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

July 23, 2021

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 NO. 1 – AUDIT PLAN AND SETUP OF ONLINE REFERENCE PORTAL FOR LICENSE AMENDMENT REQUEST REGARDING RISK-INFORMED APPROACH FOR CLOSURE OF GENERIC SAFETY ISSUE-191 (EPID L-2021-LLA-0059)

Dear Mr. Diya:

By application dated March 31, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21090A185), as supplemented by letter dated May 27, 2021 (ADAMS Accession No. ML21147A222), Union Electric Company, dba Ameren Missouri (the licensee), submitted a license amendment request (LAR), exemption request, and updated response to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors," dated September 13, 2004 (ADAMS Accession No. ML042360586), for Callaway Plant, Unit No. 1 (Callaway). The amendment would modify the Callaway licensing bases, including the affected portions of the Technical Specifications and Updated Final Safety Analysis Report. Specifically, the amendment would allow the use of a risk-informed approach to address safety issues discussed in U.S. Nuclear Regulatory Commission (NRC) Generic Safety Issue-191, "Assessment of Debris Accumulation on PWR [Pressurized-Water Reactor] Sump Performance."

The NRC staff will perform a regulatory audit to support its review of the LAR in accordance with the enclosed audit plan. The audit will be conducted remotely on August 10–17, 2021. A regulatory audit is a planned activity that includes the examination and evaluation of primarily non-docketed information. The audit will be conducted to increase the NRC staff's understanding of the LAR, verify information, and identify information that will require docketing to support the basis for the licensing decision regarding the LAR.

The NRC staff plans to initially conduct a desk audit to review the documentation provided on the portal. The online reference portal would allow the NRC staff to audit basis documents to determine whether the information included in the documents is necessary to reach a safety conclusion on the application. Documents identified as necessary for analysis of the application will be identified by the NRC staff. The licensee will be formally requested to submit those documents on the NRC docket. Use of the online reference portal is acceptable, as long as the following conditions are met:

- The online reference portal will be password-protected, and passwords will be assigned to those directly involved in the review on a need-to-know basis.

- The online reference portal will be sufficiently secure to prevent staff from printing, saving, or downloading any documents; and

- Conditions of use of the online reference portal will be displayed on the login screen and will require concurrence by each user.

The NRC staff would like to request that the portal be populated with the documents listed in the enclosure to this letter. This is the initial list identified by the NRC staff. The NRC staff may request additional documents during the review, which will be transmitted to you via email. This will help with the preparation for a virtual audit. Please provide NRC staff access to the portal and send me the information needed to access the portal, such as username and password, as soon as possible. The conditions associated with the online reference portal must be maintained throughout the duration of the review process. Please provide written confirmation that Ameren Missouri agrees to the terms and conditions set forth in this letter.

If you have any questions, please contact me at 301-415-8371 or by e-mail at Mahesh.Chawla@nrc.gov.

Sincerely,

/RA/

Mahesh L. Chawla, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-483

Enclosure:
Audit Plan

cc: Listserv

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

REMOTE AUDIT PLAN

REGARDING RISK-INFORMED APPROACH

FOR CLOSURE OF GENERIC SAFETY ISSUE-191

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT NO. 1

DOCKET NO. 50-483

1.0 BACKGROUND

By application dated March 31, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21090A184), as supplemented by letter dated May 27, 2021 (ADAMS Accession No. ML21147A222), Union Electric Company, dba Ameren Missouri (the licensee), submitted a license amendment request (LAR), exemption request, and updated response to Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors," dated September 13, 2004 (ADAMS Accession No. ML042360586), for Callaway Plant, Unit No. 1 (Callaway). The amendment would modify the Callaway licensing bases, including the affected portions of the Technical Specifications and Updated Final Safety Analysis Report. Specifically, the amendment would allow the use of a risk-informed approach to address safety issues discussed in U.S. Nuclear Regulatory Commission (NRC) Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on PWR [Pressurized-Water Reactor] Sump Performance."

2.0 REGULATORY AUDIT SCOPE

The scope of this audit includes the calculations, analysis, and supporting documentation described or referenced in the licensee's application, as supplemented. The intent of the audit is to gain an understanding of the licensee's approach described in the application, as supplemented, including use of NRC staff guidance applicable to the resolution of GL 2004-02.

