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

April 28, 2020

Mr. Bradley J. Sawatzke
Chief Executive Officer
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MD 1023
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P.O. Box 968
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SUBJECT: COLUMBIA GENERATING STATION – STAFF REVIEW OF SEISMIC PROBABILISTIC RISK ASSESSMENT ASSOCIATED WITH REEVALUATED SEISMIC HAZARD IMPLEMENTATION OF THE NEAR-TERM TASK FORCE RECOMMENDATION 2.1: SEISMIC (EPID NO. L-2019-JLD-0009)

Dear Mr. Sawatzke:

The purpose of this letter is to document the staff's evaluation of the Columbia Generating Station (Columbia), 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 Columbia.

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 September 26, 2019 (ADAMS Accession No. ML19273A907), Energy Northwest (the licensee), provided its SPRA submittal in response to Enclosure 1, Item (8) of the 50.54(f) letter for Columbia. As applicable, the NRC staff assessed the licensee's implementation of the Electric Power Research Institute's Report 1025287, "Seismic Evaluation Guidance - Screening, Prioritization, and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic" (ADAMS Accession No. ML12333A170). This report was endorsed by the NRC by letter dated February 15, 2013 (ADAMS Accession No. ML12319A074). In addition, consistent with the licensee's submittal, the NRC staff utilized a reviewer checklist that is based on American Society of Mechanical Engineers (ASME) /American Nuclear Society (ANS) (RA-S Case 1 "Case for ASME/ANS Ra-Sb-2013, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (herein called the "Code Case Standard"). Use of this reviewer checklist for licensees choosing to use the Code Case Standard was described in a letter to the Nuclear Energy Institute (NEI) dated July 12, 2018 (ADAMS Accession No. ML18173A017).

The reviewer checklist for the Columbia SPRA assessment is contained in Enclosure 1 to this letter. As described below, the NRC has concluded that the Columbia 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 12, 2015 (ADAMS Accession No. ML15078A243), Energy Northwest submitted the reevaluated seismic hazard information for Columbia. The NRC performed a staff assessment of the submittal and issued a response letter on November 4, 2016 (ADAMS Accession No. ML16285A410). The NRC's assessment concluded that Energy Northwest conducted the hazard reevaluation using present-day regulatory guidance and methodologies, appropriately characterized the site, and met the intent of the guidance for determining the reevaluated seismic hazard at Columbia.

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, Columbia was expected to complete an SPRA with an estimated completion date of March 31, 2019, which would also assess high frequency ground motion effects. By letter dated September 6, 2018 (ADAMS Accession No. ML18249A360), the licensee requested to extend the SPRA submittal to September 30, 2019. The staff responded in a letter dated November 20, 2018 (ADAMS Accession No. ML18291A556). In addition, Energy Northwest was expected to perform a limited-scope evaluation for the spent fuel pool (SFP). This SFP limited-scope evaluation was submitted by letter dated December 28, 2017 (ADAMS Accession No. ML18002A424). The staff provided its assessment of the Columbia SFP evaluation by letter dated April 17, 2018 (ADAMS Accession No. ML18106B119).

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

In its letter dated September 26, 2019, Energy Northwest provided the SPRA submittal that initiated the NRC's Phase 2 decisionmaking process for Columbia. The NRC described this

Phase 2 decision making process in a guidance memorandum from the Director of the Division of Operating Reactor Licensing to the Director of the Office of Nuclear Reactor Regulation (NRR) dated March 2, 2020 (ADAMS Accession No. ML20043D958). This memorandum describes a Senior Management Review Panel (SMRP) consisting of NRR Division Directors that are expected to reach a screening decision for each plant submitting an SPRA. The SMRP is supported by appropriate technical staff who are responsible for consolidating relevant information and developing the recommendation for the screening decisions for consideration by the panel. In presenting recommendations to the SMRP, the supporting technical staff is expected to recommend placement of each SPRA plant into one of three groups:

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

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

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

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

EVALUATION

Upon receipt of the licensee's SPRA submittal, a technical team of NRC staff members performed a completeness review to determine if the necessary information to support Phase 2 decisionmaking had been included in the licensee's submittal. The technical team performing the review consisted of staff experts in the fields of seismic hazards, fragilities evaluations, and plant response/risk analysis. On November 1, 2019 (ADAMS Accession No. ML19305C934), the technical team determined that sufficient information was available to perform the detailed technical review in support of the Phase 2 decisionmaking.

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

Appendix A of the licensee's submittal also included the open SPRA finding level facts and observations (F&Os) along with the licensee's dispositions. These elements were reviewed by NRC staff in the context of the regulatory decisionmaking associated with the 50.54(f) letter.

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 Energy Northwest as the licensee for Columbia site. The staff exercised the audit process by reviewing selected licensee documents via an electronic reading room (eportal) as documented in Enclosure 3 to this letter.

During the audit process, the staff developed questions to clarify information in the licensee's submittal and to gain understanding of non-docketed information that supports the docketed SPRA submittal. The staff's clarification questions and request for supporting documents dated January 10, 2020, and November 1, 2019 (ADAMS Accession Nos. ML20013G764 and ML19305C934, respectively), were sent to the licensee to support the audit. The licensee subsequently provided those supporting documents and answers to the audit questions on the eportal, which the staff reviewed. The staff determined that the answers to the questions provided in the eportal served to confirm statements that the licensee made in its SPRA submittal and supplements.

Since the licensee's internal events PRA (IEPRA) model was used as the basis for the development of the SPRA model, the NRC staff reviewed the IEPRA F&Os and the associated dispositions during the SPRA audit process to assess any potential impact on the SPRA submittal. The NRC staff identified no issues with the licensee's dispositions to these findings with respect to the SPRA submittal.

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

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

Following the staff's conclusion on the SPRA's technical adequacy, the staff reviewed the risk and safety insights contained in the Columbia SPRA submittal. The staff also used the screening criteria described in a staff memorandum dated August 29, 2017 (ADAMS Accession No. ML17146A200), titled, "Guidance for Determination of Appropriate Regulatory Action Based on Seismic Probabilistic Risk Assessment Submittals in Response to Near Term Task Force Recommendation 2.1: Seismic" to assist in determining the group in which the technical team would recommend placing Columbia to the SMRP. The criteria in the staff's guidance document includes thresholds to assist in determining whether to apply the backfit screening process described in Management Directive 8.4, "Management of Facility Specific Backfitting, Forward Fitting, Issue Finality, and Information Requests," dated September 20, 2019 (ADAMS Accession No. ML18093B087), to the SPRA submittal review. As part of this review, the staff considered potential modifications that could help identify substantial safety enhancements that could be cost-justified. Based on the SCDF and SLERF results, the NRC staff utilized the Columbia SPRA submittal and other available information in conjunction with the guidance in the staff memorandum dated August 29, 2017, to complete a detailed screening evaluation. The detailed screening concluded that Columbia should be considered a Group 1 plant because:

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

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

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

A discussion of the detailed screening evaluation completed by the NRC staff is provided in Enclosure 2 to this letter.

Based on the detailed screening evaluation and its review of the Columbia SPRA submittal, the technical team determined that recommending Columbia 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 Columbia review to the TRB with the Phase 2 recommendation that Columbia 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 Columbia as a Group 1 plant.

Subsequently, the technical review team consulted with the SMRP and presented the results of the review including the recommendation for Columbia 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 Columbia should be classified as a Group 1 plant, meaning that no further response or regulatory action is required.

AUDIT REPORT

The generic audit plan dated July 6, 2017, describes the NRC staff's intention to issue an audit report that summarizes and documents the NRC's regulatory audit of licensee's SPRA submittals associated with their reevaluated seismic hazard information.

The NRC staff's 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 Columbia 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 Milton Valentin at (301) 415-2864 or via e-mail at Milton.Valentin@nrc.gov.

