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

September 12, 2020

Mr. Fadi Diya
Senior Vice President and
Chief Nuclear Officer
Ameren Missouri
Callaway Energy Center
8315 County Road 459
Steedman, MO 65077

SUBJECT: CALLAWAY PLANT, UNIT 1 – 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-0006)

Dear Mr. Diya:

The purpose of this letter is to document the staff's evaluation of the Callaway Plant, Unit 1 (Callaway), 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 Callaway.

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

By letter dated August 12, 2019 (ADAMS Package Accession No. ML19225D321), Ameren Missouri (Ameren, the licensee), provided its SPRA submittal in response to Enclosure 1, Item (8) of the 50.54(f) letter, for Callaway. The submittal was supplemented by letters dated November 21, 2019, and July 10, 2020 (ADAMS Accession Nos. ML19325D668 and ML20192A244, respectively). As applicable, the NRC staff assessed the licensee's implementation of the Electric Power Research Institute's Report 1025287, "Seismic Evaluation Guidance - Screening, Prioritization, and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic" (ADAMS Accession No. ML12333A170). This report was endorsed by the NRC by letter dated February 15, 2013 (ADAMS Accession No. ML12319A074). In addition, consistent with the licensee's submittal, the NRC staff utilized a reviewer checklist that is based on ASME (American Society of Mechanical Engineers)/ANS (American Nuclear Society) RA-S Case 1 "Case for ASME/ANS Ra-Sb-2013, Standard for Level 1/Large Early Release Frequency Probabilistic Risk

Assessment for Nuclear Power Plant Applications" (hereafter 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 Callaway SPRA submittal is contained in Enclosure 1 to this letter. As described below, the NRC has concluded that the Callaway 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 April 17, 2014 (ADAMS Accession No. ML14090A446), Ameren submitted the reevaluated seismic hazard information for Callaway. The NRC performed a staff assessment of the submittal and issued a response letter on April 21, 2015 (ADAMS Accession No. ML15063A517). The NRC's assessment concluded that Ameren 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 Callaway.

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, Callaway was expected to complete an SPRA with an estimated completion date of December 31, 2017, which would also assess high frequency ground motion effects. In letters dated June 15, 2017, and November 14, 2018 (ADAMS Accession Nos. ML17166A474 and ML18318A059, respectively), Ameren requested extensions to submit its SPRA at later dates. The staff approved these extensions by letters dated August 22, 2017, and January 10, 2019 (ADAMS Accession Nos. ML17200D113 and ML19004A400, respectively). In addition, Ameren was expected to perform a limited-scope evaluation for the spent fuel pool (SFP). This SFP limited-scope evaluation was submitted by letter dated October 3, 2017 (ADAMS Accession No. ML17276B201). The staff provided its assessment of the Callaway SFP evaluation by letter dated January 23, 2018 (ADAMS Accession No. ML18003B419).

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

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

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

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

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

The evaluation performed to provide the basis for the staff's grouping recommendation to the SMRP for Callaway is described below. Based on its evaluation, the staff recommended to the SMRP that Callaway be classified as a Group 1 plant and therefore, no further regulatory action is 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 decision-making had been included in the licensee's submittal. The technical team performing the review consisted of staff experts in the fields of seismic hazards, fragilities evaluations, and plant response/risk analysis. On September 20, 2019, 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 licensee's dispositions. These elements were reviewed by NRC staff in the context of the regulatory decision-making associated with the 50.54(f) letter.

By letter dated July 6, 2017 (ADAMS Accession No. ML17177A446), the NRC issued a generic audit plan and entered into the audit process described in Office Instruction LIC-111, "Regulatory Audits," dated December 29, 2008 (ADAMS Accession No. ML082900195), to assist in the timely and efficient closure of activities associated with the 50.54(f) letter. By letter dated July 11, 2017 (ADAMS Accession No. ML17192A168), the NRC staff confirmed that the audit process for the seismic hazard reevaluations applies to the Callaway site. The staff exercised the audit process by reviewing selected licensee documents via an electronic reading room (eportal) as documented in Enclosure 3 to this letter.

During the audit process, the staff developed questions to clarify information in the licensee's submittal and to gain understanding of non-docketed information that supports the docketed SPRA submittal. The staff's clarification questions dated September 20, 2019, November 13, 2019, November 29, 2019, and January 2, 2020 (ADAMS Accession Nos. ML19304C325, ML19317E633, ML19333B869, and ML20101F977, respectively), were sent to the licensee to support the audit. The licensee subsequently provided answers to the questions on the eportal, which the staff reviewed. The staff determined that the answers to the questions provided in the eportal served to confirm statements that the licensee made in its 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. In its submittal, the licensee stated as dispositions to SPRA F&Os 25-19 and 25-12 that the IEPRA would be peer reviewed and F&Os dispositioned. Upon dispositioning IEPRA F&Os, the IEPRA model would be pulled in as the base model for the SPRA. In its supplement letter dated July 10, 2020 (ADAMS Accession No. ML20192A244), the licensee explained how the use of new software for the IEPRA and the disposition of the IEPRA F&Os altered the results and conclusions of the SPRA submittal dated August 12, 2019. After considering this information, the NRC staff identified no issues with the supplemental information 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 decision-making associated with Phase 2 of the 50.54(f) letter.

The staff's review process included the completion of the SPRA Submittal Technical Review Checklist (SPRA Checklist) contained in Enclosure 1 to this letter. As described in Enclosure 1, the SPRA Checklist is a document used to record the staff's review of licensees' SPRA submittals against the applicable guidance of the Code Case Standard, as described in the NRC letter to the NEI dated July 12, 2018. Enclosure 1 contains the staff's application of the SPRA checklist to Callaway's submittal. As documented in the checklist, the staff concluded that the Callaway 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 Callaway 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 Callaway to the SMRP. The criteria in the staff's guidance document includes thresholds to assist in determining whether to apply the backfit screening process described in Management Directive 8.4, "Management of Facility-Specific Backfitting, Forward Fitting, Issue Finality, and Information Requests," dated September 20, 2019 (ADAMS Accession No. ML18093B087), to the SPRA submittal review. As part of this review, the staff considered the modifications described in the Callaway SPRA which are already implemented. In its submittal, Ameren provided regulatory commitments to perform certain plant modifications at Callaway. The modifications (anchor an alternate emergency power supply transformer and provide clearance around two fire sprinklers located on the 1974' elevation of the control building at grid C4 & CC, and grid C2 & CB) were reported as completed in the supplement letter dated November 21, 2019. Based on the SCDF results, the NRC staff utilized the Callaway 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 SCDF detailed screening concluded that Callaway 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 Callaway SPRA submittal, the technical team determined that recommending Callaway 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 Callaway review to the TRB with the Phase 2 recommendation that Callaway 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 Callaway 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 Callaway 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 Callaway 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.

REGULATORY COMMITMENT

In Attachment one to letter dated August 12, 2019, the licensee proposed regulatory commitments to complete two (2) permanent plant modifications. The NRC staff notes that NEI 99-04 "Guidelines for Managing NRC Commitments" (ADAMS Accession No. ML003680088), as endorsed by the NRC in SECY-00-0045 "Acceptance of NEI 99-04, "Guidelines for Managing NRC Commitments"" (ADAMS Accession No. ML003679799), provides an acceptable method to manage commitments. In its supplement letter dated November 21, 2019 (ADAMS Package Accession No. ML19325D662), the licensee reported that the plant modifications reported in the SPRA submittal were completed. If the credited plant modifications were to be changed, the staff may revisit its conclusion.

CONCLUSION

Based on the staff's review of the Callaway 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/***

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

Docket No. 50-483

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

cc: Distribution via 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 ASME-ANS [American Society of Mechanical Engineers-American Nuclear Society] PRA Methodology Standard (the ASME-ANS Standard). According to the SPID, either the "Addendum A version" (ASME/ANS Addendum A, 2009) or the "Addendum B version" (ASME/ANS Addendum B, 2013) of the ASME-ANS Standard can be used.

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

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

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

The purpose of this staff guidance document (checklist) (NRC, 2018c) 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, 2017b) 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 (NRC, 2012), letter to NEI.

