error RHUBLKSTDXI2 (Operator fails to manually start RCIC) and EHURLY4KDXI2 (Operator fails to mitigate relay chatter for 4KV buses) were eliminated, then a SLERF reduction of 1.01×10^{-6} /rx-yr SLERF could be achieved for Unit 2. The NRC staff concluded that the combination of the above operator actions did not appreciably exceed the threshold and that additional evaluation would result in the substantial safety enhancement threshold not being met because that plant operational changes (e.g., procedure changes) would not achieve all the risk reduction reflected by the importance measures (i.e., make operator actions always successful).

Based on the information presented in the submittal, the NRC staff noted that a basic event titled "LERF Not Precluded Due to SORVs / Timing," had a high importance measure for SLERF. The submittal stated that the basic event "modeled phenomenological issues associated with the Level 2 accident progression resulting in a LERF end state." The discussion of sensitivity case 2a in Section 5.7 of the submittal provides details about the basic event which is related to the likelihood of a stuck open relief valve (SORV) leading to a LERF for so-called short-term station black out (STSBO) scenarios. The discussion cites NUREG/CR-7110, "State-of-the-Art Reactor Consequence Analyses Project, Volume 1: Peach Bottom Integrated Analysis" (ADAMS Accession No. ML120260675). The discussion in Section 5.7 of the submittal indicates that the SPRA model used a value of 15% for the conditional probability for SLERF for unmitigated STSBO sequences.

The NRC staff recognized that the value of 15% was introduced after the peer-review and that use of a value appreciably different from 15% could result in the modification to the anchorage of the DC battery racks (to increase their capacity) to be considered as a potential substantial safety enhancement. Therefore, the staff evaluated impact of the conditional probability for SLERF for unmitigated STSBO sequences further. Based on (1) the NRC staff's evaluation of the SOARCA results, (2) the relatively low impact of the DC battery rack anchorage improvement on the core damage, and (3) the diminishing impact of the DC rack anchorage improvement on containment performance for conditional probability of SLERF appreciably lower than 100%, the NRC staff determined that pursuing the DC battery rack anchorage improvement as a potential modification in the context of this review (i.e., response to the 10 CFR 50.54(f) letter and determination of potential backfits under 10 CFR 50.109) was not justified. The staff notes that anchorage of the DC battery racks is an important risk insight derived from the SPRA related to the plant risk impact. The staff reiterates that this review is limited to the context of the 10 CFR 50.54(f) response associated with NTTF Recommendation 2.1 "Seismic". 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.

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 enhancement or would warrant further regulatory analysis.

Additionally, the NRC staff reviewed the results of the licensee's Individual Plant Examination of External Events (IPEEE) and previous SAMA analyses to identify additional substantial safety improvements that would be cost justified. No other potential substantial safety enhancements were identified based on that review.

Conclusion

Based on the analysis of the submittal and supplemental information, the NRC staff concludes that no modifications are warranted under 10 CFR 50.109 because:

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

- no 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 enhancement.

