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

September 14, 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 SUMMARY FOR LICENSE AMENDMENT REQUEST AND REGULATORY EXEMPTIONS FOR A RISK-INFORMED APPROACH TO ADDRESS GENERIC SAFETY ISSUE-191 AND RESPOND TO GENERIC LETTER 2004-02 (EPID L-2021 LLA 0059 AND EPID L-2021-LLE-0021)

Dear Mr. Diya:

By letter dated March 31, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21090A184), as supplemented by letters dated May 27, 2021 and July 22, 2021 (ADAMS Accession Nos. ML21147A222 and ML21203A192, respectively), Union Electric Company, dba Ameren Missouri (the licensee), requested an amendment for the licensee's final resolution to address the concerns of Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on PWR [Pressurized-Water Reactor] Sump Performance," and for responding to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.90, "Application for amendment of license, construction permit, or early site permit."

Additionally, in accordance with the regulations in 10 CFR 50.12, "Specific exemptions," the licensee requested exemptions from certain requirements of 10 CFR 50.46(a)(1), and Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, specifically General Design Criterion (GDC) 35, "Emergency core cooling"; GDC 38, "Containment heat removal"; and GDC 41," Containment atmosphere cleanup."

To support its review, the U.S. Nuclear Regulatory Commission (NRC) staff conducted a virtual regulatory audit on August 10–12 and August 17, 2021. The NRC staff reviewed documents and held discussions with members of Ameren Missouri and its contractors. The regulatory audit summary is enclosed with this letter.

If you have any questions, please contact me by telephone 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 Summary

cc: Listserv

OFFICE OF NUCLEAR REACTOR REGULATION

REGULATORY AUDIT SUMMARY FOR AUGUST 10-12 AND AUGUST 17, 2021

IN SUPPORT OF LICENSE AMENDMENT REQUEST FOR A

RISK-INFORMED APPROACH TO ADDRESS

GENERIC SAFETY ISSUE-191 AND RESPOND TO GENERIC LETTER 2004-02

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 letters dated May 27, 2021 and July 22, 2021 (ADAMS Accession Nos. ML21147A222 and ML21203A192, respectively), 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."

On July 23, 2021 (ADAMS Accession No. ML21197A063), the NRC issued an audit plan, which provided the list of requested documents and other details pertaining to the audit. An audit team, consisting of NRC staff and two contractors from the Southwest Research Institute (SwRI), conducted a remote regulatory audit to support the review of the LAR from August 10-12, and August 17, 2021. The audit was conducted following the guidance in the Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, Revision 1, "Regulatory Audits" (ADAMS Accession No. ML19226A274).

The regulatory audit is a planned license or regulation-related activity that includes the examination and evaluation of primarily non-docketed information. The regulatory audit is conducted with the intent to gain understanding, to verify information, and/or to identify information that will require docketing to support the basis of a licensing or regulatory decision. Performing a regulatory audit of the licensee's information is expected to assist the NRC staff in efficiently conducting its review or gain insights on the licensee's processes or procedures. Information that the NRC staff relies upon to make the safety determination must be submitted on the docket. However, the NRC staff may review supporting information retained as records under Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.71, "Maintenance of records, making of reports," and/or 10 CFR 54.37, "Additional records and recordkeeping requirements," which, although not required to be submitted as part of the licensing action,would help the NRC staff better understand the licensee's submitted information.

2.0 AUDIT ACTIVITIES

The NRC audit team consisted of the following NRC staff members and the contractors from SwRI. The NRC audit team held an entrance meeting on Tuesday, August 10, 2021, with the licensee's staff and contractors. The list of Ameren participants is provided in Attachment 1. During the remainder of the audit, the NRC audit team participated in technical discussions with the licensee based on discipline according to Section 3.0 of the audit plan.