The checklist in this document is comprised of the 16 "Topics" that require additional staff guidance because the SPID contains specific guidance that differs from the Code Case Standard or expands on it. Each is covered below under its own heading, "Topic 1," "2," etc. The checklist also includes the SR table at the end that was cited earlier.

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 Access and Documents Management System (ADAMS) Accession No. ML16350A181).

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

- Topic 2: Site Seismic Response (SPID Section 2.4)

- Topic 3: Definition of the Control Point for the SSE [Safe Shutdown Earthquake] - to- GMRS [Ground Motion Response Spectra] - Comparison Aspect of the Site Analysis (SPID Section 2.4.2)

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

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

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

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

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

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

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

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

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

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

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

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

- Topic 16:  Review of Plant Modifications and Licensee Actions

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

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

**TOPIC 2: Site Seismic Response (SPID Section 2.4)**

| | |
|---|---|
| The site under review has updated/revised its site response analysis from what was submitted to NRC in response to the NTTF Recommendation 2.1: Seismic 50.54(f) letter. | Yes |
| Notes from staff reviewer:<br><br>The licensee [Ameren Missouri] updated the site seismic response to account for the removal of glacial till and the installation of compacted backfill beneath the nuclear island of the Callaway Plant, Unit 1 (Callaway). Guidance in the SPID was followed for the development of the revised site response.<br><br>Deviation(s) or deficiency(ies) and Resolution: N/A<br><br>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.<br><br>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.<br><br>The SPID (Section 2.4.1) recommends one of two approaches for establishing the control point for a logical SSE-to-GMRS comparison: | |
| A) If the SSE control point(s) is defined in the final safety analysis report (FSAR), it should be used as defined. | No |
| B) If the SSE control point is not defined in the FSAR, one of three criteria in the SPID (Section 2.4.1) should be used. | Yes |
| C) An alternative method has been used for this site. | N/A |
| The control point used as input for the SPRA is identical to the control point used to establish the GMRS and previously accepted by the staff. | Yes |
| If yes, the control point can be used in the SPRA and the NRC staff's earlier acceptance governs. | |
| If no, the NRC staff's previous reviews might not apply. The staff's review of the control point used in the SPRA is acceptable. | N/A |
| Notes from staff reviewer: None<br><br>Deviation(s) or deficiency(ies) and Resolution: N/A<br><br>Consequence(s): N/A | |

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

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

| | |
|---|---|
| The NRC staff review of the structural model finds an acceptable demonstration of its adequacy | |
| Used an existing structural model | No |
| Used an enhancement of an existing model | Yes |
| Used an entirely new model | Yes |
| Criteria 1 through 7 (SPID Section 6.3.1) are all met. | Yes |
| Notes from staff reviewer:<br><br>During the audit, the NRC staff reviewed the March 2019 F&O closure report for the SPRA and the licensee's fragility analysis reports as well as the submittal. These reports indicate that the structural modeling performed in aid of the fragility analyses presented in the submittal was predominantly done using lumped-mass stick models (LMSMs). A 3D finite element building model was performed for one building, the Auxiliary Control Building (ACB). Based on the dispositions to F&Os, the LMSMs meet criteria 1-7 outlined in the SPID Section 6.3.1.<br><br>In dispositions presented in the F&O closure report for the SPRA, the reviewers state that given an update to the licensee's analysis specific concerns about the LMSMs have been resolved and these F&Os are closed by the independent assessment team.<br><br>The licensee's fragility analysis report and dispositions provided in the F&O closure report for the SPRA (for F&O 23-5 and 23-7) indicate that the licensee's 3D finite element building modeling approach is acceptable.<br><br>The NRC staff finds the modeling approach adequate because it meets the intent of the guidance for performing detailed structural modeling and because the F&Os associated with the modeling were resolved by the licensee and closed by the independent assessment team.<br><br>Deviation(s) or deficiency(ies) and Resolution: None.<br><br>Consequence(s): N/A | |
| The NRC staff concludes that:<br><br>• The peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to the SR requirement SFR-B3 in the Code Case Standard, as well as to the requirements in the SPID. | Yes |

| | |
|---|---|
| • Although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis. | N/A |
| • The licensee's structural model meets the intent of the SPID guidance. | Yes |
| • The licensee's structural model does not meet the intent of the SPID guidance, but is acceptable on another justified basis. | N/A |

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

| | |
|---|---|
| Fixed-based dynamic seismic analysis of structures was used, for sites previously defined as "rock."<br><br>If no, this issue is moot.<br><br>If yes, on which structure(s)?<br>Structure name: West Penetration Room (Unit 3)<br><br>Structure #1:<br>If used, is $V_S$ > about 5,000 feet (ft.)/second (sec.)?<br><br>If 3,500 ft./sec. < $V_S$ < 5,000, was peak-broadening or peak shifting used?<br><br>Potential Staff Finding:<br>The demonstration of the appropriateness of using this approach is adequate. | No<br><br><br><br><br><br><br><br><br>N/A |
| Notes from staff reviewer:<br><br>According to the SPRA submittal, page 35, "All major Callaway structures are founded on soil. Therefore, fixed-base analyses were not applicable. It is noted however, that intermediate fixed-base modal analyses were performed to check the modeling fidelity when LMSM were recreated from design models, as well as for validation purposes."<br><br>The NRC staff review confirmed that the fixed-base analyses were only used as an intermediate check on the LMSMs.<br><br>Deviation(s) or deficiency(ies) and Resolution: None.<br><br>Consequence(s): N/A | |
| The NRC staff concludes that:<br><br>• 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.<br><br>• Although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis | Yes<br><br><br><br><br><br>N/A |

| | |
|---|---|
| • The licensee's use of fixed-based dynamic analysis of structures for a site previously defined as "rock" adequately meets the intent of the SPID guidance. | N/A |
| • The licensee's use of fixed-based dynamic analysis of structures for a site previously defined as "rock" does not meet the intent of the SPID guidance, but is acceptable on another justified basis. | N/A |

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

| | |
|---|---|
| Seismic response scaling was used.<br><br>If no, this issue is moot.<br><br>If yes, on which structure(s)?<br><br>Potential Staff Findings:<br>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.<br><br>If the shape is not similar, the justification for seismic response scaling is adequate.<br><br>Consideration of non-linear effects is adequate. | No<br><br><br><br><br><br><br>N/A<br><br>N/A<br><br>N/A |
| Notes from staff reviewer:<br><br>During the audit, the NRC staff reviewed the March 2019 F&O closure report for the SPRA and the licensee's fragility analysis reports as well as the submittal. The submittal, the licensee's analysis reports, and dispositions presented in the F&O closure report for the SPRA all indicate that scaling of the ISRS to account for higher ground motion levels was not performed. The licensee states in its fragility analysis report that "Scaling analysis was not used." | |
| Deviation(s) or deficiency(ies) and Resolution: None.<br><br>Consequence(s): N/A | |
| The NRC staff concludes that:<br><br>• 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.<br><br>• Although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.<br><br>• The licensee's use of seismic response scaling adequately meets the intent of the SPID guidance.<br><br>• The licensee's use of seismic response scaling does not meet the intent of the SPID guidance but is acceptable on another justified basis. | <br><br>N/A<br><br><br><br><br><br>N/A<br><br><br>N/A<br><br><br>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.<br><br>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 (SSI) analysis (for soil sites), the analysis of how the seismic energy enters the base level of a given building, and the in-structure-response-spectrum (ISRS) analysis. Said another way, an acceptable SPRA must use these analysis pieces together in a consistent way.<br><br>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.<br><br>GMRS corresponds to ground motions at plant grade, elevation 840 ft., which is the Callaway surface control point.<br><br><u>Elevation 829 ft:</u><br>Reactor Building (RB)<br>Diesel Generator Building (DGB)<br>Ultimate Heat Sink Cooling Tower (UHSCT)<br><br><u>Elevation 808.5 ft:</u><br>Auxiliary/Control Building (ACB)<br>Emergency Service Water Pumphouse (ESWP)<br><br><u>Elevation 832 ft:</u><br>Alternate Emergency Power System | |

| | |
|---|---|
| Are all structures appropriately considered?<br><br>2. Are models adequate to provide realistic structural loads and response spectra for use in the SPRA?<br><br>1. Is the SSI analysis capable of capturing uncertainties and realistic?<br>2. Is the probabilistic response analysis capable of providing the full distribution of the responses? | Yes<br><br>Yes<br><br><br>Yes<br><br>Yes |

Notes from staff reviewer:

According to the SPRA submittal (page 36), "All major structures are founded on or embedded in soil and therefore an SSI analysis was required for a realistic estimate of response." Also, the licensee indicated that all the buildings' SSI calculations are included in a fragility analysis report. During the audit, NRC staff reviewed this report and notes that references provided in this report are given for the source of the soil layer properties and associated calculations for both for the 3D finite element models and the LMSMs. Relevant F&Os were resolved by the licensee and closed by the independent assessment team. Based on its review, the NRC staff finds the approach to be acceptable.