Table 1. Importance Analysis Results of Top Contributors to Unit 2 and 3 SCDF

Fragility Group/Event	Description	Failure Mode	Unit 2		Unit 3	
			RRW	MCR (/rx-yr)	RRW	MCR (/rx-yr)
SSC Fragility Groups – Seismically Failed						
OSP	Offsite Power	Functional	52.632	2.63E-05	52.632	2.63E-05
S-DCBT1-	DC Batteries 2(A-D)D01, 3(A-D)D01	Anchorage	1.136	3.22E-06	1.135	3.19E-06
S-CNWG2-	Conowingo Hydroelectric Plant (OSP)	Functional	1.046	1.18E-06	1.056	1.41E-06
S-CEP1-	Panel 20C003, 20C004C, 30c003, 30C004C, 00C29(A-D)	Anchorage	1.040	1.02E-06	1.039	1.01E-06
S-CC359A-	Correlated Relay Chatter Group 359A (52B-TD5 relays) (All EDGs - Unrecoverable)	Functional	1.011	2.87E-07	1.012	3.06E-07
S-DCBS4-	DC Panel 20D24, 30D21	Anchorage	NA	NA	1.010	2.71E-07
S-DGPA1	D/G Room Supply Temp Control Panel 0(A-D)C479	Functional	1.004	1.01E-07	1.007	1.95E-07
*Significant Operator Errors**						
AHUBTL-RDXI2 AHUBTL-RDXI3	Operator fails to valve-in N2 Bottles after accumulator depletion (early)		1.029	7.50E-07	1.023	6.14E-07
AHU-CADDXI2 AHU-CADDXI3	Operator fails to align Cad Tank to Unit 2/3 ins 'B'		1.027	7.13E-07	1.023	5.98E-07
AHUBTL-RDXD2 AHUBTL-RDXD3	Ops fail to valve-in N2 bottles after accumulator depletion (late; conditional)		1.027	7.05E-07	1.021	5.47E-07
AHU-CADDXD2 AHU-CADDXD3	Operator fails to align Cad Tank to Unit 2/3 ins 'B' - delayed, conditional		1.025	6.62E-07	1.019	5.07E-07
QHUFXL13DXI2 QHUFXL13DXI3	Operator fails to align FLEX generator to LC E124 or E324		1.019	4.98E-07	1.016	4.31E-07
EHURLY4KDXI2 EHURLY4KDXI3	Operator fails to mitigate rely chatter for 4kV buses (seismic)		1.016	4.31E-07	1.015	3.83E-07
QHULS-ACDXI2 QHULS-ACDXI3	Operator fails to perform deep DC load shed		1.013	3.54E-07	1.012	3.08E-07
EHU-SE11DXI0	Operator fails to cross-tie 4kV Emergency buses		1.007	1.86E-07	1.010	2.66E-07
KHUDGFANDXI0	Operator fails to manually initiate supplemental fan		1.007	1.79E-07	NA	NA

* *Operator action basic events with two entries identify the same operator action modeled separately for Units 2 and 3.*

Table 2. Importance Analysis Results of Top Contributors to Unit 2 and 3 SLERF

Fragility Group/Event	Description	Failure Mode	Unit 2 RRW	Unit 2 MLR (/rx-yr)	Unit 3 RRW	Unit 3 MLR (/rx-yr)
SSC Fragility Groups – Seismically Failed						
OSP	Offsite Power	Functional	10.204	6.62E-06	10.417	6.64E-06
SCRAM	RPV Internals (Scram)	Anchorage	1.252	1.48E-06	1.253	1.48E-06
S-DCBT1-	DC Batteries 2(A-D)D01, 3(A-D)D01	Anchorage	1.144	9.25E-07	1.114	7.49E-07
S-CNWG2-	Conowingo Hydroelectric Plant (OSP)	Functional	1.054	3.75E-07	1.052	2.75E-07
BOC	Break Outside Containment	Anchorage	1.040	2.84E-07	1.039	2.10E-07
SML	Seismic Induced Medium LOCA	Anchorage	1.032	2.29E-07	1.031	1.84E-07
S-CEPA1-	Panel 20C003, 20C004C, 30c003, 30C004C, 00C29(A-D)	Anchorage	1.027	1.95E-07	1.055	3.61E-07
S-DCBS4	DC Panel 20D24, 30D21	Anchorage	NA	NA	1.026	1.74E-07
S-PCI2	Primary Containment Isolation (Inboard and Outboard MSIVs)	Functional	1.023	1.66E-07	1.024	1.09E-08
S-CEPA7-	Panel 20C32 (U2 Engineering Sub Systems I Relay Cabinet)	Functional	1.014	1.04E-07	NA	NA
S-CNCT1-	Condensate Storage Tank 20T010, 30E010	Anchorage	1.014	1.01E-07	1.015	1.01E-07
S-DCBS10	250 VDC Bus 30D11	Anchorage	NA	NA	1.014	7.85E-08
S-SGTK1-	SGIG Nitrogen Tank	Anchorage	1.012	8.51E-08	1.008	6.03E-08
S-CEPA6-	Panel 20C32 (U2 HPCI Relay Panel)	Functional	1.012	8.44E-08	NA	NA
S-CC190A-	Correlated Relay Chatter Group 190A (52B-151N relays)(EDGs A and D - Unrecoverable)	Functional	1.009	6.74E-08	1.000	0
S-CEPA8-	Panel 20C33 (U2 Engineering Sub Systems II Relay Cabinet)	Functional	1.008	5.60E-08	NA	NA
S-CC138-	Relay Chatter Group 138 (150G relay) (4KV Bus 20A15 - Recoverable)	Functional	1.007	5.29E-08	NA	NA
S-DCBS6-	DC Panel 2(A-D)D17, 3AD17, 3CD17. 3DD17	Functional	1.006	4.55E-08	NA	NA