List of NRC Participants

Division	Branch	NRC staff
Division of Safety Systems	Technical Specifications Branch	Steve Smith Andrea Russell
	Nuclear Methods and Fuel Analysis Branch	Ben Parks
Division of Risk Assessment	Probabilistic Risk Assessment Licensing Branch B	Bob Vettori Shilp Vasavada
Division of New and Renewed Licenses	Corrosion and Steam Generator Branch	Paul Klein Matt Yoder Samantha Platt (observer)
	Vessels and Internals Branch	John Tsao
Division of Operating Reactor Licensing	Plant Licensing Branch IV	Mahesh Chawla Bhagwat Jain (observer)
SwRI	Contractor	Osvaldo Pensado Stuart Stothoff

3.0 RESULTS OF THE AUDIT

Technical discussions were focused on the following major areas (Attachment 2 provides the summary of the technical discussions):

1. General Information and Licensing Bases
2. Debris Generation/Zone of Influence (Excluding Coatings)
3. Transport
4. Head Loss and Vortexing
5. Net Positive Suction Head (NPSH)
6. Coatings
7. In-Vessel Evaluation
8. Chemical Effects
9. Risk-Informed Basis
10. Defense in Depth and Safety Margin
11. LAR, Exemption Request, and Performance Monitoring Program

The NRC audit team participated in an audit exit meeting with the licensee on Tuesday, August 17, 2021, where the NRC staff provided a brief summary of the technical discussion conclusions.

The NRC staff provided a brief conclusion of the audit objectives that were met and details on the path forward. There were no open items in the discussion and no deviation from the audit plan. The licensee agreed to provide a supplement to the application to address audit discussion points.

Additional information needed to support the LAR review, if necessary, will be formally communicated by the NRC staff using the RAI process in accordance with NRR Office Instruction LIC-101, Revision 6, "License Amendment Review Procedures" (ADAMS Accession No. ML19248C539).

Attachments:
1. List of Ameren Participants
2. Summary of Discussions

LIST OF AMEREN PARTICIPANTS

Union Electric Company dba Ameren Missouri, Callaway, Team

Roger Andreasen	Engineering Design Group Supervisor
Justin Hiller	Probabilistic Risk Assessment (PRA) Group Supervisor
Jim Kovar	Senior Licensing Engineer
Tom Elwood	Supervisor, Regulatory Affairs
Steve Meyer	Manager, Regulatory Affairs
Jerry Doughty	PRA Engineer
Jonathan Cordz	Safety Analysis Engineer
Bruce Letellier	Consultant (Serco North America)
Jainisha Shah	Consultant (Serco North America)
Jeremy Tejada	Consultant (SIMCOM Solutions)
Matthew Ballan	Consultant (SIMCOM Solutions)

Appendix – Summary of Discussions

In addition to the items discussed below, the U.S. Nuclear Regulatory Commission (NRC) staff reviewed the information requested in Section 3.0 of the audit plan, dated July 23, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21197A063) regarding 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," submitted by Union Electric Company, dba Ameren Missouri (the licensee) for Callaway Plant, Unit No. 1 (Callaway). These reviews contributed to the NRC staff understanding of the issues discussed below.

The listing below retains the original questions in the Attachment to the audit plan and provides the NRC staff understanding of the discussion of these questions during the audit. The NRC staff expects that the items that require a revision to the LAR will be addressed via one or more supplements to the LAR. The NRC will request additional information, as necessary, to attain information required to make its safety and regulatory conclusions regarding the LAR after review of any supplements submitted by the licensee.

The NRC staff did not reach any regulatory conclusions during the audit. Regulatory decisions will be based on information that has already been received on the docket or a future licensee supplement to the LAR.

Note that the page numbers referenced in this Appendix are based on the PDF page numbers from the licensee's LAR dated March 31, 2021 (ADAMS Accession No. ML21090A184).

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.

The licensee stated that similar issues were addressed in a previous supplement to the LAR, but these related items were identified that require resolution. The licensee stated that items described above in this issue will be addressed in a supplement to the LAR.

(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.

A discussion between the NRC staff and the licensee clarified the licensee's intent for the requested exemptions.