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

Consequence(s): N/A

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

| | |
|---|---|
| • 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)**

<table>
<tr><td>The selection of SSCs for seismic fragility analysis used a screening approach by capacity following Section 6.4.3 of the SPID.<br><br>If <u>no</u>, see items D and E.<br><br>If <u>yes, see items A, B, and C.</u><br><br><u>Potential Staff Findings</u>:<br><br>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.<br><br>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.<br><br>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.<br><br>D) The Standard has been followed.<br><br>E) An alternative method has been used and its use has been appropriately justified.</td><td>Yes<br><br><br><br><br><br><br><br><br>Yes<br><br><br><br>Yes<br><br><br><br>Yes<br><br><br><br><br>N/A<br><br>N/A</td></tr>
<tr><td colspan="2">Notes from staff reviewer:<br><br>During the audit, the NRC staff reviewed the March 2019 F&O closure report for the SPRA and the licensee's fragility analysis reports as well as the submittal. According to the submittal Section 4.4.1.1, "the methods in EPRI reports NP-6041-SL [31], TR-1019200 [38] and TR-103959 [39] were used for calculation of seismic fragility parameters." Section 4.4.1.2 of the submittal describes the steps used to apply the CDFM to screen-in SSCs and references the SPID approach.<br><br>The licensee's fragility analysis report discusses this treatment and references the SPID approach. These descriptions indicate that the guidance outlined in Section 6.4.3 of the SPID was followed.<br><br>Deviation(s) or deficiency(ies) and Resolution: None.<br><br>Consequence(s): None</td></tr>
<tr><td>The NRC staff concludes:<br><br>• 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</td><td>Yes</td></tr>
</table>

| | |
|---|---|
| SR requirements SFR-C1, SFR-C2, and SPR-B5 in the Code Case Standard, as well as to the requirements in the SPID. | |
| • Although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis. | N/A |
| • The licensee's use of a screening approach for selecting SSCs for fragility analysis meets the intent of the SPID guidance. | Yes |
| • The licensee's use of a screening approach for selecting SSCs for fragility analysis does not meet the intent of the SPID guidance but is acceptable on another justified basis. | N/A |

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

| | |
|---|---|
| The CDFM/Hybrid method was used for seismic fragility analysis.<br><br>If <u>no</u>, See item C) below and next issue.<br><br>If <u>yes</u>:<br><br><u>Potential Staff Findings</u>:<br>A) The recommendations in Section 6.4.1 of the SPID were followed appropriately for developing the CDFM High Confidence Low Probability of Failure (HCLPF) capacities. | Yes<br><br><br><br><br><br><br>Yes |
| B) The Hybrid methodology in Section 6.4.1 and Table 6-2 of the SPID was used appropriately for developing the full seismic fragility curves. | Yes |
| C) An alternative method has been used appropriately for developing full seismic fragility curves. | N/A |
| Notes from staff reviewer:<br><br>During the audit, the NRC staff reviewed the March 2019 F&O closure report for the SPRA and the licensee's fragility analysis reports as well as the submittal. The submittal states that the CDFM/Hybrid method was used for the fragility analysis. However, the submittal states that as a refinement for important contributors a scaling approach was also applied that is not described in Section 6.4.1 of the SPID. The licensee states in Section 4.4.1.2 of the submittal that, for the refined approach, it used guidance provided in Section 3.4 of EPRI TR-1019200, "Seismic Fragility Applications Guide Update," which is guidance that Table 6-1 of the SPID recommends. In addition, the submittal states that recommended values for $\beta c$ and $\beta r$ from Table 6-2 of the SPID was used.<br><br>The peer review findings have been addressed. The relevant peer review finding, F&O 23-9, relate to the requirements in the SPID. The finding states, "The RB Containment fragility is based on scaling the design basis load combination involving DL, OBE and Internal Pressure. This would result in an unrealistic estimate of fragility for GMRS." To resolve this finding, the licensee has removed conservatism from the reactor building fragility analysis. Realistic soil strain properties and parameters were considered, resulting in a more refined HCLPF. This F&O has been closed by the independent assessment team.<br><br>Deviation(s) or deficiency(ies) and Resolution: None.<br><br>Consequence(s): N/A | |

| | |
|---|---|
| The NRC staff concludes that:<br><br>• 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.<br><br>• Although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.<br><br>• The licensee's use of the CDFM/Hybrid method for seismic fragility analysis meets the intent of the SPID guidance.<br><br>• 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 | <br><br>Yes<br><br><br><br><br>N/A<br><br><br>Yes<br><br><br>N/A |

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

| | |
|---|---|
| The SPID requires that certain SSCs that are sensitive to high frequency seismic motion must be analyzed in the SPRA for their seismic fragility using a methodology described in Section 6.4.2 of the SPID.<br><br>Potential Staff Findings:<br>The NRC staff review of the SPRA's fragility analysis of SSCs sensitive to high frequency seismic motion finds that the analysis is acceptable. | Yes |
| The flow chart in Figure 6-7 of the SPID was followed. | Yes |
| The flow chart was not followed but the analysis is acceptable on another justified basis. | N/A |
| Notes from staff reviewer:<br><br>According to Section 4.1.2 of the submittal, the licensee's fragility analysis of SSCs sensitive to high frequency seismic motion meets the intent of the SPID guidance, and follows Figure 6-7 of the SPID. An evaluation of spurious trips of breakers were performed. The main types of breakers evaluated are air breakers, drawout type breakers, and molded case circuit breakers.<br><br>Molded case circuit breakers are described as inherently high capacity items. The switchgear fragility that houses the breakers was evaluated using CDFM criteria defined in EPRI NP-6041-SL, Revision 1, "A Methodology for Assessment of Nuclear Plant Seismic Margin," which addresses the fragility for high frequency sensitive components as discussed in Section 6.4.2 of the SPID.<br><br>Deviation(s) or deficiency(ies) and Resolution: None<br><br>Consequence(s): N/A | |
| The NRC staff concludes that: | |
| • The peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to the SR SFR-E5 in the Code Case Standard, as well as to the requirements in the SPID. | N/A |
| • Although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis. | N/A |
| • The licensee's fragility analysis of SSCs sensitive to high frequency seismic motion meets the intent of the SPID guidance. | Yes |

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

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

| | |
|---|---|
| The SPID requires that certain relays and related devices (generically, "relays") that are sensitive to high-frequency seismic motion must be analyzed in the SPRA for their seismic fragility. Although following the Standard is generally acceptable for the fragility analysis of these components, the SPID (Section 6.4.2) contains additional guidance when either circuit analysis or operator-action analysis is used as part of the SPRA to understand a given relay's role in plant safety. When one or both of these are used, the NRC reviewer should use the following elements of the checklist. | |
| i) <u>Circuit analysis</u>: The seismic relay-chatter analysis of some relays relies on circuit analysis to assure that safety is maintained.<br>(A) If <u>no</u>, then (B) is moot.<br>(B) If <u>yes:</u><br><u>Potential Staff Finding</u>:<br>The approach to circuit analysis for maintaining safety after seismic relay chatter is acceptable. | Yes<br><br><br><br><br>Yes |
| ii) <u>Operator actions</u>: The relay-chatter analysis of some relays relies on operator actions to assure that safety is maintained.<br>(A) If <u>no</u>, then (B) is moot.<br>(B) If <u>yes:</u><br><u>Potential Staff Finding</u>:<br>The approach to analyzing operator actions for maintaining safety after seismic relay chatter is acceptable. | Yes<br><br><br><br><br>Yes |
| Notes from staff reviewer:<br><br>During the audit, the NRC staff reviewed the March 2019 F&O closure report for the SPRA and the licensee's fragility analysis reports. According to Section 4.1.2 of the submittal, "An extensive relay chatter evaluation was performed for the CEC S-PRA, in accordance with [the SPID]." By performing this evaluation in accordance with the SPID, circuit analysis has been performed.<br><br>The F&Os related to the approach for analyzing operator actions after seismic relay chatter were closed by the independent assessment team. Specifically, the independent assessment team stated in the resolution of F&O 19-11 presented in the F&O closure report that the Modeling Notebook now addresses how spurious indication from relay chatter is handled implicitly in the use of the EPRI HRA method. | |