3.0 INFORMATION NEEDED

The appendix to this audit plan lists the specific topics and questions that the NRC staff plans to discuss with the licensee during the audit. The following is an initial list of documents or information that the NRC staff would like to have access to prior to, and during, the audit:

- Spreadsheet with the debris types (generated and transported to the strainer) for every weld, break size, and break orientation. Include the weld identifier and weld coordinates.

- Documentation of debris transport computations, in support of transport fractions in Table 3.e-1 of Enclosure 3 of the LAR, including a description of assumed pump configurations and flow rates.

- The document(s) that describe the different pump configurations analyzed in the risk-informed approach, specifically related to the in-vessel fiber buildup computations (e.g., Figures 3.n-2 to 3.n-5 of LAR Enclosure 3, Attachment 3-2). Include a description of strainer filtration and shedding as functions of debris loads, as well as flow rates considered in the analyses.

- ALION-REP-CEC-9143-014 Revision 0, "GSI-191 Risk Aggregation Methodology Report," Alion Science and Technology, 2018. This is Reference 6 on page 59 of 59 of LAR Enclosure 3, Attachment 3-3.

The audit team will not remove non-docketed information from the audit site (in this case the electronic portal). NRC contractors will maintain control of proprietary materials in accordance with NRC procedures and non-disclosure agreements.

4.0 AUDIT TEAM

The audit team will consist of:

- Steve Smith, Technical Reviewer
- Paul Klein, Technical Reviewer
- Matt Yoder, Technical Reviewer
- Bryce Lehman, Technical Reviewer
- Nate Jordan, Branch Chief
- Andrea Russell, Technical Reviewer
- Bob Vettori, Technical Reviewer
- John Tsao, Technical Reviewer
- Ben Parks, Technical Reviewer
- Shilp Vasavada, Acting Branch Chief
- Samantha Platt, Observer
- Osvaldo Pensado, Contractor
- Stuart Stothoff, Contractor
- Mahesh Chawla, Project Manager

5.0 LOGISTICS AND AGENDA

The audit will be conducted remotely from Tuesday, August 10, 2021, to Tuesday, August 17, 2021. Entrance and exit briefings will be held at the beginning and end of this audit, respectively.

The above initial list of documents or information should be made available to the NRC staff and its contractors via an online portal (or electronic reading room) at least 3 or 4 weeks prior to the start of the audit. During the audit, the NRC staff may request additional documents be made available via the online portal. NRC staff and contractor's access to the online portal should be terminated 2 weeks after the end of the audit.

Suggested Agenda[1]

Tuesday, August 10

9:00 a.m.	Introductions and opening remarks
9:30 a.m.	Overview of audit questions and topics
10:00 a.m.	Review of information and discussions
12:00 p.m.	Lunch
1:00 p.m.	Continue review of information and discussions
4:00 p.m.	Wrap-up meeting

Wednesday, August 11

9:00 a.m.	Continue discussions
12:00 p.m.	Lunch
1:00 p.m.	Continue discussions
4:00 p.m.	Wrap-up meeting

Thursday, August 12

9:00 a.m.	Continue discussions
12:00 p.m.	Lunch
1:00 p.m.	Continue discussion
4:00 p.m.	Wrap-up Meeting

Tuesday August 17

9:00 a.m.	Review open questions
10:00 a.m.	Discuss open questions
12:00 p.m.	Lunch
1:00 p.m.	Continue discussion of open questions
2:00 p.m.	Break and NRC break-out meeting
3:15 p.m.	Exit meeting
4:00 p.m.	Adjourn

The NRC project manager will coordinate any changes to the audit schedule, location, or agenda with the licensee. Breaks will be scheduled as determined by the attendees. If the discussions are completed in less time than allotted in the schedule adjustments may be made.

6.0 SPECIAL REQUESTS

The NRC staff would like access to the following equipment and services during the audit:

- A computer running the version of the CASA Grande software used to support the approach described in the application, as supplemented. The interest is in visualizing one or two zones of influence of critical breaks, especially for breaks that become critical only at the double-ended guillotine break limit.

[1] All times shown are in Eastern Daylight Time (EDT).

The licensee should establish an online portal that allows the NRC staff and contractors to access documents remotely. The following conditions associated with the online portal must be maintained throughout the duration that the NRC staff and contractors have access to the online portal:

- The online portal will be password-protected, and separate passwords will be assigned to the NRC staff and contractors who are participating in the audit on a need to know basis.