(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).

The licensee stated that it will revise the FSAR to include the basis for the completion time and consider how to address the comment regarding the irrelevance of the number of sumps.

(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.

The licensee stated that it would add CASA Grande to the list of key methods in the FSAR. The identification of CASA Grande as a key method applies to integrated calculations. Less complex calculations may be accomplished using other methods.

(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

The licensee stated that it will update the list of key methods in the FSAR to include the items identified above. Some of the items may be included in guidance that is already referenced and may not need to be part of the updated list.

(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?

The licensee stated that it will consider updating the table to provide clarity for plant staff.

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.

The licensee clarified that ZOIs are centered at the center of the pipe for partial breaks.

(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.

The licensee stated that the failure locations identified by this item are considered to be secondary risk contributors. The NRC stated, that for completeness, these potential break locations should be identified and evaluated to ensure that they do not contribute significantly to risk. The licensee stated that it will review the submittal to make sure that break locations of this type are identified and quantitatively or qualitatively screened to show that the risk contribution is insignificant to the overall risk quantification.

(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.

The licensee stated that it will evaluate these potential break locations to determine whether previously evaluated locations bound their potential for debris generation. The licensee further stated that the results of the evaluation will be provided in a supplement to the LAR.

(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.

The licensee stated that the material was initially modeled as Nukon but was later identified to be FOAMGLAS®. The licensee explained that a supplement will provide additional details on how the debris source is treated in the risk analysis. The information will include results of testing conducted to assess the transport properties of the material, assumptions regarding its contribution to chemical effects, the acceptability of the amount of fiber assumed to be generated (based on the material density vs. Nukon density), and will address the consequences that could occur if the material was transported to the strainer and behaved as particulate. This item is related to Item 33 of this Appendix. The issue is also related to Item 20 of this Appendix as the resolution for Item 20 may result in additional margin in the particulate debris source term.

(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.

The licensee discussed actions that have been taken to reduce and control the amount of miscellaneous debris in containment. The licensee stated that these actions will be described in a supplement to the LAR.

(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.

The licensee discussed the reasons that RMI is not expected to transport to the strainers. The licensee stated that a supplement will include a description and basis for the assumption that significant RMI debris will not transport into the sump pits.

<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.

The NRC staff reviewed the transport calculation that was made available for review on the audit portal.

(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.

The licensee stated that it would review the statement and correct the description as necessary to accurately reflect the assumptions used in the qualified coatings transport calculation.

Head Loss and Vortexing (Attachment 3-2, Item 3f)

(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.

This entry covers Items 15, 18, and 19 since they cover issues related to testing and evaluation of the small break LOCA and large break LOCA conditions. The licensee stated that it will clarify the small break and large break LOCA cases for headloss testing and net positive suction head (NPSH) margin. The licensee further stated that a supplement will clarify the treatment of the small break and large break LOCA cases for NPSH, flashing, and deaeration. The licensee stated that there are three cases considered. Small break LOCA, large break LOCA with residual heat removal (RHR) only, and large break LOCA with RHR and containment spray. The licensee explained that the information in the supplement will include a summary of the relevant assumptions and parameters from the headloss calculation for strainer headloss, sump level (strainer submergence), credited containment pressure, and flowrates for each case. The licensee stated that it may include information on margins that are provided by the assumptions used in the headloss and NPSH calculations. For example, bubble collapse and re-absorption of gasses were not credited, maximum voiding is assumed in all NPSH cases, some test cases included significant flow margin compared to plant flow rates, and the debris headloss was not scaled to higher temperatures for any of the cases.

(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.

The licensee stated that it would correct the title of the figure in a supplement.

(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.

The NRC staff reviewed the headloss calculation and assumptions used to determine the amount of margin available above the pressure credited to suppress flashing and reduce deaeration for both the small break and large break LOCA cases. The licensee stated that they are crediting only the amount of pressure needed to prevent flashing, and that amount is small compared to the available pressure. The licensee stated that they will provide a summary of the relevant information from the calculation on the docket that will enable the NRC staff to reach a regulatory conclusion on this issue.