| | |
|---|---|
| The peer review findings have been addressed and the analysis approach is acceptable by the staff for this SPRA submittal. The relevant peer review findings related to assessing the impact of relay chatter (against SR SPR B6 and the requirements in the SPID) are F&Os 25-7, 25-9, and 19-6. These F&Os were addressed by the licensee by updating the SPRA documentation and correcting errors in the modeling.<br><br>Deviation(s) or deficiency(ies) and Resolution: None<br><br>Consequence(s): N/A | |
| The NRC staff concludes that: | |
| • the peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The relevant peer review findings are those that relate to SR SPR-B6 in the Code Case Standard, as well as to the requirements in the SPID. | Yes |
| • although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis. | N/A |
| • the licensee's analysis of seismic relay-chatter effects meets the intent of the SPID guidance. | Yes |
| • the licensee's analysis of seismic relay-chatter effects does not meet the intent of the SPID guidance, but is acceptable on another justified basis. | N/A |

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

| | |
|---|---|
| The CDFM methodology has been used in the SPRA for analysis of the bulk of the SSCs requiring seismic fragility analysis.<br><br>If no, the staff review will concentrate on how the fragility analysis was performed, to support one or the other of the “potential staff findings” noted just below.<br><br>If yes, significant risk contributors for which use of separation of variables (SOV) fragility calculations would make a significant difference in the SPRA results have been selected for SOV calculations.” | Yes |
| Potential Staff Findings:<br>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. | No |
| B) The recommendations in Section 6.4.1 were not followed, but the analysis is acceptable on another justified basis. | Yes |

Notes from staff reviewer:

Section 4.4 of the SPRA submittal states that the level of detail used in the fragility analysis for an SSC was based the licensee’s best understanding of its importance to the plant seismic response. Fragility calculations were performed in stages making use of feedback from quantification of the plant response model. Seismic analyses were initially performed using the CDFM method as described in EPRI NP-6041-SL. Based on the results of this initial modeling, refined CDFM analyses were performed for dominant contributors to seismic risk. If the SSC remained risk dominant, the fragility analysis was further refined by using the separation of variables (SOV) method. The NRC staff reviewed Table 5-3 and Table 5-7, which presents the Fussell-Vesely (F-V) importance values for highest risk events and Fragility Groups used in the SPRA. The NRC staff noticed that the bulk of the values in these listings resulted from fragility modelling based on the CDFM or a refined CDFM method. However, the listing also contained importance values based on using the SOV method and generic fragilities. Specifically, Relay Fragility Group “Relay_0.18DG” and the Seismic-Induced Failure of the Steam Generator Supports (SF-NSSG) were analyzed with the SOV methodology.

Concerning generic fragilities Section 5.7.4 of the submittal states that non-safety component basic events were assigned a generic fragility value and were assumed to be fully correlated. This treatment is generally regarded to be conservative. However, to ensure that the impact of crediting non-safety equipment with a generic fragility is not a significant contributor to seismic risk the licensee performed a sensitivity study to confirm this assumption. The NRC staff noted that an exception to this conclusion concerns the fragility analysis for Seismic-Induced Loss of Offsite Power (SF-IE-T1). SF-IE-T1 is listed as having a high CDF F-V importance value, but its fragility analysis was performed using generic fragility values. However, the NRC staff observes that this fragility group consists of highly distributed equipment largely not under plant control

| | |
|---|---|
| making more refined analysis challenging. Also, the NRC staff observes that the licensee's approach is consistent with the state-of-practice.<br><br>Concerning the use of refined CDFM analysis (e.g., using seismic test data), Section 6.4.1 of the SPID does not refer to use of refined CDFM analysis before the SOV fragility analysis approach is used. However, the premise of using more refined fragility analysis for removing conservatism and lowering the importance of specific events and fragility groups is met using this approach.<br><br>Concerning SR SFR-E3 on estimating seismic fragilities, Section A.6.7 of the submittal states that SFR-E3 was reassessed during the F&O closure review and determined to be Met at Capability Category (CC) II.<br><br>Deviation(s) or deficiency(ies) and Resolution: None<br><br>Consequence(s): N/A | |
| The NRC staff concludes: | |
| • the peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to SFR-E3 and the requirements in the SPID. | Yes |
| • although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis. | N/A |
| • the licensee's method for selecting the "dominant risk contributors" for further seismic fragilities analysis using the separation-of-variables methodology meets the intent of the SPID guidance. | 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)**

<table>
<tr><td colspan="2">The NRC staff review of the SPRA's analysis of LERF finds an acceptable demonstration of its adequacy.<br><br><u>Potential Staff Findings</u>:<br>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<br><br>B) The LERF analysis does not follow the guidance in Table 6-3 but the analysis is acceptable on another justified basis. — N/A</td></tr>
<tr><td colspan="2">Notes from staff reviewer:<br><br>Section 4.1 of the submittal states that a seismic equipment list (SEL) was developed for Callaway's end states of core damage and large early release. The submittal states that SSCs include seismic failures that could either cause an initiating event or degrade the capability of the plant to mitigate initiating event. The SEL forms the basis for the seismic fragility and system analysis tasks.<br><br>Section 5.1.5 states that level 2 modeling developed for the internal events PRA was also used for the SPRA because the same failures and radioactive release phenomenology apply to both PRAs. The submittal states that these components were included in the SEL and therefore were included in the fragility analysis. The submittal states that containment penetrations are treated as correlated and are assigned to one fragility group (SF RB-PEN). The submittal also states that failure of containment penetrations is modeled separately from Reactor Building collapse associated with Reactor Building equipment hatches<br><br>Section 3.0 of the submittal states that the results of the SPRA are based on CDF and LERF. Table 5-7 of the submittal presents the important events and fragility groups that contribute to seismic LERF. Table 5-6 shows that the top 10 cutsets all include failure of containment isolation due to the failure of containment penetrations which leads to seismic LERF.<br><br>Deviation(s) or deficiency(ies) and Resolution:<br><br>Consequence(s): N/A</td></tr>
<tr><td>The NRC staff concludes that:<br><br>• 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.</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 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.<br><br>Potential Staff Findings:<br>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).<br><br>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).<br><br>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).<br><br>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).<br><br>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:<br>• the SPRA findings should be based on a consensus process, and not based on a single peer review team member<br>• a final review by the entire peer review team must occur after the completion of the SPRA project<br>• an "in-process" peer review must assure that peer reviewers remain independent throughout the SPRA development activity. | Yes<br><br>Yes<br><br>Yes<br><br>Yes<br><br>N/A |
| If no, go to (F).<br><br>If yes, the "in process" peer review approach is acceptable. Go to (G). | |

| | |
|---|---|
| E) The "end-of-process" peer review process followed the peer review guidance in the SPID (Section 6.7) as supplemented by NRC staff comments in the NRC letter dated March 7, 2018 (NRC, 2018a, 2018b).<br><br>If no, go to (F).<br><br>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.

An SPRA peer review was conducted by the Pressurized Water Owners Group (PWROG) in June 2018 against the CC-II supporting requirements of PRA Standard ASME/ANS RA-S Case 1 (ASME/ANS, 2017) and associated NRC clarifications in a letter dated March 12, 2018 (NRC, 2018). ASME/ANS RA-S Case 1 has been approved by the NRC for use in regulatory applications, subject to conditions and clarifications (NRC, 2018).