Sincerely,

/RA/

Mohamed Shams, Deputy Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-397

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

cc w/encls: Listserv

NRC Staff SPRA Submittal Technical Review Checklist

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

- Topic 7: Use of New Response Analysis for Building Response, ISRS [In-Structure Response Spectra], and Fragilities
- Topic 8: Screening by Capacity to Select SSCs [Structures, Systems, and Components] for Seismic Fragility Analysis (SPID Section 6.4.3)
- Topic 9: Use of the CDFM [Conservation Deterministic Failure Margin]/Hybrid Methodology for Fragility Analysis (SPID Section 6.4.1)
- Topic 10: Capacities of SSCs Sensitive to High-Frequencies (SPID Section 6.4.2)
- Topic 11: Capacities of Relays Sensitive to High-Frequencies (SPID Section 6.4.2)
- Topic 12: Selection of Dominant Risk Contributors that Require Fragility Analysis Using the Separation of Variables Methodology (SPID Section 6.4.1)
- Topic 13: Evaluation of LERF [Large Early Release Frequency] (SPID Section 6.5.1)
- Topic 14: Peer Review of the SPRA, Accounting for NEI 12-13 (SPID Section 6.7)
- Topic 15: Documentation of the SPRA (SPID Section 6.8)
- Topic 16: Review of Plant Modifications and Licensee Actions

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

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

The licensee used the reevaluated seismic hazard for the Columbia Generating Station (CGS) site submitted March 12, 2015 (ADAMS Accession No. ML15078A243), and approved for use in the SPRA by the NRC staff (ADAMS Accession No. ML16285A410). However, peer review finding and observation (F&O) 20-10 pointed out that the approved reevaluated hazard needed to be justified as appropriately reflecting spectral shapes from a PSHA. To resolve this finding, the licensee (Energy Northwest, the licensee for CGS) revised the seismic hazard and reassessed the fragilities to determine the impacts of using the revised hazard in the SPRA. Topics 6 and 14 have additional notes on this resolution. Before the end of the audit review, the licensee informed the staff that F&O 20-10 was closed.</td></tr>
<tr><td>The NRC staff concludes that:
the peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to the SHA requirements in the Code Case Standard, as well as to the requirements in the SPID.
although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.
the guidance in the SPID was followed for developing the probabilistic seismic hazard for the site.
an alternate approach was used and is acceptable on a justified basis.</td><td>YES

N/A

YES

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

TOPIC 2: Site Seismic Response (SPID Section 2.4)

The site under review has updated/revised its site response analysis from what was submitted to NRC in response to the NTTF Recommendation 2.1: Seismic 50.54(f) letter.	NO
Notes from staff reviewer: None Deviation(s) or deficiency(ies) and Resolution: N/A Consequence(s): N/A	
The NRC staff concludes that:	
• the peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to all SRs under HLR-SHA-E in the Code Case Standard, as well as to the requirements in the SPID.	YES
• although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.	N/A
• the licensee's development of PSHA inputs and base rock hazard curves meets the intent of the SPID guidance or another acceptable approach.	YES
• the licensee's development of a site profile for use in the analysis adequately meets the intent of the SPID guidance or another acceptable approach.	YES
• although the licensee's development of a shear wave velocity (V_s) profile for use in the analysis does not meet the intent of the SPID guidance, it is acceptable on another justified basis.	N/A

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

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

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

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

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

Notes from staff reviewer:

Section 4.3 of the CGS SPRA submittal describes the analysis of structures which support the safety-related components and systems. Table 4-2 of the submittal provides a summary of the structural modeling and the analysis methods used for the Reactor Building (RB), Radwaste / Control Building (RWCB), Diesel Generator Building (DGB), and Turbine Building (TB). All the buildings, except the TB, contained structures, systems, and components (SSCs) in the Seismic Equipment List (SEL). The TB, however, posed concern for potential seismic interaction with adjacent RB and RWCB structures. The table also identifies structural analysis methods for the Condensate Storage Tanks (CST) and Service Building.

New finite element models were developed for the RB, RWCB, DGB, and TB because existing Lumped Mass Stick Models (LMSM) of these CGS structures did not meet the SPID modeling requirements. In addition to the load bearing internal and external structural components, the NRC staff's audit review confirmed that the RB structural finite element model includes representation of SEL SSCs listed in Section 4.1.1 of the SPRA submittal (e.g., primary containment vessel, drywell, biological shield wall), while the reactor pressure vessel and internals models that are connected to the finite element model were based on existing lumped mass stick model. The licensee stated in its submittal that fixed-base structural modal analyses at the reference earthquake was performed to assess potential cracking and the extent of concrete cracking, and to confirm the dynamic properties used in the models. Fixed-base analysis was not used to determine structural fragilities, except for the Service Building.

The SPRA submittal explains that CGS is a soil site requiring soil-structure interaction (SSI) analysis for determining building response and in-structure response spectra (ISRS) needed to determine SSC fragilities. Probabilistic SSI analyses were performed where variability of soil and structural stiffness properties and damping were sampled. The results from the probabilistic SSI analysis was used to develop median and 84 percentile Non-Exceedance Probability (NEP) in-structure response spectra and displacements, where the SEL systems and components are located, for fragility evaluation of the SSCs. The CGS structural analysis includes evaluation of potential impact between all the buildings separated by small gaps at appropriate floor elevations. The NRC staff's audit confirmed that the licensee addressed potential effects of soil-liquefaction, lateral spreading, and settlement at the site and precluded these hazards based on site-specific evaluation.

<table>
<tr><td colspan="2">New finite element modeling including probabilistic seismic response was also performed for CST, which is a flat bottom cylindrical steel tank anchored to the foundation. The steel tank and concrete foundation were modeled using a 3-D finite element model. The contained fluid was modeled using a stick model representative of the first horizontal and vertical modes of vibration that was coupled to the finite element model.

The NRC staff used the audit process to assess the structural modeling and response analyses and confirmed that 3D-finite element structural modeling is capable of capturing structural response, torsional effects resulting from eccentricities, and in-plane floor flexibility. The NRC staff's audit review indicates that appropriate modes of vibration of the structures were considered in the analysis and the modeling approach applied requirements of ASCE/SEI 4-16. Thus, NRC staff finds that SPID (Section 6.3.1) criteria 1 through 7 were met and that the licensee used realistic mathematical models to represent the three-dimensional dynamic characteristics of the building structures for seismic response calculations in accordance with ASME/ANS Code Case SFR-B3 requirements.

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

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

• The peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to the SR requirement SFR-B3 in the Code Case Standard, as well as to the requirements in the SPID.

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

• The licensee's structural model meets the intent of the SPID guidance.

• The licensee's structural model does not meet the intent of the SPID guidance, but is acceptable on another justified basis.</td><td>N/A

N/A

YES

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

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

Fixed-based dynamic seismic analysis of structures was used, for sites previously defined as "rock." If no, this issue is moot. If yes, on which structure(s)? Structure name: West Penetration Room (Unit 3)	NO
Structure #1: If used, is V_S > about 5,000 feet (ft.)/second (sec.)? Review of the Columbia SPRA report shows that the mean shear wave velocity of the rock in the area where the West Penetration Room is located is greater than 5,000 ft./sec. If 3,500 ft./sec. < V_S < 5,000, was peak-broadening or peak shifting used?	N/A
Potential Staff Finding: The demonstration of the appropriateness of using this approach is adequate.	YES

Notes from staff reviewer: The CGS site is not a "rock" site, but a soil site. Fixed-based analysis was performed only for the Service Building, which is a steel-framed structure with a large basement. The CGS SPRA submittal states that the Service Building does not include SEL components and ISRS are not required; however, collapse of the structure could affect the safety-related piping from CSTs to RB located in the basement. The fixed-base analysis was performed for fragility evaluation associated with structural integrity of the building. The licensee considered the large basement as a rigid base for the light weight steel superstructure and determined the seismic forces in the structural members using lumped mass stick model and response spectrum analysis. Although CGS is a soil site, the fixed-base analysis used for the Service Building is likely to generate a conservative estimate of seismic demand in the force resisting structural members. Fixed-based modal analyses were also performed for the RB, RWCB, DG, TB and CST at the reference earthquake ground motion to assess the extent of concrete cracking and confirm that model properties capture dynamic characteristics (modal frequencies and damping). However, fixed-based models were not used for evaluating building response, in-structure response spectra, or fragility assessment. There were no F&Os associated with fixed-base analysis. Deviation(s) or deficiency(ies) and Resolution: None.