Net Positive Suction Head (NPSH)

(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.

See the entry for Item 15 of this Appendix.

(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.

See the entry for Item 15 of this Appendix.

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.

The licensee stated that there was a large amount of unqualified coatings predicted to transport to the strainer. The maximum mass predicted to deposit on the strainer resulted in negative margins in the code acceptable stress values for the strainer structural strength calculation. The licensee chose to reduce the stress by decreasing the mass load of unqualified coatings that can transfer to the strainer by reconsidering the potential for this unqualified coating system to remain in place following a LOCA. The licensee stated that the coating was initially applied as a qualified system but was later determined to be unqualified by the vendor because of potential issues with the preparation of the substrate to which the coating is applied. There were no issues identified with the qualification of the materials in the Carboline 193LF primer or 191HB topcoat. The licensee stated that they performed pull tests during the 2018 refueling outage and found that the coating adhered at forces greater than those required by the American National Standards Institute standard. The NRC staff requested the licensee to provide the adhesion values. The licensee stated that the values were significantly higher than the acceptance limit. The licensee explained that the coating system will not be reclassified as qualified but will be evaluated to show that it is unlikely to contribute to strainer debris loading. Coating adhesion was credited only for the structural calculation. However, the licensee may also consider the adhesion of this coating system to evaluate the particulate debris load that may result from FOAMGLAS® as discussed in items 10 and 33 of this Appendix, to support the assertion that strainer tests bound possible particulate loads. The licensee stated that they will provide the information in a supplement so that the NRC staff can come to a regulatory conclusion regarding the adhesion of the coating system following a LOCA and its potential effect on relevant calculations.

(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.

The licensee stated that the LAR contains the particle size distributions and provided examples of how the coatings amounts were calculated to reflect the volume of coatings debris that could be generated in the plant. The licensee also provided examples of how the coatings were scaled to the test surrogates based on the surrogate density.

<u>In-Vessel</u>

(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.

The licensee provided a more detailed description of the cases that were evaluated for in-vessel debris effects. The following is a summary of the licensee's description. The cases evaluated were minimum possible sump switchover time with maximum safeguards, full dual train, and one CSS train failed. Cases were analyzed that considered recirculation via CSS back to the pool and no recirculation back to the pool. The cases were also analyzed using both the minimum and maximum plant-specific values of K_{split} from WCAP-17788-NP, "Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090)," Volume 1, Revision 0. Westinghouse, July 2015 (ADAMS Accession No. ML15210A669). Minimum sump volume was assumed in the evaluation. The calculations also assume that all fiber that transported to the core is retained there. CSS flow rates were minimized to reduce the amount of fiber recirculated through the system. CSS starts 7.75 minutes after RHR swapover in the plant, and that this timing is modeled in the fiber mass balance calculation. Any fiber that recirculates to the sump pool via containment spray is immediately assumed to be mixed in the pool. The full flow rate was maintained through the strainer during the penetration test to ensure some margin in the test results. Based on the discussions during the audit, the NRC staff understands the test conditions, the scenarios considered and modeled in the evaluation, and how the strainer fiber penetration test and evaluation relate to each other.

(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.

The NRC staff reviewed the fiber penetration test report. The licensee performed three preliminary tests and a final quality assured test (Test 2) to quantify penetration. The NRC staff was unable to confirm that the model used to calculate the amount of penetration at various strainer loads was realistic or conservative compared to the test results. Specifically, the NRC staff identified that Equation A-50 in the strainer penetration test report (ALION-CAL-CEC-9345-003, Rev. 0) underpredicted the amount of fiber penetration when compared to strainer fiber penetration Test 2 data. The licensee stated that equation A-50 was not directly used to calculate the fiber amounts in the core for the submittal, but that the issue would be investigated because Equation A-50 was pertinent for conditions in the tests. Differences between Equation A-50 and Test 2 results may indicate artifacts in estimates of parameters of empirical fiber penetration functions. The licensee will provide information in a supplement that includes details on the method used to calculate the fiber penetration as the strainer fiber loading changes so that the staff can confirm that the values are realistic when compared to the test results.