The peer review team utilized the process defined in NEI 12-13 (NEI 2012). Though not explained in the SPRA submittal, the NRC staff's review of the SPRA peer review report during the audit found that the peer review used the peer-review guidance in NEI 12-13 considering the comments from the NRC staff in letter dated March 7, 2018 letter. All elements of the SPRA were peer reviewed, including those identified in Section 6.7 of the SPID. The full scope SPRA peer review performed in 2018 resulted in 52 Finding-level F&Os.

The SPRA submittal states that the peer review lead and reviewer qualifications were confirmed by Ameren to be consistent with the requirements in Section 1-6.2.2 of the ASME/ANS PRA Standard (ASME/ANS Addendum B 2013 and Case 1 2017) and the guidelines of NEI 12-13 (NEI ,2012). The submittal states that members of the peer review team were independent of development of the Callaway SPRA, and therefore not involved in performing or directing work on any PRA element in the overall SPRA. The submittal states that, for each PRA technical element, a team of at least two were assigned, one having lead responsibility for that area

The submittal does not specify if any of the peer review members focusing on seismic response and fragility analysis have successfully completed the seismic qualification

<table>
<tr><td colspan="2">user group (SQUG) Walkdown Screening and Seismic Evaluation training course or equivalent. During the audit, the NRC staff reviewed the detailed resumes for the peer reviewers provided in the SPRA peer review report. Two of the three reviewers that focused on review of the fragility analysis (SFR technical element) had significant SQUG experience. However, neither of the two reviewers to plant response F&Os (SPR technical element) were identified as having had SQUG training. However, the resumes for each of these peer reviewers were shown to demonstrate significant SPRA experience, which is judged by the NRC staff to be “equivalent” to the SQUG training.<br><br>An SPRA F&O closure review was performed in March 2019 using the independent assessment process outlined in Appendix X (NEI, 2017) to NEI 12-13 (NEI, 2012) along with the conditions specified in NRC acceptance letter dated May 3, 2017 (NRC, 2017a). The F&O closure review reviewed the dispositions to the Finding level F&Os from the full scope SPRA peer review performed in June 2018. The peer review was performed against the Code Case for ASME/ANS RA-Sb-2013 (ASME/ANS, 2017). The submittal states that all F&Os were determined to be Met or Met at CC II and were, therefore, closed except for two seismic F&Os. These two seismic F&Os (i.e., F&Os 25-12 and 25-19) concerned open internal events F&Os associated with the internal events PRA model on which the SPRA was based. However, during the audit the licensee explained that an internal events PRA closure review was performed in November 2019 using the independent assessment process outlined in Appendix X (NEI, 2017) along with the NRC staff comments in staff letter dated May 3, 2017 (NRC, 2017a). The review was performed against ASME/ANS RA-Sa-2009 (ASME/ANS Addendum A, 2009) and RG 1.200, Revision 2 (NRC, 2009). As a result of this F&O closure review all internal events F&Os were closed except F&Os 13-1 and 22-3.<br><br>During the audit, the licensee explained that the internal events PRA model updated to resolve F&Os for the 2019 internal event events F&O closure review was used to update the SPRA. The supplement dated July 10, 2020 (ADAMS Accession No. ML20192A244) provides an updated SPRA report with regenerated risk results based on the updated internal events PRA.<br><br>A second F&O closure review for the SPRA was performed in June 2020 in combination with an internal events and fire F&O closure review. The combined seismic, internal events, and fire F&O closure review was performed using the independent assessment process outlined in Appendix X (NEI, 2017) along with the conditions specified in NRC acceptance letter dated May 3, 2017 (NRC, 2017a). As a result of this F&O closure review all remaining seismic, internal events, and fire F&Os were closed.<br><br>Deviation(s) or deficiency(ies) and Resolution: None<br><br>Consequence(s): N/A</td></tr>
<tr><td>The NRC staff concludes:<br><br>• the licensee’s peer-review process meets the intent of the SPID guidance as supplemented by NRC staff comments in the NRC letter dated March 7, 2018 (NRC, 2018a, 2018b).</td><td>Yes</td></tr>
</table>

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

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

| | |
|---|---|
| The NRC staff review of the SPRA's documentation as submitted finds an acceptable demonstration of its adequacy.<br><br>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<br><br><br>Yes |

Notes from staff reviewer:

Tables 2-1 and 2-2 of the SPRA submittal provide a cross-reference of information required by 10 CFR 50.54(f) and specified in Section 6.8 of the SPID to the sections of the submittal where the information can be found. The level-of-detail of the information provided appears to be generally consistent with that specified in Section 6.8 of the SPID. The SPID requires that there should be sufficient information provided so that the results to all key aspects of the analysis can be assessed. Section 5.3.2 of the submittal identifies and discusses the key assumptions and sources of model uncertainty for SPRA. Sections 5.4 and 5.5 of the submittal presents and discusses the SPRA CDF and LERF quantification results. Section 5.6 of the submittal presents the parametric data uncertainty analysis results including the total SCDF and LERF point estimate, mean, 5th percentile, median, and 95th percentile. Section 5.7 of the submittal discusses sensitivity studies and presents results on parameters that could be important to regulatory decisions that could result from the submittal. The NRC staff notes that the submittal does not refer to or describe pertinent information from the site's Plant Examination of External Events (IPEEE) program as suggested in the SPID (e.g., all functional/systemic event trees). However, the NRC staff also notes that the SPID only identifies this IPEEE information as guidance for consideration in the 50.54(f) response.

Section 6.8 of the SPID states that level of detail needed in the submittal should be sufficient to enable NRC to understand and determine the validity of all input data and calculation models used, to assess the sensitivity of the results to all key aspects of the analysis.

During the audit, the licensee clarified the extent to which FLEX equipment and actions are credited in the SPRA that support the submittal. The licensee explained that the Callaway SPRA credits two FLEX strategies: (1) FLEX steam generator (SG) makeup pumps, and (2) 480VAC portable backup generators supplying power to the battery chargers for 125 VDC buses NKO1, NKO2, and NKO4. Failure of each of these two FLEX strategies are represented by single basic events (i.e., FLEXAFWFAIL and FLEXACTODCFAIL) that are set conservatively at a failure probability of 0.99. Though the strategies involve use of equipment and operator actions, no FLEX equipment or operator failures are modelled. The licensee also explained that it performed a sensitivity analysis on this FLEX credit by reducing the failure probability of the two basic events (FLEXAFWFAIL and FLEXACTODCFAIL) to 0.1. The results of the sensitivity study show that the seismic CDF decreased by 11.47% and the seismic LERF decreased by 1.38%. The NRC staff notes that this conservative modeling is sufficient for this submittal because it does not impact the conclusions of the SPRA.

During the audit, the NRC staff noted that the F-V importance measure values derived from the SPRA quantification report were inconsistent with the values presented in the submittal or supplement. During the audit, the licensee explained that the Callaway

SPRA quantification report presents two sets of importance measure results. The licensee stated it considered the first set of importance values to be "*overestimated*" and the second set "more realistic," and therefore, the second set of importance values were presented in the SPRA supplement dated July 10, 2020 (ADAMS Accession No. ML20192A244). The NRC staff notes that the approach for determining the first set of importance values is identical to that followed by licensees in other SPRA submittals. The licensee explained that rather than setting the seismic failure probability of a particular fragility group to zero, it set the fragility of the group to "completely rugged" and calculated a non-zero seismic failure probability for that capacity at each ground motion.

The NRC staff understands that the plant modifications to decrease seismic risk likely involve increasing the seismic capacity of risk-important components and are not expected to decrease the probability of those failures to zero. The NRC staff also notes that the F-V importance values determined by the two approaches for the top risk contributors are not significantly different. The NRC staff found the difference to be less than 10% for any given failure. The NRC staff finds that the licensee's approach to calculate SPRA importance values is acceptable for this submittal because the results of using the alternative approach does not change the NRC staff's conclusions on the updated SPRA supplement dated July 10, 2020. As a result, the NRC staff did not make a conclusion on whether the licensee's approach for calculating the F-V importance measures from its SPRA is "more realistic."