Consequence(s): N/A	
The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to the requirements in the SPID. No requirements in the Code Case Standard specifically address this topic.	N/A
• Although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis	N/A
• The licensee's use of fixed-based dynamic analysis of structures for a site previously defined as "rock" adequately meets the intent of the SPID guidance.	YES
• The licensee's use of fixed-based dynamic analysis of structures for a site previously defined as "rock" does not meet the intent of the SPID guidance, but is acceptable on another justified basis.	N/A

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

Seismic response scaling was used. If no, this issue is moot. If yes, on which structure(s)? All CGS SSCs on SEL	YES
Potential Staff Findings: If a new UHS [uniform hazard spectra] or RLE [review level earthquake] is used, the shape is approximately similar to the spectral shape previously used for ISRS generation.	YES
If the shape is not similar, the justification for seismic response scaling is adequate.	N/A
Consideration of non-linear effects is adequate.	N/A
Notes from staff reviewer: The NRC staff notes that seismic response scaling, as described in SPID Section 6.3.2, was not used to develop ISRS for this submittal from the ISRS generated for previous SPRA. The NRC-approved seismic hazard (ADAMS Accession No. ML16285A410) was used for ISRS development and the seismic fragilities in the CGS submittal. However, in response to F&O 20-10 to SHA-G1, (Tables A-2 and A-3 of CGS SPRA submittal), the disposition stated that the site-specific seismic hazard was since updated and its downstream impact on seismic fragilities was reassessed in a sensitivity study using seismic response scaling. The disposition stated that seismic fragilities were reevaluated using the scaling approach because the spectral shape of the base case seismic hazard approved by the NRC and the revised reference earthquake were similar. The licensee determined the impact of the revised seismic hazard and reevaluated the seismic fragility as a "sensitivity case study" and found that although the SCDF increase was marginal, the increase in SLERF was higher; however, risk-informed decisions and conclusions in the submittal remained unchanged. The NRC staff notes that scaling was only performed for failures associated with horizontal ground motion since the original and revised reference earthquakes spectral shapes were similar up to 10 Hertz (Hz). Above 10 Hz the revised spectra is of higher magnitude. For the soil site, the primary structural response modes in the horizontal direction are below 5 Hz. The NRC staff used the audit process to confirm that the scaling approach used was appropriate to address F&O 20-10 and that it meets the adequacy of structural models, foundation characteristics, and similarity of input ground motion as required in ASME/ANS Code Case requirement SFR-B2. The F&O 20-10 is associated with SHA-G1. There are no F&Os related to SFR-B2.	
Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): N/A	

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

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

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

1. Is the SSI analysis capable of capturing uncertainties and realistic? 2. Is the probabilistic response analysis capable of providing the full distribution of the responses?	YES YES

Notes from staff reviewer:

The CGS SPRA submittal explains in Section 4.3 the structural response analysis including soil-structure interaction analysis (SSI) that was used to develop the in-structure response spectra (ISRS) and fragility analysis. The licensee used a probabilistic response analysis accounting for variabilities in strain-compatible soil profiles, structural characteristics, and the earthquake acceleration time histories performed using a Latin Hypercube sampling (LHS) method. The licensee explains that the LHS process includes generating 30 equal-probability bins and a parameter value is sampled from each bin for each random variable to generate inputs for SSI simulations. The data for LHS simulation was developed from 30 sets of acceleration time history records in 3 orthogonal directions that spectrally matched the Reference Earthquake (uniform hazard response spectrum at mean annual probability of exceedance of 1×10^{-5}); a set of 30 strain-compatible soil profiles consistent with the reference ground motion considering a soil column of 85-ft deep for the site response analyses; and 30 structural frequency and damping parameters while accounting for the statistical variations of these parameters.

The staff audited the CGS information for SSI analysis, which is based on sub-structuring approach that separates the kinematic interaction (foundation scattering of seismic motions) from the inertial interactions (dynamic coupling of structure and foundation impedances). The licensee used a combination of industry standard and in-house software for the soil-foundation models and fixed-based structural models for the soil-structure interaction analysis.

The probabilistic structural response was used to calculate ISRS, and foundation and building displacements including relative displacements at seismic gaps. The ISRS was calculated, for a range of damping values, from acceleration time history output generating median and 84 percent non-exceedance probability (NEP) spectral accelerations at selected locations.

Based on the NRC review of information in the submittal and auditing of structural responses in the e-Portal, the staff finds the CGS probabilistic approach to evaluate structural response and floor response spectra to be appropriate. The probabilistic simulation approach, consideration of variability in soil and structural properties, and the number of simulations used are consistent with ASCE/SEI 4-16 recommendations and industry practice.

There were no Peer Review findings related to all SRs under SFR-B.

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

Consequence(s): N/A

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

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

The selection of SSCs for seismic fragility analysis used a screening approach by capacity following Section 6.4.3 of the SPID. If no, see items D and E. If yes, see items A, B, and C. Potential Staff Findings:	NO
A) The recommendations in Section 6.4.3 of the SPID were followed for the screening aspect of the analysis, using the screening criteria therein.	N/A
B) The approach for retaining certain SSCs in the model with a screening-level seismic capacity follows the recommendations in Section 6.4.3 of the SPID and has been appropriately justified.	N/A
C) The approach for screening out certain SSCs from the model based on their inherent seismic ruggedness follows the recommendations in Section 6.4.3 of the SPID and has been appropriately justified.	N/A
D) The Standard has been followed.	YES
E) An alternative method has been used and its use has been appropriately justified.	YES
Notes from staff reviewer: The screening process is described in Section 4.4.1 of the CGS SPRA submittal. The submittal stated that SSCs judged to have high seismic capacity are identified as inherently rugged components. The rugged components were identified during walkdowns based on whether the component was lightweight, and had robust anchorage, no physical distress, or interaction concerns. These components are expected to have negligible effect on seismic risk and were not included in the SPRA modeling. The licensee performed a sensitivity analysis (Case 10, Table 5.7-1 of the submittal) where a screening HCLPF [high confidence of low probability of failure] was assigned to these SSCs resulting in negligible change in SCDF and SLERF. The NRC staff finds that the CGS approach to screening rugged components is consistent with guidance in SPID. The NRC staff used the audit process to clarify that SSCs that had a “High” capacity identified during the walkdown (Section 4.2.1 of the submittal) were not screened from the PRA model and the in-structure HCLPF capacity assigned is in accordance with the EPRI-6041 (1991) and EPRI 1019200 (2009). The CGS submittal stated in Section 4.4.1 that capacity-based screening was also performed. The licensee used the suggested guidance in SPID Section 6.4.3 to develop a screening level HCLPF value at the initial stage, however, it developed an alternative screening criterion based on Fussell-Vesely (FV) risk importance measure. Screening-level fragility is based on component’s FV ranking less than 0.005 in the SPRA model.	

The licensee stated that only three components were identified based on this screening level criterion. The NRC staff confirmed the CGS approach, to screen SSCs with FV less than 0.005 based on CDF and LERF, is consistent with the recommendations in ASME/ANS PRA Standard (2013) and the *Federal Register* Notification 11488 (FR Vol. 65, No. 43, 2000). However, as stated in the submittal, these components were retained in the SPRA model with the screening level fragility assigned. The CGS submittal screened seismically induced failure of upstream dams from further consideration in SPRA model. The NRC staff's audit review confirmed that CGS evaluated several upstream dams and based on hydrologic studies of flooding from potential dam breaches and concluded that plant grade level was above the flood elevation. There are no F&Os identified related to SFR-C1, SFR-C2 and SPR-B5. Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): None.	
The NRC staff concludes: • The peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to the SR requirements SFR-C1, SFR-C2, and SPR-B5 in the Code Case Standard, as well as to the requirements in the SPID. • Although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis. • The licensee's use of a screening approach for selecting SSCs for fragility analysis meets the intent of the SPID guidance. • 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 N/A YES N/A

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

The CDFM/Hybrid method was used for seismic fragility analysis. If no, See item C) below and next issue. If yes: Potential Staff Findings: A) The recommendations in Section 6.4.1 of the SPID were followed appropriately for developing the CDFM High Confidence Low Probability of Failure capacities. B) The Hybrid methodology in Section 6.4.1 and Table 6-2 of the SPID was used appropriately for developing the full seismic fragility curves. C) An alternative method has been used appropriately for developing full seismic fragility curves.	YES N/A YES YES N/A
Notes from staff reviewer: The NRC staff notes that the licensee has used representative fragilities for initial quantification and refined the fragilities using conservative deterministic failure margin (CDFM) and separation of variables (SOV) methodologies for dominant risk contributors. The CGS SPRA submittal explains that the Hybrid Method was used in developing the fragility for representative and risk significant structures and components. Generic variability values in Table 6-2 of the SPID was used for representative fragilities. For the refined fragility estimate for risk-significant SSCs, CGS used an enhanced hybrid method, where the composite logarithmic standard deviation was estimated for each SSC. The NRC staff used the audit process to understand application of the enhanced hybrid approach and the sources of uncertainties on selected structures and components. Both anchorage and functional fragilities were considered for the selected SSCs. Capacity of SSCs was based on site-specific qualification test data along with guidance provided by EPRI. Demand was based on ISRS at the SSC location in the structures. The licensee stated during the audit that the high uncertainties in CGS fragilities are driven by the high uncertainty in the SSI response. This included high variation in the strain-compatible soil properties and the steeply sloped shape of the reference earthquake (RE) ground motion in the frequency range that governs the SSI response of major buildings. The CGS procedures for development of CDFM/Hybrid fragilities is based on guidance in technical reports EPRI NP-6041(1991) and EPRI 1019200 (2009), and consistent with EPRI SPID recommendation. The NRC staff finds the CGS approach to estimate uncertainties and variabilities for enhanced hybrid method based on site-specific information is reasonable for this submittal. In response to the NRC audit question, the licensee clarified that the fragility evaluation of Service Building is based on Hybrid Method and not based on SOV as noted in Tables 5.4-2 and 5.5-2 of the submittal.	