(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.

The licensee stated that CSS will not be secured by any operating procedure until pressure is reduced below the threshold. The system might be turned off relatively early for smaller breaks that do not challenge debris limits. For the case questioned in this issue, redundancy and single failure considerations would ensure that one train of CSS is running. There is no design basis case where there would be no CSS running following a large break LOCA for at least the time until hot-leg switchover.

(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.

The licensee stated that it would clarify this statement in a supplement.

(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.

The licensee stated that the value for decay heat applies a 1 sigma uncertainty, which equates to 2 percent at the calculated value. The licensee stated that this information will be provided in a supplement.

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.

The licensee stated that 24 hours is the earliest time that precipitation may occur based on the WCAP-17788-P test group that was used to represent Callaway. The licensee statement that chemicals will not form prior to 6 hours was related to the 6-hour sample time from the WCAP-17788-P autoclave tests where no precipitation was seen for any of the autoclave samples for Callaway. The licensee will clarify in a supplement that chemical precipitates are not expected before either 6 or 24 hours.

Additionally, the NRC staff noted that the licensee stated that the plant is represented by Test Group 36 from WCAP-17788-P. The maximum pH for Callaway was reported to be 7.6 and the post-LOCA pool is buffered with trisodium phosphate . The FSAR retains a bounding value of 9 when sodium hydroxide spray was the chemical used for post-LOCA sump buffering. Table 4-1 in WCAP-17788-P states that the maximum pH for Test Group 36 was 9.2. The NRC staff questioned the applicability of Test Group 36 to Callaway since the current plant specific pH value (7.6) is significantly lower than the maximum value for that test group. When considering precipitation for the in-vessel analysis there are competing effects due to increased aluminum corrosion and increased aluminum solubility at higher pH levels. The licensee stated that additional trisodium phosphate test group data from WCAP-17788-P at lower pH levels for plants with similar aluminum amounts will be reviewed to verify that the assumption of 24 hours as the earliest precipitation time remains valid. The licensee stated that the results of the comparison will be included in a supplement.

(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).

The licensee stated that the flow chart contained an error, and that the flow chart should have progressed through path 15 for no settlement instead of path 14 since settlement was not credited in the plant specific analysis. The licensee stated that headloss testing was performed such that full transport of chemical effects was ensured. The licensee stated that the flow chart will be corrected in a supplement.

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.

The licensee stated that information in the NFPA-805 and 10 CFR 50.69 applications is applicable to the PRA used in the risk-informed evaluation. The licensee stated that they recently had an updated 10 CFR 50.69 submittal, and that all associated facts & observations have been closed. The licensee will include this information in a supplement and confirm that it remains applicable to the risk-informed LAR.

(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.

The licensee stated that an update was made to the PRA that will allow removal of this note. The modeling issue has been corrected. The licensee also stated that the risk does not change significantly when using the updated model. An updated table and a description of the issue and how it was corrected will be provided in a supplement.

(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.

The license stated that the failure rate comes from NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants." The licensee stated that it will determine the impact of the use of plant-specific failure rates. The licensee stated that in all ECCS injection lines, the first valve is a check valve that is normally closed at power. No manual valves are credited as the first isolation valve in the risk-informed analysis and all valves defined as first isolation valves are normally closed at power. The licensee will add a definition of first isolation valve, and indicate that these valves are normally closed, via a supplement.

(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.