The NRC staff observed from the sensitivity study results presented in Section 5.7.1 of the SPRA supplement dated July 10, 2020, that the increase in LERF for the last decade decrease in truncation level was about 50% for three hazard intervals and 38% for one hazard interval. The NRC staff made similar observations about increases in SCDF, but also noted that the results seemed to be based on incomplete quantification results. During the audit, the licensee explained that the uncertainty associated with the truncation levels used in the quantification of the updated SPRA does not impact the conclusions of the submittal by providing the results of a more complete ACUBE quantification. The results show that the percent change in SCDF against the total SCDF for one decade decrease in truncation level is less than 5% for each of the ten hazard intervals. This result is also true for SLERF except for two hazard intervals in which the percent increase is slightly higher that 5%. The NRC staff finds that the PRA model truncation level used for the hazard interval provides sufficient convergence given that it is in alignment with the PRA Standard Supporting Requirement QU-B3. Therefore, truncation levels do not impact the NRC's decision for this submittal.

The NRC staff noted that the supplement dated July 10, 2020, did not present the updated results of certain sensitivity studies that were presented in the original SPRA submittal dated August 12, 2019. One of these sensitivity studies concerned how the Human Reliability Analysis (HRA) bins (which were used to assign increased operator error probabilities based on the magnitude of the seismic event) are distributed across the seismic hazard bins. During the audit, the licensee explained that the original sensitivity results should still apply to the updated SPRA results. The earlier results showed that the impact was negligible for reasonable changes in the distribution of the HRA bins across the hazard intervals. The NRC staff finds that this rationale, in combination with the fact that during the audit the licensee provided the results of a study on F-V importance values for human errors acceptable as an updated sensitivity

| | |
|---|---|
| study on the HRA bin, is very unlikely to change the NRC staff's decision on this submittal.<br><br>Another sensitivity study not updated for the updated SPRA submittal concerned the fragility parameters for three fragility groups SF-IE-S3, SF-RL0XX, and SF-NN0X. During the audit, the licensee explained that the original sensitivity study was performed to understand whether these fragilities should be refined. The licensee explained that, as a result of the original sensitivity study, the cited fragility parameters were refined and incorporated into the July 10, 2020, SPRA supplement, and therefore, these sensitivity studies did not need to be repeated.<br><br>The final sensitivity not reproduced for the July 10, 2020, SPRA supplement regards the impact of ex-control operator error human error probabilities (HEPs) on the SCDF and SLERF. During the audit, the licensee identified the ten ex-control actions modeled in the SPRA and provided the results of a sensitivity study showing the aggregate impact on SCDF and SLERF when these failure actions are set to TRUE (i.e., all actions are assumed to fail). The results show that the seismic CDF increased by 3.8% and the LERF increased by 1.5%. Therefore, the NRC staff finds that the uncertainty associated with determining the HEPs for ex-control actions does not impact the NRC's staff decision on this submittal.<br><br>The NRC staff noticed inconsistencies in the point estimate SCDF value reported in the July 10, 2020 supplement. The point estimate SCDF is mentioned as 5.59E-5 /reactor-year (/rx-yr) in Section 5.6. However, in Table 5-8 it states that the point estimate is 1.75E-5/rx-yr. Similarly, Section 5.5 states that the point estimate for SLERF is 2.90E-6/rx-yr but Table 5-8 states that it is 4.72E-06/rx-yr. Based on the figures in Section 5.6, the value in Section 5.6 for SCDF and that in Table 5-8 for SLERF appear to be the correct point estimates. During the audit, the licensee confirmed that the correct point estimate SCDF and SLERF are 5.59E-6/rx-yr and 2.90E-6/rx-yr, respectively.<br><br>Deviation(s) or deficiency(ies) and Resolution: None<br><br>Consequence(s): NA | |
| The NRC staff concludes:<br><br>• 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.<br><br>• The licensee's documentation does not meet the intent of the SPID guidance but is acceptable on another justified basis. | Yes<br><br><br><br><br>N/A |

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

| | |
|---|---|
| The licensee:<br>• identified modifications necessary to achieve seismic risk improvements<br><br>• provided a schedule to implement such modifications (if any), consistent with the intent of the guidance<br><br>• provided Regulatory Commitment to complete modifications<br><br>• provided Regulatory Commitment to report completion of modifications. | Yes<br><br>Yes<br><br>Yes<br><br>Yes |
| The licensee:<br>• completed modifications by December 31, 2020<br>• reported completion of modifications in accordance with its regulatory commitment program | |
| Notes from the Reviewer:<br><br>In Attachment 1 of the submittal dated August 12, 2019, the licensee presented two regulatory commitments:<br><br>• Install Anchorage to the Alternate Emergency Power Supply transformer (XPBO5).<br>• Provide a minimum of 1-inch clearance around two fire sprinkler heads located on the 1974' elevation of the control building at grid C4 & CC, and grid C2 & CB.<br><br>In its supplement letter dated November 21, 2019 (ADAMS Package Accession No. ML19325D662), the licensee reported that the plant modifications reported in the SPRA submittal were completed.<br><br>The NRC staff did not identify any cost justified plant improvements based on its review.<br>Refer to Enclosure 2 for the detailed screening evaluation.<br><br>Deviation(s) or Deficiency(ies), and Resolution: None<br><br>Consequence(s): N/A | |
| The NRC staff concludes that the licensee:<br>• identified plant modifications necessary to achieve the appropriate risk profile<br><br>• provided a schedule to implement the modifications (if any) with appropriate consideration of plant risk and outage scheduling | Yes<br><br>Yes |

REFERENCES

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

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

ASME/ANS, 2017: Case 1 for Standard ASME/ANS RA-Sb- 2103, "Standard for Level 1/ Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," American Society of Mechanical Engineers and American Nuclear Society, 2017

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

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

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

NEI, 2017: "Final Revision of Appendix X to NEI 05-04/07-12/12-16, *Close-Out of Facts and Observations (F&Os)*," Nuclear Energy Institute, February 21, 2017 (ADAMS Package Accession No. ML17086A431)

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

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

NRC, 2018: "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 Package Accession No. ML18017A963)

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

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

NRC, 2018c:  “US Nuclear Regulatory Commission Staff Checklist to Support Review of Seismic Probabilistic Risk Assessment Reports Provided in Response to Title 10 of the Code of Federal Regulations 50.54(f) Relating to Seismic Hazard Reevaluations for Recommendations 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident,” July 12, 2019 (ADAMS Accession No. ML18173A017)

**NRC Staff SPRA Submittal Detailed Screening Evaluation**

Introduction

The Callaway Plant, Unit 1 (Callaway) updated Seismic Probabilistic Risk Assessment (SPRA) report (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20192A244) provided in a supplement dated July 10, 2020, indicates that the mean seismic core damage frequency (SCDF) is 7.26E-05/reactor-year (/rx-yr) and the mean seismic large early release frequency (SLERF) is 8.27E-06/rx-yr. During the audit, the licensee explained that after the original Callaway SPRA report dated August 12, 2019 (ADAMS Accession No. ML19225D322) was submitted, the Callaway SPRA was updated to resolve peer review findings. The updated SPRA results are presented in the July 10, 2020, supplement and are the results used in this NRC staff evaluation. The NRC staff compared these SCDF and SLERF values against the guidance in NRC staff memorandum dated August 29, 2017 (ADAMS Accession No. ML17146A200; hereafter referred to as the SPRA Screening Guidance), 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", 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).

The NRC staff found that because the SCDF and SLERF for Callaway 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 Callaway should be considered a Group 1 plant because:

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

- Sufficient reductions in SCDF and 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.

Ameren Missouri (Ameren, the licensee), in performing its seismic analysis in response to the Near-Term Task Force (NTTF) Recommendation 2.1: Seismic, 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 Callaway Plant license, or the rules and orders of the Commission. However, in Attachment 1 of its submittal dated August 12, 2019, Ameren presents two regulatory commitments:

- Install Anchorage to the Alternate Emergency Power Supply transformer (XPBO5).
- Provide a minimum of 1-inch clearance around two fire sprinkler heads located on the 1974' elevation of the control building at grid C4 & CC, and grid C2 & CB.

In Attachment 1 of supplement dated November 21, 2019 (ADAMS Package Accession No. ML19325D662), the licensee updated its regulatory commitment to confirm that both the modifications mentioned above were completed.