Fragility Group/Event	Description	Failure Mode	Unit 2		Unit 3	
			RRW	MLR (/rx-yr)	RRW	MLR (/rx-yr)
*Significant Operator Errors**						
RHUBLKSTDXI2 RHUBLKSTDXI3	Operator fails to manually start RCIC (Black start) - seismic PRA		1.058	4.01E-07	1.106	7.02E-07
EHURLY4KDXI2 EHURLY4KDXI3	Operator fails to mitigate relay chatter for 4kV buses (seismic)		1.031	2.23E-07	1.019	1.34E-07
EHU-SE11DXI0	Operator fails to cross-tie 4kV Emergency buses		1.028	1.99E-07	1.022	1.56E-07
AHU-CADDXI2 AHU-CADDXI3	Operator fails to align Cad Tank to Unit 2/3 ins 'B'		1.024	1.73E-07	1.035	2.45E-07
QHUFXL13DXI2 QHUFXL13DXI3	Operator fails to align FLEX generator to LC E124 or E324		1.024	1.72E-07	1.014	9.84E-08
AHU-CADDXD2 AHU-CADDXD3	Ops fail to align Cad Tank to Unit 2/3 ins 'B' - delayed, conditional		1.022	1.59E-07	1.033	2.31E-07
AHUBTL-RDXI2 AHUBTL-RDXI3	Op fails to valve-in N2 bottles after accumulator depletion (early)		1.022	1.58E-07	1.068	4.70E-07
AHUBTL-RDXD2 AHUBTL-RDXD3	Ops fail to valve-in N2 bottles after accumulator depletion (late; conditional)		1.020	1.45E-07	1.060	4.18E-07
QHULS-ACDXI2 QHULS-ACDXI3	Operator fails to perform deep DC load shed		1.016	1.15E-07	1.009	6.77E-08
2CZOP-SLCLWL-H- 3CZOP-SLCLWL-H-	Operator fails to inject SLC with boron on low water level		1.014	9.98E-08	1.016	1.15E-07
RHUCSTSPDXI2 RHUCSTSPDXI3	Ops fail to swap RCIC shutdown suction from CST to Suppress Pool		1.014	9.98E-08	1.015	1.08E-07
EHULS-ACDXI2 EHULS-ACDXI3	Ops fail to perform SE-11 load shed for FLEX (single unit-RCIC)		1.011	7.08E-08	1.011	8.07E-08
EHUATT-TDXI0	Ops fails to perform SE-11 load shed for FLEX (single unit, both divisions)		1.010	7.08E-08	NA	NA
EHURLYDGDXI2 EHURLYDGDXI3	Operator fails to mitigate relay chatter for EDGs (seismic)		1.009	6.74E-08	1.010	7.05E-08

** Operator action basic events with two entries identify the same operator action modeled separately for Units 2 and 3.*

AUDIT SUMMARY BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 SUBMITTAL OF SEISMIC PROBABILISTIC RISK ASSESSMENT ASSOCIATED WITH REEVALUATED SEISMIC HAZARD IMPLEMENTATION OF THE NEAR-TERM TASK FORCE RECOMMENDATION 2.1: SEISMIC (EPID NO. L-2018-JLD-0010)

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 Title 10 CFR Part 50, Section 50.54(f). The list of applicable licensees in Enclosure 1 to the July 6, 2017, letter included Exelon Generation Company, LLC (Exelon, the licensee) as the licensee for Peach Bottom Atomic Power Station, Units 2 and 3 (Peach Bottom).