There were no F&Os for this topic. Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): N/A	
The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to the requirements in the SPID. No requirements in the Code Case Standard specifically address this Topic.	N/A
• Although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's use of 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)

<table>
<tr><td>The SPID requires that certain SSCs that are sensitive to high-frequency seismic motion must be analyzed in the SPRA for their seismic fragility using a methodology described in Section 6.4.2 of the SPID.

<u>Potential Staff Findings</u>:
The NRC staff review of the SPRA’s fragility analysis of SSCs sensitive to high frequency seismic motion finds that the analysis is acceptable.

The flow chart in Figure 6-7 of the SPID was followed.

The flow chart was not followed but the analysis is acceptable on another justified basis.</td><td>

YES

YES

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

The CGS submittal stated that the high capacity components were screened out in accordance with SPID Figure 6-7. Low capacity components were modeled in the fault tree for seismic failure. At the CGS site, because of high influence of SSI on RE response spectra, high frequency ground motion is filtered out. Consequently, the ISRS demand at high frequency was attenuated. Components in locations where building to building contact was identified were addressed using standard practices. For fragility evaluation, the licensee stated that the SOV method was used for all chatter-sensitive devices. Through the audit process, staff reviewed selected relay fragility analysis using the SOV method and found it to be consistent with the guidance in EPRI TR-103959 (1994). There are no F&Os related to SFR-E5.

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

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

• The peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to the SR requirement SFR-E5 in the Code Case Standard, as well as to the requirements in the SPID.

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

• The licensee’s fragility analysis of SSCs sensitive to high frequency seismic motion meets the intent of the SPID guidance.</td><td>N/A

N/A

YES</td></tr>
</table>

• The licensee's fragility analysis of SSCs sensitive to high-frequency motion does not meet the intent of the SPID guidance, but is acceptable on another justified basis.	N/A

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

The SPID requires that certain relays and related devices (generically, "relays") that are sensitive to high-frequency seismic motion must be analyzed in the SPRA for their seismic fragility. Although following the Standard is generally acceptable for the fragility analysis of these components, the SPID (Section 6.4.2) contains additional guidance when either circuit analysis or operator-action analysis is used as part of the SPRA to understand a given relay's role in plant safety. When one or both of these are used, the NRC reviewer should use the following elements of the checklist.	
i) Circuit analysis: The seismic relay-chatter analysis of some relays relies on circuit analysis to assure that safety is maintained. (A) If no, then (B) is moot. (B) If yes: Potential Staff Finding: The approach to circuit analysis for maintaining safety after seismic relay chatter is acceptable.	YES YES
ii) Operator actions: The relay-chatter analysis of some relays relies on operator actions to assure that safety is maintained. (A) If no, then (B) is moot. (B) If yes: Potential Staff Finding: The approach to analyzing operator actions for maintaining safety after seismic relay chatter is acceptable.	YES YES
Notes from staff reviewer: The CGS submittal stated in Section 4.1.2 that contact chatter evaluations were performed. Using the audit review process, the NRC staff finds the licensee provided adequate information on the circuit analysis including event descriptions and chatter evaluation summaries. The staff also finds operator actions were credited for relay-chatter evaluation and screening. In specific cases, where operator actions were credited, the licensee provided information on the approach used in the evaluation. Deviation(s) or deficiency(ies) and Resolution: None Consequence(s): N/A	

The NRC staff concludes that:	
• the peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The relevant peer review findings are those that relate to SR requirement SPR-B6 in the Code Case Standard, as well as to the requirements in the SPID.	N/A
• although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.	N/A
• the licensee's analysis of seismic relay-chatter effects meets the intent of the SPID guidance.	YES
• the licensee's analysis of seismic relay-chatter effects does not meet the intent of the SPID guidance, but is acceptable on another justified basis.	N/A

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

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

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

If <u>yes</u>, significant risk contributors for which use of SOV fragility calculations would make a significant difference in the SPRA results have been selected for SOV calculations."

<u>Potential Staff Findings</u>:
A) The recommendations in Section 6.4.1 of the SPID were followed concerning the selection of the "dominant risk contributors" that require additional seismic fragility analysis using the separation-of-variables methodology.

B) The recommendations in Section 6.4.1 were not followed, but the analysis is acceptable on another justified basis.</td><td>YES

N/A

YES

YES

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

Section 4.4.2 of the SPRA submittal explains that fragility values were developed for all SEL equipment and structures using the Hybrid method or CDFM approach. The licensee's method involved developing a HCLPF and a median seismic capacity for each SSC, from which site-specific variability parameters for the SSC were developed. For risk-significant SSCs, detailed fragilities were developed using the SOV method or a refined Hybrid method. Furthermore, the SOV method was used to develop fragilities for all relay devices subject to chatter during a seismic event. Tables 5.4-2 and 5.5-2 provide, for SCDF and SLERF, respectively, a listing of the risk-significant SSCs (those having a Fussell-Vesely or F-V importance value greater than 0.005) and the method used to develop the fragility for each. Accordingly, the NRC staff concluded that the licensee's approach was to achieve more detailed fragility analyses for dominant risk contributors using the SOV approach or a more refined CDFM approach.

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

Consequence(s): N/A</td></tr>
<tr><td>The NRC staff concludes:
the peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to SFR-E3 in the Code Case Standard and the requirements in the SPID.</td><td>YES</td></tr>
</table>

• although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.	N/A
• the licensee's method for selecting the "dominant risk contributors" for further seismic fragilities analysis using the separation-of-variables methodology meets the intent of the SPID guidance.	YES
• the licensee's method for selecting the "dominant risk contributors" for further seismic fragilities analysis using the separation-of-variables methodology does not meet the intent of the SPID guidance, but is acceptable on another justified basis.	N/A

TOPIC 13: Evaluation of LERF (SPID Section 6.5.1)

The NRC staff review of the SPRA's analysis of LERF finds an acceptable demonstration of its adequacy.	YES
Potential Staff Findings: A) The analysis follows each of the elements of guidance for LERF analysis in Section 6.5.1 of the SPID, including in Table 6-3.	YES
B) The LERF analysis does not follow the guidance in Table 6-3 but the analysis is acceptable on another justified basis.	N/A
Notes from staff reviewer: Section 4.1.1 of the submittal describes the development of a SEL for CGS, including identifying SSCs associated with the containment isolation and integrity safety function and seismic-induced failures that lead to a large early release. Section 5.1.5 further states that the SPRA large early release sequences are based on those developed for the internal event PRA and includes additional containment isolation pathways applicable to seismic events and additional seismic-induced structure failures that contribute to LERF. Lastly, Appendix A of the submittal explains that both the SPRA and the internal events PRA were peer reviewed and all F&Os against Technical Element HLR-SPR-E for the SPRA and against LERF supporting requirements of the internal events PRA were closed using an NRC-accepted process. Topic #14 provides the NRC staff's evaluation of the technical acceptability of the SPRA for supporting the staff's decision on this submittal. Deviation(s) or deficiency(ies) and Resolution: None Consequence(s): N/A	
The NRC staff concludes that:	
• the peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The relevant peer review findings are those that relate to the SR requirements SPR-E1, E5, and E6 in the Code Case Standard, as well as to the requirements in the SPID.	YES
• although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.	N/A
• the licensee's analysis of LERF meets the intent of the SPID guidance.	YES
• the licensee's analysis of LERF does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