Detailed Screening

The detailed screening uses information provided in the Callaway SPRA report, supplements, and supporting information reviewed via audit, 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) 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 Callaway SPRA 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 *Docket Number 50-483 Callaway Plant Unit 1 Union Electric Co Facility Operating License NPF-30 - Application for Renewed Operating License (LDCN 11-0022)*, dated December 15, 2011 (ADAMS Accession No. ML113530374), was used to calculate the RRW threshold. For this analysis, the NRC staff determined the RRW threshold from the SCDF-based MACR to be 1.049. The MACR calculation includes estimation of offsite exposures and offsite property damage, which captures the impact of SLERF. Therefore, separate SLERF-based MACR calculations were not performed. The target RRW values (as defined by the SPRA screening guidance) based on the mean and 95th percentile SCDF and SLERF were calculated by the NRC staff to be between 1.04 and 1.16.

Section 5 of the updated Callaway SPRA report provided in the supplement dated July 10, 2020, included tables and description of events and fragility groups that are the most significant seismic failure contributors to SCDF and SLERF. The descriptions of the significant contributors included the F-V for each. The NRC staff used the F-V values presented in the supplement to calculate the RRW and the contribution to SCDF or SLERF of each contributor. During the audit, the licensee also provided the F-V values for operator errors for the updated SPRA model associated with the supplement dated July 10, 2020, to show the sensitivity of the SPRA results to human errors. The results described above are provided in Table 1 for the SCDF contributors and Table 2 for the SLERF contributors. The listed seismic-induced failures that contribute to SCDF and SLERF have an RRW greater than about 1.02 and those that contribute to operator errors during a seismic event have an RRW greater than about 1.01. These tables provide the following information by column: (1) Description of the component, (2) Failure Mode, (3) RRW, and (4) maximum SCDF or SLERF reduction (MCR or MLR) from eliminating the failure.

Based on the F-V values provided in Sections 5.4 and 5.5 of the updated SPRA submittal dated July 10, 2020, elimination or reduction in the contribution of the following fragility group failures appear to have the potential to reduce the SCDF by 1E-05 per year or the SLERF by 1E-06 per year.

- SF-IE-T1 Seismic-Induced Loss of Off-site Power
- SF-SOIL Seismic Soil Failure
- SF-NSSG Seismic-Induced Failure of the Steam Generator Supports
- SF-RB-PEN Seismic-Induced Failure of Reactor Building Penetrations

Elimination of these individual failures would achieve the target risk reduction cited above and, therefore, are evaluated further below. The NRC staff considered but could not identify combinations of two failures that would achieve target risk reduction cited above.

Of the fragility groups above, the highest contributor to the SCDF is the yard centered seismically induced loss of offsite power (SF-IE-T1). Since this fragility group includes equipment outside the plant boundary, extensive upgrades to electrical yard equipment will not, by itself, prove effective in addressing this contributor. Three of the fragility groups listed above contribute to SLERF. Plant improvement associated with the seismic soil failure (SF-SOIL) would consist of extensive upgrades to Safety Category I structure foundations including to the Reactor Building, which is founded in soil and whose catastrophic failure leads to SLERF. Plant improvements associated with seismically-induced failure of the steam generator supports (SF-NSSG) would involve major hardware upgrades to safety related equipment inside containment.

Improvements associated with seismically-induced failure of Reactor Building penetrations (SF-RB-PEN) would involve major upgrades to Safety Category 1 structures and/or equipment at a number of penetration locations. The NRC staff experience from SAMA analyses on modifications that require temporarily or permanently moving, relocating, or modifying equipment or structures to support modification to an SSC is that the implementation cost is very likely to exceed the calculated MACR for this detailed screening. The NRC staff, therefore, did not pursue further potential improvements associated with SF-IE-T1, SF-SOIL, SF-NSSG, and SF-RB-PEN.

Regarding the human error events, the NRC staff finds no potential to reduce the SCDF by 1E-05 per year or the SLERF by 1E-06 per year by eliminating any single or any pair of human errors based on the F-V values provided by the licensee during the audit (which are presented in Table 1 and 2) for the updated SPRA submittal dated July 10, 2020. Additionally, experience has shown the NRC staff that the risk associated with human errors cannot be significantly reduced by improving operator procedures. Also, hardware solutions are often not possible because the action may require operator diagnoses before the action is taken or the action is already in response to a failed automatic function. Moreover, in this case, implementation of any kind of hardware solutions to eliminate or significantly reduce the probability of multiple human failures is unlikely to be cost justified.

During the audit, the licensee clarified the extent to which FLEX equipment and actions are credited in the SPRA that support the submittal. The licensee explained that the Callaway SPRA credits two FLEX strategies: (1) FLEX steam generator (SG) makeup pumps, and (2) 480VAC portable backup generators supplying power to the battery chargers for 125 VDC buses NKO1, NKO2, and NKO4. Failure of each of these two FLEX strategies are represented by single basic events (i.e., FLEXAFWFAIL and FLEXACTODCFAIL) that are set conservatively at a failure probability of 0.99. Though the strategies involve use of equipment and operator actions, no other FLEX equipment or operator failures are modelled. The licensee also explained that it performed a sensitivity analysis on this FLEX credit by reducing the failure probability of two basic events (FLEXAFWFAIL and FLEXACTODCFAIL) to 0.1. The results of the sensitivity study show that the SCDF only increased by 5 percent and the SLERF increased by 1 percent. Therefore, the NRC staff concludes that the modeling of FLEX strategies in the SPRA did not identify potential plant modifications.

Based on the analysis described above, the NRC staff concludes that no modifications, other than those actions identified by the licensee as a regulatory commitment, 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 Callaway 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. Except for the top two cutsets, the dominant LERF cutsets reported in Table 5-6 of the submittal include seismically induced failure of the reactor building penetrations which was assessed in the analysis described above. The top cutset contains seismic-induced soil failure and the second cutset contains seismically induced failure of the steam generator supports which were both assessed in the analysis 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 reviewed the results of the IPEEE and SAMA analyses previously completed for Callaway to identify additional substantial safety improvements that would be cost justified. No other potential improvements were found based on this review.

Lastly, the staff noticed inconsistencies in the point estimate SCDF and SLERF values reported in the July 10, 2020 supplement. During the audit, the staff communicated this to the licensee for information purposes. However, this analysis used the mean SCDF and SLERF values, which were reported consistently. The correct point estimates were confirmed by the licensee during the audit as discussed in Topic 15 of Enclosure 1.

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 in addition to the plant modifications identified by the licensee;

- No other 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 SCDF

| Fragility Group/Event | Descriptions | Failure Mode | RRW | MCR (/rx-yr) |
|---|---|---|---|---|
| *Seismically Induced SSC Failures* | | | | |
| SF-IE-T1 | Seismic-Induced Loss of Offsite Power | Yard Centered Loss of Offsite Power | 1.575 | 2.65E-05 |
| SF-IE-SW | Seismic-Induced failure of service water (NSCI) | Loss of Non-Nuclear Safety Equipment | 1.065 | 4.43E-06 |
| SF-FR-YDXFR | Seismic rupture of yard transformer housings, oil leakage, and subsequent ignition | Loss of Non-Nuclear Safety Equipment | 1.058 | 3.99E-06 |
| SF-NSCI | Seismic Induced failure of Non-SC-I SSCs | Loss of Non-Nuclear Safety Equipment | 1.058 | 3.99E-06 |
| SF-NB01 | Seismic Induced Failure of the 4.16 KV Switchgear NB01 | Loss of Switchgear NB01 | 1.055 | 3.78E-06 |
| SF-NK02 | Seismic Induced Failure of the 125 V DC Bus NK02 | Loss of 125V DC Bus NK02 | 1.037 | 2.61E-06 |
| Relay_0.33 | Relay Fragility Group | Relay Chatter | 1.031 | 2.18E-06 |
| SF-NG02 | Seismic Induced Failure of 480 V Load Center NG02 | Loss of 480 V Load Center NG02 | 1.028 | 1.96E-06 |
| SF-NGXC-1 | Seismic Induced Failure of the MCC NG05E and NG06E | Loss of MCCs NG05E and NG06E | 1.022 | 1.60E-06 |
| SF-NG01 | Seismic Induced Failure of 480 V Load Center NG01 | Seismic Induced Failure of 480 V Load Center NG01 | 1.022 | 1.60E-06 |
| SF-NNOX | Seismic Induced Failure of 120 VAC Distribution Panels NN01, NN02, NN03, and NN04 | Loss of 120 VAC Distribution Panels NN01/2/3/4 | 1.020 | 1.45E-06 |
| *Human Failures* | | | | |
| SH2-OP-XHE-FO-RCPTRP-CCW | Op fails to trip RCP from control room after failure of CCW (seismic) | Not provided | 1.068 | 4.65E-06 |
| SH2-NE-XHE-FO-EDG | Operator Fails to Start and Align a Diesel Generator | Not provided | 1.059 | 4.07E-06 |
| SH2-AL-XHE-FO-SBOSGL | Operator Fails to Control SG Level after Complex Seismic Event | Not provided | 1.057 | 3.90E-06 |
| SH2-OP-XHE-FO-RFLN2A | Not provided | Not provided | 1.049 | 3.41E-06 |
| SH2-EG-XHE-FO-STBTRN | Op Fails to Transfer from CCW Train A to Train B before Rx Trip | Not provided | 1.033 | 2.32E-06 |