- The online portal will be sufficiently secure to prevent the NRC staff and contractors from printing, saving, or downloading any documents.

- Conditions of use of the online portal will be displayed on the login screen and will require acknowledgement by each user.

Username and password information should be provided directly to the NRC staff and contractors. The NRC project manager will provide the licensee with the names and contact information of the NRC staff and contractors who will be participating in the audit. All other communications should be coordinated with the NRC project manager.

7.0 DELIVERABLES

Within 60 days of the exit of the audit, the NRC staff will prepare an audit summary documenting the information reviewed during the audit, and any significant observations. If the NRC staff identifies information during the audit that is needed to support the NRC staff's regulatory decision on the submittal, the staff will issue a request for additional information to the licensee after the audit.

Appendix – Topics and Questions for Discussion

Note that page numbers in this appendix are PDF page numbers from documents in the Agencywide Documents Access and Management System (ADAMS), not numbers from the enclosure headers.

General Information and Licensing Basis

(1) Technical Specification (TS) and Technical Bases issues identified:

a. In the TS bases markup under Surveillance Requirement 3.6.8.1, the word "program" is missing at the end of the last sentence. It should be Surveillance Frequency Control **Program** (see Enclosure 2 of the license amendment request (LAR) dated March 31, 2021 (ADAMS Accession No. ML21090A184, page 58 of 109).
b. The TS markups and final typed pages do not include an updated index entry for TS 3.6.8.
c. The TS final typed pages do not include footers on the final pages starting on page 60 of 109 of LAR Enclosure 2.

(2) Discuss the need for exemptions to General Design Criteria 35, 38, and 41 of Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.

(3) In the Final Safety Analysis Report (FSAR) markup in Section 6.3A.1.4, the last sentence should include that the completion time is also based on the low probability of an initiating event. This language for justification is from Technical Specifications Task Force (TSTF) Traveler TSTF-567. Also, in the same paragraph for Condition A, the number of sumps is irrelevant (see LAR Enclosure 2, page 100 of 109).

(4) In the FSAR markup of key methods—Programs or method other than CASA Grande for calculation of debris generation, transport, or sub-model calculations may be used as long as they are performed per the approved guidance or other U.S. Nuclear Regulatory Commission (NRC) approval in its safety evaluation. However, integrated calculations for changes in risk may need to be performed using this program. The NRC staff has invested significant resources to validate the CASA Grande methodology. Other methods have not been reviewed by the NRC staff. The FSAR markup is not clear on this point. The FSAR should identify CASA Grande as the method used to perform the overall risk evaluation. Alternately, other methods should be identified along with any necessary limitations (see LAR Enclosure 2, page 102 of 109). This is related to a question in the LAR, Exemption Request, and Performance Monitoring Section.

(5) It was not apparent that the key methods in the FSAR markup include the following important methods (refer to LAR Enclosure. 2, page 102 of 109):

a. transport methodology
b. methods to estimate generation of debris types other than fiber (e.g., chemical precipitates, coatings, or other potential debris sources).
c. limits on other debris types
d. methods for performing in-vessel downstream effects evaluations
e. methods for performing ex-vessel downstream effects evaluations

(6) In Table 6.3A-2 on page 108 of 109 of LAR Enclosure 2, should the note for the low-density fiber glass (LDFG) fines that states that the 300 pound mass (lbm) includes 30 lbm of latent fiber, also state that it includes 50 lbm of fine fiber margin? In the same table, the particulate debris amounts are provided as volumes, but the total is provided as a mass. Can the consistency within the table be increased by using a volume for the total and for the available margin, especially considering that Note 3 states that it is the volume of debris on the strainer that is related to headloss?

Debris Generation/Zone of Influence (Excluding Coatings)

(7) Discuss the methodology for the hemispherical break zones of influence (ZOIs) mapping. Are the ZOIs centered at the edge or center of the pipe where the break is assumed to occur? Refer to page 96 of 109 in LAR Enclosure 2 for the FSAR description.

(8) The NRC staff notes that besides the potential for pipe weld failures, there are other potential failures from non-pipe components in the reactor coolant system (RCS) pressure boundary such as steam generator manways, pressurizer nozzles (e.g., heater sleeves, safety and relief valves), pump bodies, and mechanical joints. Clarify whether the non-pipe components mentioned above were selected as a break location and evaluated for debris generation. If not, provide justification.