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

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

E) The "end-of-process" peer review process followed the peer review guidance in the SPID (Section 6.7) as supplemented by NRC staff comments in the NRC letter dated March 7, 2018 (NRC 2018a, 2018b). If no, go to (F). If yes, the "end-of-process" peer review approach is acceptable. Go to (G).	YES
F) The peer-review process does not follow the guidance in the SPID as supplemented by NRC staff comments in the NRC letter dated March 7, 2018 (NRC 2018a, 2018b), but is acceptable on another justified basis.	N/A
G) The licensee peer-review F&Os were satisfactorily resolved or were determined not to be significant to the SPRA conclusions for this review application.	YES

Notes from staff reviewer:

Section 5.2 and Appendix A of the submittal describe the peer review process used to establish the technical adequacy of the SPRA. The SPRA peer review was conducted in December 2018 against the CC-II supporting requirements (SRs) of PRA Standard ASME/ANS RA-S Case 1 (ASME/ANS RA-S Case 1, 2017) and in accordance with the peer review characteristics. ASME/ANS RA-S Case 1 has been accepted by the NRC for use in regulatory applications, subject to NRC staff comments (NRC, 2018a). During the audit the licensee confirmed that these NRC staff comments were considered during the SPRA peer review. The submittal explains that peer review team utilized the peer review process for external events defined in NEI 12-13 (NEI, 2012). Guidance document NEI 12-13 has been accepted by the NRC, subject to certain NRC staff comments (NRC, 2018b, 2018c). During the audit the licensee confirmed that these NRC staff comments were considered during the SPRA peer review.

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

All elements of the SPRA were peer reviewed against the capability category II (CC-II) requirements SPRA standard. The submittal states that all but one F&O (Finding 20-10) has been closed using an NRC-accepted process, and that all supporting requirements (SRs) have been determined to meet the CC-II requirements. During the audit the licensee confirmed that the NRC's accepted process for closure of F&Os (NRC, 2017a, 2017b) was used, which included a self-assessment by the licensee as to whether each F&O disposition was a PRA maintenance or upgrade, and an assessment by the F&O

closure team of concurrence or disagreement of this determination. The closure team assessment concluded that all but two of the F&O dispositions were PRA maintenance and that these two dispositions incorporated use of a new methodology.

During the audit, the licensee explained that a focused-scope peer review was conducted, concurrent with the F&O closure review, on two F&O dispositions (Finding 20-10 against SR SHA-G1 and Finding 19-2 against SR SPR-D5) that were assessed to be PRA upgrades. The F&O closure review and focused-scope peer review were conducted in July 2019, using the NEI 12-13 guidance, against the CC-II supporting requirements of PRA Standard ASME/ANS RA-S Case 1 (ASME/ANS RA-S Case 1, 2017). During the audit the licensee confirmed that the NRC staff comments on ASME/ANS RA-S Case 1 and proposed resolutions (NRC, 2018a) and on NEI 12-13 (NRC, 2018b, 2018c) were considered during these reviews.

Regarding Finding 19-2, a focused-scope peer review was conducted of the licensee's resolution to Finding 19-2, and a reassessment of all HLR SPR-D SRs was performed. All these SRs were determined to meet CC-II, and no Findings were assigned. The original Finding 20-10 and the results of the focused-scope peer review are provided in the submittal. The licensee's resolution to this F&O was to revise the seismic hazard used in the SPRA. The change in the seismic hazard necessitated reassessment of the seismic fragilities, which was done using a scaling approach. During the audit, the licensee clarified that this scaling approach was not previously utilized in the development of the CGS SPRA and was considered a PRA upgrade by the F&O closure team. A concurrent focused-scope peer review was conducted of the licensee's scaling approach, and a reassessment of all HLR SFR-B SRs was performed. All these SRs were determined to meet CC-II, and no Findings were developed.

The NRC staff has previously accepted the licensee's base case seismic hazard used in this submittal suitable for other actions associated with Near-Term Task Force Recommendation 2.1, 'Seismic' and this submittal is related to that recommendation. In addition, the licensee performed a sensitivity study to assess the impact on the submittal of the revised seismic hazard and seismic fragilities, which showed increased SCDF of 4 percent, increased SLERF of 34 percent, and increased importance of certain systems, structures, and components (SSCs). The NRC staff considered the impact of the sensitivity on its decision for this submittal as discussed in the Detailed Screening Evaluation provided in Enclosure 2.

Section 5.1 of the submittal states the internal events PRA (IEPRA) model-of-record as of January 28, 2019, was used as the basis for the development of the SPRA model. Section A.7 of the submittal states the IEPRA (including internal flooding) was peer reviewed in December 2009 against the CC-II requirements of the PRA standard (ASME/ANS Addendum A, 2009) and Regulatory Guide (RG) 1.200, Rev. 2 and that all F&Os have been subsequently closed. During the audit the licensee explained that this peer review utilized the peer review process in NEI 05-04. It was also explained that an F&O closure review and concurrent focused-scope peer review of the IEPRA (including internal flooding) was conducted in March 2018 using the NRC's accepted process for closure of F&Os (NRC, 2017a, 2017b), which included a self-assessment by the licensee as to whether each F&O disposition was a PRA maintenance or upgrade, and an assessment by the F&O closure team of concurrence or disagreement of this determination. The closure review team determined that all Findings were closed, that all SRs that were previously Not Met or Met at CC-I were determined to be met at CC-II, and that all changes made to the PRA were maintenance updates and not PRA

<table>
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upgrades.

The licensee further explained during the audit that, subsequent to the original peer review, changes were made to the HRA methodology that were determined by the licensee to be a PRA upgrade, which was agreed with by the F&O closure team. A concurrent focused-scope peer review was conducted of HLRs HR-G, HR-H, HR-I, and relevant SRs under HLR QU-C against the CC-II requirements of the PRA standard (ASME/ANS Addendum A, 2009), RG 1.200, Rev. 2, and NEI 05-04. This peer review developed 11 Finding-level F&Os and assigned a Not Met at CC-II to 2 SRs. Subsequent to the focused-scope peer review, an F&O closure review was conducted in May/June 2018 using the NRC's accepted process for closure of F&Os (NRC, 2017a, 2017b). As a result of this review, all Finding-level F&Os were closed, the two SRs that were previously Not Met were determined to be met at CC-II, and the changes made to the PRA to resolve the Findings were determined to be maintenance updates and not PRA upgrades.

Because the licensee used NRC-accepted processes for performing the peer review, focused-scope peer review, and F&O closure processes, the NRC staff concluded that the licensee's IEPRA is of sufficient technical acceptability to form the base for the development of the SPRA used in this submittal.

Based on the NRC staff's assessment that the licensee's revised seismic hazard analysis is acceptable, that a sensitivity analysis was performed to assess the impact of these changes on the submittal, the licensee peer-reviewed its SPRA using an accepted Code Case and peer-review guidance, an NRC-accepted process was used to close the SPRA F&Os, and that the technical acceptability of the underlying internal events PRA, the NRC staff concluded that the licensee's SPRA is of sufficient technical adequacy for its decision on this SPRA submittal.

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

Consequence(s): N/A
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The NRC staff concludes:

• the licensee's peer-review process meets the intent of the SPID guidance as supplemented by NRC staff comments in the NRC letter dated March 7, 2018 (NRC 2018a, 2018b).

• the licensee's peer-review process does not meet the intent of the SPID guidance as supplemented by NRC staff comments in the NRC letter dated March 7, 2018 (NRC 2018a, 2018b), but is acceptable on another justified basis.
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YES

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

The NRC staff review of the SPRA's documentation as submitted finds an acceptable demonstration of its adequacy. The documentation should include all of the items of specific information contained in the 50.54(f) letter as described in Section 6.8 of the SPID.	YES YES

Notes from staff reviewer:

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

Section 5.6 of the submittal presents the SPRA quantification uncertainty results for SCDF and SLERF (i.e., the median (50 percent) and the 95th percentiles). The mean from the uncertainty analysis was not provided, but rather the SCDF and SLERF point estimates of 2.0E-5 per year and 8.8E-6 per year, respectively, were stated in the submittal to be more realistic of the mean. During the audit, the licensee provided the actual mean SCDF and SLERF from the uncertainty analysis, which are 4.83E-05 per year and 1.58E-05 per year, respectively. In addition, the 95th percentile SCDF and SLERF of 1.15E-04 per year and 4.38E-05 per year, respectively, from the uncertainty analysis were provided in the submittal. These mean and 95th percentile values were used in the NRC staff's screening evaluation reported in Enclosure 2 of this document.