REGULATORY AUDIT SCOPE AND METHODOLOGY

The areas of focus for the regulatory audit are the information contained in the SPRA submittal and all associated and relevant supporting documentation used in the development of the SPRA submittal including, but not limited to, methodology, process information, calculations, computer models, etc.

AUDIT ACTIVITIES

The NRC staff developed questions to verify information in the licensee's submittal and to gain understanding of non-docketed information that supports the docketed SPRA submittal. The staff's clarification questions dated, February 6, 2019, and February 11, 2019 (ADAMS

Accession Nos. ML19037A483, and ML19044A356, respectively), were sent to the licensee to support the audit.

The licensee provided clarifying information in the following areas:

- Discussion of commitments made by Peach Bottom as part of their request to adopt risk-informed categorization to update the EDG cooling fan success criteria and remove credit for core melt arrest in-vessel at high RPV pressure conditions in the PRA models.
- Discussion of the definition of the term "not closely spaced" used as the basis for not correlating SOV determined fragilities of different component failure modes.
- Discussion of the technical basis and justification for the significant changes made in the SPRA model reflected in "Quantification 5" which changed the dominant risk contributors and the corresponding importance measures.
- Discussion of the sensitivity of SPRA results to truncation limits for seismic hazard initiating event bins %G6 and %G7.
- Discussion of the sensitivity of SPRA results to how the interval for the highest seismic hazard initiating event bin was defined in combination with uncertainty about the feasibility of FLEX operator actions.
- Discussion of whether the event OSP included failures whose frequencies could be reduced using plant modifications.
- Discussion of structural fragility provided for the Conowingo Dam.

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

DOCUMENTS AUDITED

- Plant Document PB-ASM-04, "Peach Bottom Atomic Power Station Probabilistic Risk Assessment – Application Specific Model (ASM) Notebook," January 2015.
- Plant Document PB-ASM-06, "Peach Bottom Atomic Power Station Probabilistic Risk Assessment – Application Specific Model (ASM) Notebook," November 2016.
- Plant Document PB-ASM-13, "Application Specific Model Notebook," May 2018.
- Plant Document PB-PRA-20.006, Rev. 0, "Peach Bottom Seismic Probabilistic Risk Assessment – Seismic Quantification Notebook," August 2018
- File: "NRC Info Request 3 Item 2 FLEX FPIE PRA Info_12-07-18.docx" – Excerpts from the internal events notebook related to FLEX modeling

- ENERCON Report EXLNPB081-REPT-014, Revision 0, Attachment 5, "Bounding Estimation in the Seismic Fragility of the Conowingo Dam"
- Plant Document PB-PRA-20.005, Volume 1, Rev. 2, "Peach Bottom Seismic Probabilistic Risk Assessment – Fragility Modeling Notebook," August 2018.
- Sections of ENERCON Report EXLNPB081-REPT-013, Revision 1, "Peach Bottom Atomic Power Station, Seismic Probabilistic Risk Assessment Project Fragility Analysis Main Report"

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 July 6, 2017, generic audit plan.

AUDIT CONCLUSION

The issuance of this document, containing the staff's review of the SPRA submittal, concludes the SPRA audit process for Peach Bottom.

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – STAFF REVIEW OF SEISMIC PROBABILISTIC RISK ASSESSMENT ASSOCIATED WITH REEVALUATED SEISMIC HAZARD IMPLEMENTATION OF THE NEAR-TERM TASK FORCE RECOMMENDATION 2.1: SEISMIC (EPID NO. L-2018-JLD-0010) DATE: JUNE 10, 2019

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