| | | | | |
|---|---|---|---|---|
| SH2-OP-XHE-FO-AEPS_REALIGN | Operator Fails to Realign AEPs | Not provided | 1.028 | 1.96E-06 |
| SH2-NG-XHE-FO-PA1410 | Not provided | Not provided | 1.026 | 1.82E-06 |
| SH2-NE-XHE-FO-EDG-RLYSET | Operator Fails to Start EDG Following seismic-induced relay chatter | Not provided | 1.016 | 1.16E-06 |
| SH2-OP-XHE-FO-AEPS1 | Operator Fails to Align AEPs | Not provided | 1.014 | 1.02E-06 |
| SH2-OP-XHE-FO-ACRECV | Operator Fails to Recover from a Loss of Offsite Power (Seismic) | Not provided | 1.011 | 7.99E-07 |
| SH2-EF-XHE-FO-MANESW | Operator Fails to Manually Start and Align ESW System | Not provided | 1.011 | 7.99E-07 |

Table 2: Importance Analysis Results of Top Contributors to SLERF

| **Fragility Group/Event** | **Descriptions** | **Failure Mode** | **RRW** | **MLR (/rx-yr)** |
|---|---|---|---|---|
| *Seismically Induced SSC Failures* | | | | |
| SF-SOIL | Seismic-Induced Soil Failure | Soil Bearing Capacity | 1.242 | 1.61E-06 |
| SF-NSSG | Seismic-Induced Failure of the Steam Generator Supports | Structural (SG Column) | 1.241 | 1.60E-06 |
| SF-RB-PEN | Seismic-Induced Failure of the Reactor Building Penetrations | Shear Failure | 1.171 | 1.21E-06 |
| SF-IE-T1 | Seismic-Induced Loss of Offsite Power | Yard Centered Loss of Offsite Power | 1.103 | 7.69E-07 |
| *Human Failures* | | | | |
| SH2-OP-XHE-FO-RCPTRP-CCW | Operator Fails to Manually Start and Align ESW System | Not provided | 1.003 | 2.18E-07 |

AUDIT SUMMARY BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO

CALLAWAY PLANT, UNIT 1

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-0006)

## BACKGROUND AND AUDIT BASIS

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

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

By letter dated July 6, 2017 (ADAMS Accession No. ML17177A446), the NRC issued a generic audit plan and entered into the audit process described in Office Instruction LIC-111, "Regulatory Audits," dated December 29, 2008 (ADAMS Accession No. ML082900195), to assist in the timely and efficient closure of activities associated with the letter issued pursuant to 10 CFR Part 50, Section 50.54(f). By letter dated July 11, 2017 (ADAMS Accession No. ML17192A168), the NRC staff confirmed that the audit process for the seismic hazard reevaluations applies to Callaway Plant, Unit 1 (Callaway).

## REGULATORY AUDIT SCOPE AND METHODOLOGY

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

## AUDIT ACTIVITIES

The NRC staff developed questions to verify information in the licensee's submittal and to gain understanding of non-docketed information that supports the docketed SPRA submittal. The staff's clarification questions dated September 20, 2019, November 13, 2019, November 29, 2019, and January 2, 2020 (ADAMS Accession Nos. ML19304C325, ML19317E633, ML19333B869, and ML20101F977, respectively), were sent to the licensee to support the audit.

The licensee provided clarifying information in the following areas:

- Status of the Internal Events PRA (IEPRA) resolution of finding-level fact and observations (F&Os) and their effect over the SPRA.
- Discussion of fragility methods and approaches used for certain structures, systems and components (SSCs).
- Insights about soil-structure interaction, in-structure response spectra, and foundation input response spectra calculations.
- Consideration of human errors in different seismic hazard intervals.
- Discussion of the SCDF and SLERF importance measures.
- Discussion of the use of Diverse and Flexible Coping Strategies (FLEX) equipment in the SPRA model and associated human actions.

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

DOCUMENTS AUDITED

- PWROG-19011-P, Revision 0, "Independent Assessment of Facts & Observations Closure and Focused Scope Peer Review of the Callaway Seismic Probabilistic Risk Assessment," May 2019
- PWROG-18044-P, Revision 0, "Peer Review of the Callaway Seismic Probabilistic Risk Assessment," September 2018
- Ameren Report No. AMN#PES00031-REPT-001, Revision 0, "Callaway Energy Center Probabilistic Risk Assessment Focused Scope Peer Review," July 2020
- Ameren Report No. AMN#PES00031-REPT-002, Revision 0, "Callaway Energy Center Probabilistic Risk Assessment Peer Review F&Os Closure," July 2020
- CEC Document PRA-SPRA-001, Draft Revision 0-F, "Seismic Probabilistic Risk Assessment Modeling Notebook," March 2019
- CEC Document PRA-SPRA-002, Draft Revision 0-E, "Seismic Probabilistic Risk Assessment, Quantification Analysis Notebook," March 2019
- Project Document 11-4695B, Revision 2, "Probabilistic Seismic Hazard Analysis Seismic Probabilistic Risk Assessment Project Callaway Energy Center, Unit 1," Rizzo

International, February 2019

- WEC Document 15C4310-CAL-010, Revision 3, “HCLPF Analysis of Select Equipment Anchorage,” Callaway Plant, May 14, 2019

- WEC Document 15C4310-CAL-003, Revision 0, “Detailed Fragility Analysis for Select Equipment Functionality,” Callaway Plant, December 14, 2018

- WEC Document 15C4310-RPT-003, Revision 6, “Seismic Fragility Analysis Results of CEC Structures, Systems and Components,” Callaway Plant, September 2019

- WEC Document 15C4311-CAL-002, Revision 0, “Auxiliary and Control Building Soil-Structure Interaction Analysis,” Callaway Plant, May 11, 2016

- WEC Document LTR-RAM-15-52, Revision 0, “Callaway Energy Center Seismic PRA HRA Walkdown Summary,” January 6, 2016

## 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 Callaway.

SUBJECT: CALLAWAY PLANT, UNIT 1 – 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-0006) DATED SEPTEMBER 12, 2020

**DISTRIBUTION**:
PUBLIC
LPMB R/F
RidsNrrLaSLent Resource
RidsNrrDorlLPMB Resource
RidsNrrDorlLPL4 Resource
RidsNrrPMCallaway Resource
MValentin, NRR
RidsACRS_MailCTR Resource
DWrona, NRR
SRosenberg, NRR

**ADAMS Accession No.: ML20210L323** ***concurrence via email**

| | | | |
|---|---|---|---|
| OFFICE | NRR/DORL/LPMB/PM* | NRR/DNRL/NLRP/LA* | NRR/DORL/LPMB /BC* |
| NAME | MValentin | SLent | DWrona (Morgan-Butler for) |
| DATE | 7/29/2020 | 7/29/2020 | 8/14/2020 |
| OFFICE | NRR/DRA/D* | NRR/DEX/DD* | NRR/DORL/DD* |
| NAME | MFranovich | JMarshall | GSuber |
| DATE | 9/12/2020 | 9/8/2020 | 9/12/2020 |

**OFFICIAL RECORD COPY**