According to Section 4.1.1 of the SPRA submittal, Diverse and Flexible Coping Strategies (FLEX) is credited in the SPRA to provide emergency ac power, via credit for the FLEX portable diesel generators, and low-pressure injection, via credit for the FLEX portable diesel fire pumps. The NRC memorandum dated May 30, 2017, "Assessment of the Nuclear Energy Institute 16-06, 'Crediting Mitigating Strategies in Risk-Informed Decision Making,' Guidance for Risk-Informed Changes to Plants Licensing Basis" (ADAMS Accession No. ML17031A269), provides the NRC staff's assessment of the challenges of incorporating FLEX coping strategies and equipment into a PRA model in support of risk-informed decisionmaking in accordance with the guidance of RG 1.200, Revision 2 (ADAMS Accession No. ML090410014). However, Sections 5.4 and 5.5 for the SPRA submittal indicates that credit for FLEX is not significant to the risk results and the conclusions of the submittal. Because FLEX equipment and actions would not change the staff's decision for this submittal, the licensee's treatment of FLEX was not pursued by the NRC staff.

Appendix A of the submittal explains that the SPRA was peer reviewed and all F&Os against Technical Elements HLR-SHA-J, HLR-SFR-G, and HLR-SPR-F were closed using an NRC-accepted process. Topic #14 provides the NRC staff's evaluation of the technical acceptability of the SPRA to support decisionmaking on this submittal.

During its review of the SPRA submittal the NRC staff observed that none of the top 10

cutsets provided for both SCDF and SLERF included Basic Event S_E-MC-7A_8A_S11D, failure of motor control centers on elevation 467 of the Radwaste Building (RWCB) due to seismic-induced fires, even though this basic event is in the top five list of risk-significant failures. During the audit the NRC staff asked the licensee to explain the rationale for risk-significant basic events not showing up repeatedly in the topmost cutsets. The licensee explained that the appearance of basic events in the topmost cutsets is not necessarily an indication of their importance because the importance lists are generated using ACUBE while the cutsets are generated using CAFTA, and that CAFTA cutsets are provided because ACUBE does not generate cutsets. ACUBE quantification results are used in the SPRA submittal because it's Binary Decision Diagram (BDD) methodology improves the cutset probability calculation compared to the Min Cut Upper Bound (MCUB) methodology used in CAFTA. The licensee further noted that the basic event identified by the staff does appear in almost one-quarter of the top 1,000 SCDF cutsets and over 10 percent of the top 10,000 SLERF cutsets. The NRC staff agrees that the BDD methodology is a more accurate methodology for estimating the cutset probability than MCUB and with the licensee's explanation of the identification of risk-significant basic events. Deviation(s) or deficiency(ies) and Resolution: None Consequence(s): None	
The NRC staff concludes: • The licensee's documentation meets the intent of the SPID guidance. The documentation requirements in the Code Case Standard can be found in HLR-SHA-J, HLR-SFR-F, and HLR-SPR-F. • The licensee's documentation does not meet the intent of the SPID guidance but is acceptable on another justified basis.	YES N/A

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

The licensee: • identified modifications necessary to achieve seismic risk improvements • provided a schedule to implement such modifications (if any), consistent with the intent of the guidance • provided Regulatory Commitment to complete modifications • provided Regulatory Commitment to report completion of modifications	NO N/A N/A N/A
Plant will: • complete modifications by: • report completion of modifications by:	N/A N/A
Notes from the Reviewer: Refer to Enclosure 2 for the detailed screening evaluation. Deviation(s) or Deficiency(ies), and Resolution: None Consequence(s): N/A	
The NRC staff concludes that the licensee: • identified plant modifications necessary to achieve the appropriate risk profile • provided a schedule to implement the modifications (if any) with appropriate consideration of plant risk and outage scheduling	N/A N/A

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

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

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NRC Staff SPRA Submittal Detailed Screening Evaluation

Introduction

The Columbia Generating Station (CGS) Seismic Probabilistic Risk Assessment (SPRA) report (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19273A907) provides the point estimate seismic core damage frequency (SCDF) as 2.0E-05/reactor-year (/rx-yr) and seismic large early release frequency (SLERF) as 8.8E-06/rx-yr. The mean SCDF and SLERF values are not provided in the SPRA report but the 50 percent and 95 percent values were provided. During the audit the licensee provided the mean SCDF of 4.83E-5/rx-yr and SLERF of 1.58E-05/rx-yr, which are used in this evaluation. The report also provides the results of a sensitivity study that assesses the impact of the revised seismic hazard, as the resolution to a SPRA peer review finding level F&O, which shows that these point estimate SCDF and SLERF values increase by 4 percent and 34 percent, respectively. The NRC staff compared these values, including the results of the sensitivity study, against the guidance in NRC staff memorandum dated August 29, 2017, titled, "Guidance for Determination of Appropriate Regulatory Action Based on Seismic Probabilistic Risk Assessment Submittals in Response to Near Term Task Force Recommendation 2.1: Seismic" (ADAMS Accession No. ML17146A200; hereafter SPRA Screening Guidance), which establishes a process the NRC staff uses to develop a recommendation on whether the plant should move forward as a Group 1, 2, 3 plant.[1]

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

[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).

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

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

- Additional consideration of containment performance, as described in NUREG/BR-0058, 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.

<u>Detailed Screening</u>

Energy Northwest (the licensee for CGS), 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 CGS license, or the rules and orders of the Commission. For these reasons, the licensee and the staff did not identify a potential modification necessary for adequate protection or compliance with existing requirements.

The detailed screening uses information provided in the CGS SPRA report, particularly the importance measures, SCDF, and SLERF, as well as other information described below, to establish threshold and target values that are used to identify areas where potential cost-justified substantial safety improvements might be identified. The detailed screening process makes several simplifying assumptions, similar to a Phase 1 SAMA analysis (NEI 05-01, ADAMS Accession No. ML060530203) used for license renewal applications. The detailed screening process uses risk importance values as defined in NUREG/CR-3385, "Measures of Risk Importance and Their Applications" (ADAMS Accession No. ML071690031). The NUREG/CR-3385 states that the risk reduction worth (RRW) importance value is useful for prioritizing feature improvements that can most reduce the risk. The CGS SPRA report provides Fussell-Vesely (F-V) importance values, which were converted to RRW values by the NRC staff for this screening evaluation using a standard relationship formulation.

Data used to develop the maximum averted cost-risk (MACR) for the severe accident mitigation alternative (SAMA) analysis provided in the License Renewal Application, Columbia Generating Station, dated January 2010 (ADAMS Accession Nos. ML100250656, ML100250658, ML100250654, and ML100250666), and associated supplements were used to calculate the RRW threshold. For this analysis, the NRC staff determined the RRW threshold from the SCDF-based MACR to be 1.048. 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 RRWs based on the mean and 95th percentile SCDF and SLERF were also calculated by the NRC staff and ranged between 1.02 and 1.26.

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

The NRC staff considered both single and combinations of basic events in accordance with the SPRA Screening Guidance. It is not the intent of that aspect of the guidance to aggregate several disparate basic events that individually have RRW values close to the mean target RRW. A review of these model elements in Tables 1 and 2 of this enclosure reveals that most modifications or sets of modifications to achieve a SCDF reduction of at least 1.0E-05/rx-yr or a SLERF reduction of at least 1.0E-06/rx-yr will have to mitigate or prevent multiple failure types (e.g., seismically-induced failures, random failures[2], and failure of operator actions) and failure modes (e.g., seismically-induced structural failures of multiple SSCs and seismically-induced functional failures of multiple SSCs).