(9) Discuss whether the break selection process considered non-weld locations where there is an elevated potential for failure, for example highly stressed locations, branch connections, and elbows. Refer to page 9 of LAR Enclosure 3.

(10) In LAR Enclosure 3, on pages 12 and 17, the licensee states that FOAMGLAS® is treated as fiber and that this results in an overprediction of risk due to the additional LDFG being transported to the strainer. Considering the relatively low particulate debris margins for some cases, how was this evaluated? The NRC staff would be able to perform confirmatory calculations for this issue if a database of debris generation and transport for each break scenario, including FOAMGLAS®, is made available to the staff. This is related to a question in the Risk-Informed Bases section and the request for the spreadsheet of debris generation and transport.

(11) In LAR Enclosure 3, on pages 15 and 21, the submittal states that the assumption for miscellaneous debris is 200 square feet. It is not clear whether the containment was inspected for signs, tags, labels, or other similar items that could transport to the strainers to validate this value. The submittal discusses that these items are generally not used but does not state that any inspection was done to ensure that the actual amount of miscellaneous debris is bounded by the assumption. Discuss any actions taken to verify the assumption.

(12) Page 18 of LAR Enclosure 3 discusses reflective metal insulation (RMI) debris. The NRC staff agrees that RMI in a typical debris bed will generally result in lower headloss. Discuss the potential that RMI could fill the voids in the sump between the strainer stacks and be covered with a debris bed that has an area similar to the sump opening in the floor.

<u>Transport</u>

(13) Discuss the calculation of the erosion fractions for small and large fiber pieces. The NRC staff understands that the effect is small, and margin is included in the analysis, but would like to understand the methodology. Refer to page 139 of LAR Enclosure 3.

(14) The response to 3.h.2 on page 54 of LAR Enclosure 3 states that the transport phase applied to qualified coatings is the recirculation phase. Explain this assumption.

<u>Head Loss and Vortexing (Attachment 3-2, Item 3f)</u>

(15) Describe how the lower submergence for the small break loss-of-coolant accident (LOCA) (and potentially medium break LOCA) is considered in the flashing and degasification evaluations? Does the submergence increase significantly after sump switchover? Is reduced debris headloss considered for the smaller break cases? Refer to page 29 of LAR Enclosure 3.

(16) The title for Figure 3.f-4 is incorrect. It states that it is a containment spray system (CSS) process flow diagram, but actually depicts the coatings surrogate size distribution.

(17) Explain and justify the method used to determine and justify the amount of pressure credited to suppress flashing across the strainer. (LAR Enclosure 3, Attachment 3-2, page 42) The NRC staff had difficulty understanding the method described in the response to 3.f.14. Provide the containment pressure(s) and sump temperature(s) used for this analysis. What are the potential ranges for these parameters that could occur for a similar scenario depending on the assumptions used? For example, what are the assumptions for containment air cooler and CSS operation? How does service water temperature affect the response? Is any containment pressure credit needed to suppress flashing at temperatures lower than 212 degrees Fahrenheit (°F) (e.g. 211 °F)? Why was 212 °F chosen as the lowest temperature? NRC staff guidance is to use assumptions that minimize containment pressure and maximize sump temperature. However, a demonstration of large margin using a design basis calculation is also acceptable. The NRC staff agrees that sump pool temperature and containment pressure are related, but one parameter can lag the other.

<u>Net Positive Suction Head (NPSH)</u>

(18) Are the NPSH calculations for a small break LOCA modified to account for a lower pool level that could result from lack of injection from the accumulators and reduced inventory from the RCS? The NPSH margin results are shown on page 52 of Enclosure 3 of the LAR. The sump pool mass inputs are on page 51, and the discussion of Section 3.g.2 on page 44 states that the static head is constant and is based on a minimum large-break LOCA. This issue is related to the issue regarding submergence assumptions for flashing and degasification in the head loss and vortexing section.

(19) On page 46 of LAR Enclosure 3, the submittal states that the CSS is not expected to start for a small break LOCA. Explain whether this has any effect on the headloss or NPSH calculations for small break LOCA scenarios.

Coatings

(20) Discuss the credit for previously unqualified coating system Carboline 193LF primer with 191HB topcoat as remaining adhered on page 77 of LAR Enclosure 3.

(21) Provide examples of how the scaling of surrogate debris volumes (silica sand and ground silica) to represent plant conditions was performed. Refer to the response to 3.h.3 on page 55 of LAR Enclosure 3.