The licensee stated that the valves are insulated. The licensee stated that CASA Grande assumes that there is additional insulation included at the valves because of the physical layout of the insulation system. The initial insulation amounts at valve locations were developed for South Texas Project (STP), and the Callaway values could be different. Therefore, the licensee considered it appropriate to perform a sensitivity study for the valve insulation amounts. The licensee stated that they performed a sensitivity study for the insulation at valves and pipe hangers and noted that the changes are small. The licensee also noted that there is 50 lbm fiber margin included for every break. Item 38 is related to this issue. The licensee will provide a description of the logic for performing this sensitivity study and how it was performed in a supplement.

(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®

In a supplement the licensee will 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. See Item 10 for a discussion of this issue.

<u>Probabilistic Risk Assessment (PRA)</u>

(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.

The licensee stated that the debris generation from the initiating events discussed in this issue are either bounded or approximated by results from previously evaluated welds or are too small to have a significant effect on risk. The licensee also stated that water hammer is not postulated to result in a LOCA and that internal fires and floods do not create consequential piping LOCAs inside containment. Non-piping LOCAs inside containment are limited to locations where debris generation is insufficient to challenge recirculation sump functionality. The licensee also noted that it takes a very large break to exceed the maximum analyzed fiber load. The NRC staff stated that the risk analysis should have screening arguments for all potential initiating events, including those listed in this question, to assure that all potential failure modes were assessed. A thorough initiating event screening ensures that the analysis is complete. The licensee will provide a description of how these initiating events were screened out in a supplement.

(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.

The licensee stated that they performed limited studies for the main steam (MS)/feedwater (FW) breaks using typical 17D ZOI and a smaller ZOI based on lower steam pressure (vs. RCS pressure). Some studies also considered a smaller ZOI based on flow restrictors in the steam lines. The results of the studies were used to quantify the amount of fiber that could reach the strainers for one and two train operation. The licensee stated that the MS line break frequency comes from the PRA and is not location based. Generic values were used for the MS/FW line break frequency and are based on the linear feet of pipe (ratio inside to outside containment). Double-ended guillotine breaks were assumed for all breaks. The licensee stated that the increase in risk from these breaks is negligible (two orders of magnitude lower than the base risk). The licensee will discuss how these breaks screened out in a supplement. A statement that the breaks were screened out from consideration in the risk-informed analysis will be included.

(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," [Rev. 2, May 2011(ADAMS Accession No. ML100910006)]. 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.

The licensee and NRC staff discussed this issue in a broader sense under item 38. Specifically, for this issue, the NRC staff noted that the arithmetic mean exceeds the Region III boundary by a significant amount. The NRC staff also noted that all plants using a risk-informed methodology for resolution of GL 2004-02 have performed sensitivities to address arithmetic mean values. The sensitivity studies allow the NRC to make informed decisions by adding to the understanding of how alternative models impact the estimated risk. The NRC staff stated that they are interested in mean values as discussed in RG 1.174, not the tails. The licensee and NRC staff discussed using the 50 lbm fiber margin that is included in each break scenario to address the arithmetic mean sensitivity result. The NRC staff cautioned counting margins twice but noted that the study of the fiber margin might provide a good perspective on how it impacts risk as long as the manner that the fiber margin will be used by the licensee is recognized. The licensee stated that the 50 lbm margin was chosen to prevent downstream effects scenarios from failing and to keep the arithmetic mean sensitivity risk below 5×10^{-6} per year. In general, sensitivity studies that indicate significant increases in risk should include discussions on available margin and available mitigation actions. It was also recognized that there is margin in the baseline risk value due to the assumption that only a single train operates when it is more likely to have two trains operating. The licensee will provide an evaluation of the arithmetic mean sensitivity in a supplement. The licensee should provide a basis for the conclusions that margins available indicate that risk remains very low (R-III) even though the sensitivity result is in Region-II.

(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.

The licensee stated that the Bounded Johnson values were used only for illustration purposes to show the curve between the points taken from NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies through the Elicitation Process." The Bounded Johnson values were used simply to help with visualization and were not used for any calculations.

(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.