The highest contributor to SCDF and second highest contributor to SLERF was seismically-induced loss of offsite power (S_SEIS-SWY-LOSP), which exceeded the mean target RRW for both SCDF and SLERF. According to Table 5.4-5 of the submittal, this basic event is a contributor to 3 of the top 10 SCDF cutsets. During the audit, the licensee explained that S_SEIS-SWY-LOSP represents seismic-induced loss of offsite power from both the plant switchyard and from offsite power lines and that the fragility used in the SPRA is a single representative fragility representing both. Because this event involves seismic-induced failures outside of the plant boundary, the NRC staff did not pursue potential improvements to S_SEIS-SWY-LOSP. The highest contributor to SLERF and third highest contributor to SCDF was seismically-induced failure of the Radwaste, Turbine, and Reactor Buildings (S_RW_TB_RB), which exceeded the mean target RRW for SLERF. According to Tables 5.4-5 and 5.5-5 of the submittal, this basic event is a contributor to the top 7 SCDF cutsets and all of the top 10 SLERF cutsets. The NRC staff experience from SAMA analyses is that the implementation cost of modifications to plant structures that are important to safety, and which would be sufficient to substantially reduce the probability of structural failure, exceed the calculated MACR for this detailed screening. During the audit, the staff did not find anything that would exclude CGS from the generic conclusion made from the SAMA analyses. The NRC staff therefore did not pursue potential improvements to S_RW_TB_RB.

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

The second highest contributor to SCDF and fourth highest contributor to SLERF is seismically-induced chatter of electrical contact devices (ECDs) in Cabinet E-MC-4 (S_CHTR-GR-4), which exceeded the mean target RRW for both SCDF and SLERF. The third highest contributor to SLERF is seismically-induced chatter of ECDs due to correlated building-to-building impacts (S_CHTR-GR-1), which exceeded the mean target RRW for SLERF (and which shows up on the list of risk-significant basic events for SCDF). In addition, other seismically-induced chatter events are shown as risk-significant (e.g., S_CHTR-GR-5A for SCDF and S_CHTR-GR-5B for SLERF). According to the submittal, all unscreened ECDs were modeled using SOV fragilities and, generally, with operator actions to recover the chatter events. During the audit, the licensee provided the results of a sensitivity analysis that showed the composite reduction in SCDF and SLERF from eliminating the above identified seismically-induced ECD chatter failure events was less than 1.0E-05/rx-yr and 1.0E-06/rx-yr, respectively. Based on this result, the licensee concluded that there were no cost-justified plant improvements that could reduce the SCDF or SLERF contributions of these chatter events by 1.0E-05/rx-yr or 1.0E-06/rx-yr, respectively. The NRC staff's assessment of the licensee's sensitivity analysis is that it was based on using the point-estimate values rather than the actual calculated mean values from the parametric uncertainty analysis, and that the composite reduction in SCDF and SLERF would be higher if the calculated means were used. However, since operator actions to recover the chatter events are already credited in the SPRA, the NRC staff concludes that plant modifications to achieve all of the risk reduction reflected by the importance measures for these ECDs would exceed the maximum averted cost, and not be cost-justified.

For the sensitivity study performed by the licensee in response to the peer-review finding level F&O 20-10, Appendix A of the CGS SPRA report also provides F-V importance values for four SSCs in which the F-V values increased from the base case results. The NRC staff evaluation of the results of the sensitivity study did not change the outcome of this detailed screening evaluation.

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

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

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

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

Conclusion

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

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

- no potential cost-justified substantial safety improvement was identified based on the estimated achievable reduction in SCDF and/or SLERF; and

- additional consideration of containment performance, as described in NUREG/BR-0058 and assessed via SLERF, did not identify a modification that would result in a substantial safety improvement.

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

Description	Failure Mode	RRW	MCR (/yr)
Seismically-failed SSCs			
Loss of offsite power	Loss of Offsite Power	2.000	2.42E-05
Chatter Group 4 - E-MC-4 interaction	Chatter	1.316	1.16E-05
CGS Structures: Radwaste, Turbine and Reactor Buildings	Functional Failure (Composite fragility is derived from the individual building fragilities)	1.176	7.25E-06
RWCB 467 Elevation Motor Control Centers	Seismic-Induced Fire	1.105	4.59E-06
Motor Control Centers 7F and 8F	Functional Failure	1.092	4.06E-06
Service Building Failure	Functional Failure	1.043	1.98E-06
HVAC Ducts in DG-3 Room	Functional Failure	1.037	1.74E-06
Chatter Group 5 - E-SL-73 Interactions	Chatter	1.030	1.40E-06
Chatter Group 1 - Building-to-building Impact	Chatter	1.024	1.11E-06
E-MC-7AA and E-MC-8AA	Functional Failure	1.022	1.06E-06
MCR Cabinet Correlation Group B	Loss of Control Room Instrumentation/Control	1.019	9.18E-07
MCR Cabinet Correlation Group A3	Loss of Control Room Instrumentation/Control	1.019	9.18E-07
MCR Cabinet Correlation Group A2	Loss of Control Room Instrumentation/Control	1.019	9.18E-07
MCR Cabinet Correlation Group A1	Loss of Control Room Instrumentation/Control	1.019	9.18E-07
HPCS ENGINE DG-ENG-1C HI CRANKCASE PRESS ALARM & SHUTDOWN (1"H2O) – Contact Chatter	Chatter	1.018	8.69E-07
Chatter Group 29 - Switchgear Lockout	Chatter	1.017	8.21E-07
CRITICAL SWGR ROOMS AIR HANDLING UNITS	Functional Failure	1.016	7.73E-07
HVAC ducting in the RWCB BLDG on elevation 525 & 527+	Seismic-Induced Fire	1.015	7.25E-07
Chatter Group - RCIC Auto Isolation	Chatter	1.013	6.28E-07
DG ROOM STANDBY AIR HANDLING UNITS	Functional Failure	1.010	4.69E-07
ALTERNATE SOURCE 480 VAC DIESEL GENERATOR SET (DG 4)	Functional Failure	1.009	4.54E-07
Loss of RWCU pressure boundary	Seismic-Induced Flood, HELB, Seismic-Induced Break Outside Containment	1.008	3.86E-07
E-MC-6C Damaged on RWCB 437 - Fire Potential	Seismic-Induced Fire	1.007	3.38E-07
DIV 1 AND 2 CRITICAL POWER SUPPLY INVERTERS	Functional Failure	1.007	3.38E-07
CONDITIONAL PROBABILITY OF SEISMIC E-MC-6C FIRE	N/A	1.007	3.38E-07
HVAC ducting in the RWCB control room area on elevation 501	Functional Failure	1.006	2.95E-07
Chatter Group 5 - Interactions	Chatter	1.006	2.95E-07

Description	Failure Mode	RRW	MCR (/yr)
Randomly-failed SSCs			
EMERGENCY DG SYSTEM DOES NOT CONTINUE TO RUN FOR 24H	Not Applicable	1.029	1.35E-06
RCIC PUMP FAILS TO RUN FOR 6 TO 36 HOURS	Not Applicable	1.010	4.78E-07
EMERGENCY DG-3 DOES NOT CONTINUE TO RUN FOR 4 TO 24 H	Not Applicable	1.009	4.44E-07
DG-3 OUT FOR MAINTENANCE	Not Applicable	1.008	3.82E-07
Human Failure Events			
Dependent HFE: SEIHUMN-HPCS-NR (failure to recover HPCS given pump suction isolation), CR-HUMN-CR-HVAC (failure to align alternate control room HVAC), ADSHUMNSTARTH3LT (failure to depressurize the RPV)	Not Applicable	1.054	2.46E-06
Dependent HFE: CR-HUMN-CR-HVAC, SEIHUMN-ALT_IC (failure to align alternate RPV level indication), OP-HUMNRSP (failure to shut down using remote shutdown panel).	Not Applicable	1.026	1.21E-06
Failure to recover HPCS given pump suction isolation by realigning suction path or stopping pump	Not Applicable	1.013	6.28E-07
Failure to locally operate RCIC without dc or ac power	Not Applicable	1.007	3.53E-07
Dependent HFE: SEIHUMN-HPCS-NR, CR-HUMN-CR-HVAC	Not Applicable	1.007	3.19E-07
Dependent HFE: SEIHUMN-HPCS-NR, RHRHUMNSPCOOLLL (failure to align suppression pool cooling)	Not Applicable	1.006	2.90E-07
Dependent HFE: SEIHUMN-HPCS-NR, RCIHUMN-CST-H3LL (failure to align RCIC suction to suppression pool)	Not Applicable	1.006	2.80E-07
Dependent HFE: SEIHUMN-HPCS-NR, CR-HUMN-CR-HVAC, OP-HUMN-RSP	Not Applicable	1.006	2.70E-07
Dependent HFE: SEIHUMN-EDG-RECOV-LOC (failure to locally recover DG – contact chatter), CR-HUMN-CR-HVAC, ADSHUMNSTARTH3LT	Not Applicable	1.005	2.46E-07