In-Vessel

(22) Provide clarification on the scenarios evaluated for the in-vessel evaluation, and the assumptions used in the evaluations. Refer to the description starting on page 102 of Enclosure 3 of the LAR. The information provided in the supplemental response dated May 27, 2021 (ADAMS Accession No. ML21147A222), did not provide the assumptions used to develop the fiber penetration and transport model used to calculate accumulation of fiber at the core inlet. Provide a list of the scenarios considered for the in-vessel fiber transport evaluations. For each scenario, provide the equipment that is considered to be operating and the resulting assumptions for the scenario. It appears that at least two scenarios were modeled. One scenario is that two trains of the emergency core cooling system (ECCS) and CSS are operating while another is that two ECCS trains are operating along with one CSS train. The submittal states that it is assumed that 300 lbm of fiber is transported to two operating ECCS strainers. Provide the assumptions for transport to, and penetration through each strainer. Is debris transport proportional to flow? What are the flow rate assumptions for each strainer for each scenario? What are the flow rates through the in-vessel and bypass paths? How is sump switchover timing affected by the number of pumps running and how does this affect the results? Provide the assumptions for depletion rates of fiber in the pool. Describe the strainer filtration and shedding functions as function of debris loads.

(23) Provide fiber penetration test results and penetration model assumptions that are based on the testing to the extent that the NRC staff can perform confirmatory calculations to validate the in-vessel fiber values used in the analysis.

(24) The in-vessel evaluation did not provide any results or evaluation of cases where CSS does not run. Provide the basis for the assumption that CSS will start and continue to run for the duration of the strainer penetration analysis period. The NRC staff understands that the submittal states that once CSS is started it will not be secured until the containment pressure reaches 4.5 pounds per square inch gauge per emergency operating procedures.

(25) On page 104 of LAR Enclosure 3, the licensee states that “full dual-train spray and dual-train ECCS was assumed and tested for the purpose of recirculation strainer performance characterization.” Explain what is meant by this statement. State whether it relates only to the in-vessel analysis or if it applies to other aspects of the submittal.

(26) On page 102 of LAR Enclosure 3, the submittal provides an evaluation of the decay heat for the WCAP-17788 reference plant as compared to the decay heat estimated using the American National Standards Institute/American Nuclear Society (ANSI/ANS)-5.1-1979 decay heat standard. The 1979 standard notes that a maximum positive uncertainty between 0- and 1,000-seconds post-shutdown is 20 percent, while the uncertainty from

1,000 seconds to a period significantly onward is 10 percent. Clarify whether the estimated value at 11.7 minutes (83.2 megawatt thermal (MWt)) includes any allowance for uncertainty, and if so, explain how much.

Chemical Effects

(27) On page 102 of LAR Enclosure 3, the licensee states that testing demonstrated that chemicals will not form prior to 6 hours. In the supplement dated May 27, 2021, it is stated that chemical effects will not occur until at least. 24 hours. Clarify which of these statements reflects the earliest time at which chemicals may form with respect to the in-vessel evaluation.

(28) Figure 3.o-1, “Chemical Effects Evaluation Process Flow Chart,” indicates that near field settlement was credited in the chemical effects evaluation. Based on other discussions in the submittal the NRC staff believes that the flow chart is in error. Confirm this or provide additional information that clarifies the treatment of near-field settlement for chemical effects (see page 112 of LAR Enclosure 3).

Risk-Informed Basis

(29) Confirm that the technical acceptability of the licensee’s probabilistic risk assessments (PRAs), including, dispositions of the open peer review finding level facts and observations and key assumptions and sources of uncertainty, provided in the licensee’s 10 CFR 50.69 and National Fire Protection Association (NFPA)-805 LARs is applicable in its entirety to this request.

(30) LAR Enclosure 2, page 23, a note states, in part, “The PRA model risk metrics provided herein have not been adjusted to account for an open modeling issue recently identified in the Callaway PRA.” Discuss this open modeling issue providing details and demonstrating the impact, or lack of impact, on this application.