The licensee concluded that the guidance (models and methods used) contains adequate margin so that the evaluation of uncertainties is not required. The NRC staff stated that STP and Vogtle Electric Generating Plant (Vogtle) both performed parametric studies on several parameters to ensure that they understood which parameters could significantly impact risk. Identification of parameters that can have a significant effect allows them to be inspected to ensure that the values used were adequately estimated. The NRC staff stated that the licensee should provide a summary of why the results are conservative. For example, why is it certain that there is no risk contribution from debris in the reactor vessel? Explore parameters that affect the result and how sensitive the analysis is to these parameters. Concentrate on parameters that strongly drive the risk result. Provide assurance that the parameters have been treated acceptably. Parametric sensitivities should be discussed along with the limiting factors and the basis for why they have been addressed adequately. The discussion should be qualitative with quantitative results for support. Alternately, if all parameters are shown to be biased conservatively, it is possible that no sensitivities would be required. The sensitivities are done to identify which parameters are important and as additional insight to the conclusion that Callaway operates in RG 1.174 Region-III. The licensee will consider an acceptable method of addressing this issue and provide information in a supplement.

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.

The licensee stated that they used break frequencies in NRC NUREG-1829, that did not include special treatment of dissimilar metal welds. The licensee further stated that the plant has a program to inspect, evaluate, and mitigate the dissimilar metal welds. Special treatment of the dissimilar metal welds in the CASA Grande analyses would be considered a bottom up approach, which was abandoned by STP, the pilot risk-informed plant. The licensee indicated that NUREG-1829 has some recommendations for treatment of dissimilar metal welds.

However, the licensee did not use the uneven frequency weld break recommendations in the analysis. Based on discussions during the audit, the licensee stated that it will provide a description of its dissimilar metal weld inspection and evaluation program and believes that American Society of Mechanical Engineers Code Case N-770-5, “Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated with UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities, Section XI, Division 1,” as conditioned by 10 CFR 50.55a(g)(6)(ii)(F) must be use for the inspection of dissimilar metal welds. The licensee stated that it will discuss the inspection, evaluation, and mitigation of these welds in a supplement. In addition, the licensee will identify any unmitigated welds in the list of 60 critical welds in the submittal.

(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.

The licenses stated that it will discuss these periodic inspection and monitoring programs in its supplement, including the inspections that will be performed for license renewal to monitor the structural integrity of the RCS piping and components as part of defense-in-depth measures.

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.

The licensee stated that it will describe configuration management, the periodic PRA review (discussed in Item 43 below), periodic strainer inspections, containment walkdowns, and similar programs and inspections in a supplement. The licensee stated that it will explain how the programs and actions ensure that the risk-analysis remains valid.

(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.

The key elements are the items listed in the FSAR markup in Section 6.3A.2.1 (Page 101-109 of LAR Enclosure 2). The licensee stated that this list will be updated to include debris limits (it will likely refer to Table 6.3A-2 for debris limits) and CASA Grande in a supplement. (See Question 4 of this Appendix.)

(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.

Callaway will consider changing the FSAR Section 6.3A.2 to be similar to the Vogtle response to Limitation and Condition 8 in its risk-informed LAR for GL 2004-02. The intent of the identification of relevant elements that should be considered for update is to assure that the risk assessment remains valid. The update will be provided, as determined by the licensee, in a supplement.

(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.

The discussion of this item focused on whether reporting should be based on exiting Region III or Region II. The licensee will evaluate this issue and determine whether it intends to use Region II or Region III and clarify this point with a justification in a supplement.

SUBJECT: CALLAWAY PLANT, UNIT NO. 1 – AUDIT SUMMARY FOR LICENSE AMENDMENT REQUEST AND REGULATORY EXEMPTIONS FOR A RISK-INFORMED APPROACH TO ADDRESS GENERIC SAFETY ISSUE-191 AND RESPOND TO GENERIC LETTER 2004-02 (EPID L-2021 LLA 0059 AND EPID L-2021-LLE-0021 DATED: SEPTEMBER 14, 2021.

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