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

Description	Failure Mode	RRW	MCR (/yr)
Seismically-failed SSCs			
CGS Structures: Radwaste, Turbine and Reactor Buildings	Functional Failure (Composite fragility derived from the individual building fragilities)	1.961	7.74E-06
Loss of offsite power	Loss of Offsite Power	1.351	4.11E-06
Chatter Group 1 - Building-to-building Impact	Chatter	1.086	1.25E-06
Chatter Group 4 - E-MC-4 interaction	Chatter	1.076	1.12E-06
RWCB 467 Elevation Motor Control Centers	Seismic-Induced Fire	1.054	8.06E-07
Loss of RWCU pressure boundary	Seismic-Induced Flood, HELB, Seismic-Induced Break Outside Containment	1.050	7.58E-07
Chatter Group 5B - E-MC-7BA Interactions	Chatter	1.046	6.95E-07
SW Spray Pond Structure[3]	Functional Failure	1.029	5.91E-07
Motor Control Centers 7F and 8F	Functional Failure	1.026	3.95E-07
Chatter Group - RCIC Auto Isolation	Chatter	1.025	3.79E-07
Service Building Failure	Functional Failure	1.024	3.63E-07
HVAC Ducts in DG-3 Room	Functional Failure	1.022	3.48E-07
HVAC ducting in the RWCB BLDG on elevation 525 & 527+	Seismic-Induced Fire	1.021	3.32E-07
HPCS ENGINE DG-ENG-1C HI CRANKCASE PRESS ALARM & SHUTDOWN (1"H2O)	Chatter	1.021	3.32E-07
E-MC-7AA and E-MC-8AA	Functional Failure	1.018	2.84E-07
CRITICAL SWGR ROOMS AIR HANDLING UNITS	Functional Failure	1.015	2.37E-07
Reactor Building Recirculation Air Fan Cooler 10	Functional Failure	1.014	6.78E-07
Reactor Building Recirculation Air Fan Cooler 10	Functional Failure	1.014	6.78E-07
CONDITIONAL PROBABILITY OF SEISMIC FOR E-MC-7BB FIRE	N/A	1.010	1.58E-07
CONDITIONAL PROBABILITY OF SEISMIC FOR E-MC-7F FIRE	N/A	1.010	1.58E-07
Chatter Group 13 - HPCS Impacted due to contact chatter; recoverable	Chatter	1.009	1.36E-07
Control Rod Drive Hydraulic Control Units	Functional Failure	1.009	4.66E-07
Chatter Group 29 - Switchgear Lockout	Chatter	1.007	1.15E-07
MCR Cabinet Correlation Group A1	Loss of Control Room Instrumentation/Control	1.007	1.12E-07
Chatter Group 5 - E-SL-73 Interactions	Chatter	1.007	1.12E-07
DG ROOM STANDBY AIR HANDLING UNITS	Functional Failure	1.007	1.06E-07
Control Rod Drive Hydraulic Control Units	Functional Failure	1.009	4.66E-07
MCR Cabinet Correlation Group B	Loss of Control Room Instrumentation/Control	1.006	9.95E-08

[3] Added from results of the sensitivity study (see Table A-3 of the CGS SPRA Report).

MCR Cabinet Correlation Group A3	Loss of Control Room Instrumentation/Control	1.006	9.95E-08
MCR Cabinet Correlation Group A2	Loss of Control Room Instrumentation/Control	1.006	9.95E-08
RPV COOLDOWN VENT TO EQUIPMENT DRAIN MOTOROPERATED VALVES	SLOCA	1.006	8.69E-08
Randomly-failed SSCs			
EMERGENCY DG SYSTEM DOES NOT CONTINUE TO RUN FOR 24H	Not Applicable	1.007	1.12E-07
Human Failure Events			
Failure to recover HPCS given pump suction isolation by realigning suction path or stopping pump – Plant Damage Bin 2	Not Applicable	1.014	2.21E-07
Failure to recover HPCS given pump suction isolation by realigning suction path or stopping pump – Plant Damage Bin 3	Not Applicable	1.009	1.42E-07
Dependent HFE: SEIHUMN-HPCSNR, RCIHUMN-CST-H3LL (failure to align RCIC suction to suppression pool)	Not Applicable	1.005	8.06E-08

AUDIT SUMMARY BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO

COLUMBIA GENERATING STATION

SUBMITTAL OF SEISMIC PROBABILISTIC RISK ASSESSMENT ASSOCIATED WITH

REEVALUATED SEISMIC HAZARD IMPLEMENTATION OF THE

NEAR-TERM TASK FORCE RECOMMENDATION 2.1: SEISMIC

(EPID NO. L-2019-JLD-0009)

BACKGROUND AND AUDIT BASIS

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

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

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

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 and request for supporting documents dated January 10, 2020, and November 1, 2019 (ADAMS Accession Nos. ML20013G764 and ML19305C934, respectively), were sent to the licensee to support the audit.

The licensee provided clarifying information in the following areas:

- Information describing the relationship between Internal Events PRA and the Seismic PRA.
- Discussion of conservatism on fragilities for certain structures, systems, and components (SSCs) in response to Fact and Observation (F&O) 22-5 and ruggedness ranking during walkdowns.
- Uncertainties on fragility calculations for different SSCs.
- Use of full-scope and focused-scope peer review, and consideration of NRC staff comments and resolutions on NEI-12-13 (ADAMS Accession Nos. ML18025C025 and ML18025C022).
- Potential plant modifications that could reduce SCDF and SLERF (none identified).

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

<u>DOCUMENTS AUDITED</u>

- Simpson Gumpertz & Heger Inc., Seismic Fragility Evaluation of Columbia Generating Station Structures, Systems, and Components, Report No. 168059-R-04, Revision 2, Newport Beach, CA, 2019.
- Simpson Gumpertz & Heger Inc., Seismic Response Analysis of Columbia Generating Station Structures, Report No. 168059-R-03, Revision 2, Newport Beach, CA, 2019.
- Simpson Gumpertz & Heger Inc., Criteria Document for the Columbia Generating Station Seismic Fragility Evaluation, Document No. 168059-CD-01, Revision 4, Newport Beach, CA, 2019.
- Simpson Gumpertz & Heger Inc., Seismic Fragility Update for Revised Seismic Hazard of Columbia Generating Station, Report No. 168059-R-05, Revision 0, Newport Beach, CA, 2019.
- SPR-CONTACT-CHATTER, Columbia Generating Station Seismic Probabilistic Risk Assessment: Contact Chatter Report, ENERCON Services, Inc., Revision 1, 2019.
- Simpson Gumpertz & Heger Inc., Seismic Walkdown of Columbia Generating Station, Report No. 168059-R-02, Revision 2, Newport Beach, CA, 2019.
- Bechtel Power, "Dynamic Geotechnical Engineering Properties, Calculation 25709-000-K0C-0000-00001, Revision 0," 2014.
- BWR Owners Group, Columbia Generating Station, Seismic PRA Peer Review Report Using ASME/ANS PRA Standard Requirements, Revision 0, 2019.

- 026022-RPT-01, Columbia Generating Station SPRA Finding-Level Fact and Observation Independent Assessment and Focused-Scope Peer Review, Revision 0, 2019.

- SPR-QU, CGS SPRA Quantification, ENERCON Services, Inc., Revision 4, 2019.

- SPR-PRM, CGS Seismic PRA: Plant Response Model and Human Reliability Analysis, Revision 3, 2019.

- 026016-RPT-01, Columbia Generating Station PRA Finding-Level Fact and Observation Independent Assessment and Focused-Scope Peer review, Revision 0, June 29, 2018.

OPEN ITEMS AND REQUEST FOR INFORMATION

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

DEVIATIONS FROM AUDIT PLAN

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

AUDIT CONCLUSION

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

SUBJECT: COLUMBIA GENERATING STATION – STAFF REVIEW OF SEISMIC PROBABILISTIC RISK ASSESSMENT ASSOCIATED WITH REEVALUATED SEISMIC HAZARD IMPLEMENTATION OF THE NEAR TERM TASK FORCE RECOMMENDATION 2.1: SEISMIC DATED APRIL 28, 2020

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ADAMS Accession No. ML20076A547 ***via e-mail**

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