(31) It is unclear how first isolation valves are defined and what the required pre-scenario positions or post-initiating event requirements for these valves would be. The LAR states that valves beyond the first isolation valve are considered secondary risk contributors. To take credit for a valve isolating a break it would have to be normally closed and isolated from RCS pressure or close rapidly upon a LOCA. In that case, significant debris generation could occur prior to the valve closing. If debris generation occurs, but the leak can be isolated, recirculation may not be required. Provide a more detailed definition of “first isolation valves” and how these scenarios were evaluated. Provide additional information on the assumption that the isolation valve failure rate is conservatively estimated to be 1.11E-03. Refer to Enclosure 3, Attachment 3-3, page 146.

(32) Explain the rationale behind performing the sensitivity study to add insulation at valves? Is there uncertainty about the amount of insulation on valves? How many valves are modeled? For which breaks are the valves within the ZOI? Explain why a sensitivity study was not developed for fiber amount that could be evaluated in a more straightforward manner. For example, simply adding and subtracting some percentage of fiber with respect to the baseline could provide a better generic understanding of sensitivity to fiber amount. Refer to page 152 of LAR Enclosure 3.

(33) LAR Enclosure 3, page 12, the licensee states, in part,

> FOAMGLAS® is located on the steam generator blowdown system and Residual Heat Removal (RHR) system. FOAMGLAS® was discovered in containment in the summer of 2019 and is not evaluated for debris generation. Approximately 146 ft^3 or 1167 lbm of FOAMGLAS® are in containment. In the analysis documented in this LAR, low density fiber glass (LDFG) is modeled at the location of FOAMGLAS®. This results in an over prediction of destroyed LDFG and risk, but an under prediction of destroyed particulate at break locations that have the potential to destroy FOAMGLAS®

Explain if the "under prediction of destroyed particulates at break locations that have the potential to destroy FOAMGLAS®" has an effect on change in risk estimates presented in the application.

(34) Discuss how the following potential initiating events were examined and accounted for, or excluded with respect to the risk-informed analysis: (1) Internal fire LOCAs, (2) Internal flood LOCAs, (3) Non-piping LOCAs (e.g., manway covers, valves, control element drive assemblies, and instrument lines), (4) water hammer-induced LOCAs.

(35) Describe how the secondary line breaks (main steam and feedwater) that result in recirculation were evaluated in the PRA. The NRC staff understands that a conditional sump failure probability was computed using CASA Grande. This conditional probability was input into the PRA, and then delta core damage frequency (ΔCDF) contributions were calculated. Discuss whether CASA Grande was executed with breaks only on the main steam line and main feedwater lines (to compute the sump failure probability conditional on the line break and conditional on a break size). Provide the magnitudes of the conditional sump failure probabilities. Provide the source and basis for the initiating event (secondary line break) frequency used for the estimate of change in risk from such breaks. Discuss, which justification, whether different frequencies were used for different secondary line break sizes.

(36) Section 9.1 of LAR Enclosure 3 provides sensitivity studies for "alternative initiating frequency aggregation methods…" The sensitivity using arithmetic mean shown in Figure 9-2 indicates that the mean value for change in CDF exceeds Region III acceptance guidelines in Regulatory Guide (RG) 1.174, "An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," by non-trivial amounts. Section 9 of LAR Enclosure 3 states that "[a]rithmetic aggregation weights all elicited values equally, including outliers, and is explored for informational purposes." However, the NRC staff considers the use of the geometric mean as a key assumption, and consistent with guidance on risk-informed decision making, uses the results of sensitivities with alternative models to inform its decision. Therefore, the results of the sensitivity on initiating frequency aggregation are used by the NRC staff for more than "informational purposes." Justify how the proposed license amendment and exemption request are consistent with the risk-informed resolution of Generic Letter 2004-02 if sensitivities demonstrate that acceptance guidelines for Region III in RG 1.174 are exceeded. The justification can include demonstration of the impact of conservatisms in the approach to calculate the change in risk.

(37) On page 140 of LAR Enclosure 3, the submittal states "[t]he Bounded Johnson distributions are used to create the blue dots in Figure 7-1 and are supplemental information." On page 141 and 151 of LAR Enclosure 3, it appears that it is used to calculate the ΔCDF for use in this application. Clarify whether or not the Bounded Johnson distribution is used to calculate the ΔCDF.

(38) Based on the information provided in Section 9 of LAR Enclosure 3, the uncertainty in the LOCA initiating frequencies appears to be the only parametric uncertainty investigated by the licensee. However, uncertainty exists in the other input parameters used to calculate the conditional strainer failure probability. Therefore, please provide the following:

a. A sensitivity analysis to identify which inputs have the greatest impact on the risk quantification results. Include the process of identifying input variables to evaluate, selecting the minimum and maximum value for each variable, and quantify the risk in terms of delta CDF.

b. A parametric sensitivity analysis that biases the inputs to the most conservative values, if not already biased in the base case.

Defense In Depth and Safety Margin

(39) Enclosure 3, Attachment 3-4, Section 2.5.2.2, "Reactor Coolant System Weld Mitigation," states that "All large bore reactor vessel welds susceptible to [primary] water stress corrosion cracking (PWSCC) have been mitigated by water jet peening in 2017." The NRC staff noted that nickel-based Alloy 600/82/182 components and welds are susceptible to PWSCC. Besides large bore pipe welds, some pressurized water reactor plants have Alloy 600/82/182 material in various pressurizer nozzle welds, reactor vessel closure head penetration nozzles and associated attachment welds, welds for the nozzles attached to the reactor vessel, pressurizer, and steam generator. Besides the welds in the large bore RCS piping that are susceptible to PWSCC, identify Alloy 600/82/182 dissimilar metal butt welds and components in the RCS pressure boundary that have not been mitigated to minimize PWSCC. Discuss whether the non-mitigated Alloy 600/82/182 weld and component locations were analyzed for the debris generation with a higher probability than for the mitigated welds and components. If not, provide justification.

(40) The NRC staff notes that to monitor structural integrity, pressurized water reactor plant owners periodically inspect RCS piping and associated components beyond the NRC regulations such as operator walkdowns, opportunistic inspections, the boric acid corrosion program, and the fatigue monitoring program per Materials Reliability Program (MRP)-146, Revision 1, "Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines." Discuss any periodic inspections at Callaway that monitor the structural integrity of the RCS piping and components beyond the NRC regulations that could minimize the potential for pipe and component failures.

License Amendment Request, Exemption Request, and Performance Monitoring Program

(41) The LAR and exemption request do not provide information relevant to Key Principle 5 in RG 1.174, Revision 3 (i.e., the impact of the proposed licensing basis change should be monitored using performance measurement strategies). Demonstrate that the LAR and exemption request will meet Key Principle 5 of RG 1.174, Revision 3, using existing and new performance monitoring strategies. The demonstration should identify the strategies and how they support meeting Key Principle 5.

(42) Identify key elements of the risk-informed analysis and corresponding methods, approaches, and data that, if changed, would constitute a departure from the method used in the safety analysis as defined by 10 CFR 50.59.

(43) Identify the relevant elements of the risk-informed assessment that may need to be periodically updated. The licensee must describe the program or controls that will be used to ensure relevant elements of the risk-informed assessment are periodically updated.

(44) Describe a reporting and corrective action strategy for addressing situations in which an update to the risk-informed assessment reveals that the acceptance guidelines described in Section 2.4 of RG 1.174, Revision 3, have been exceeded.

SUBJECT: CALLAWAY PLANT, UNIT NO. 1 – AUDIT PLAN AND SETUP OF ONLINE REFERENCE PORTAL FOR LICENSE AMENDMENT REQUEST REGARDING RISK-INFORMED APPROACH FOR CLOSURE OF GENERIC SAFETY ISSUE-191 (EPID L-2021-LLA-0059) DATED JULY 23, 2021

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JTsao, NRR
JMa, NRR
RHoffman, NMSS
BParks, NRR
SPlatt, NRR
DBradley RIV
SJanicki, RIV
NOKeefe, RIV

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OFFICE	NRR/DORL/LPL4/PM	NRR/DORL/LPL4/LA	NRR/DNRL/NCSG/BC	NRR/DSS/STSB/BC(A)
NAME	MChawla	PBlechman	SBloom	NJordan
DATE	07/21/2021	07/21/2021	07/22/2021	07/22/2021
OFFICE	NRR/DNRL/NVIB/BC	NRR/DEX/ESEB/BC	NRR/DSS/SFNB/BC	NRR/DRA/APLB/BC(A)
NAME	ABuford	JColaccino	RLukes	SVasavada
DATE	07/22/2021	07/23/2021	07/22/2021	07/22/2021
OFFICE	NRR/DORL/LPL4/BC	NRR/DORL/LPL4/PM		
NAME	JDixon-Herrity (SLee for)	MChawla		
DATE	07/23/2021	07/23/2021		

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