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

March 20, 2023

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 – REGULATORY AUDIT PLAN AND SETUP OF ONLINE REFERENCE PORTAL FOR LICENSE AMENDMENT REQUEST AND EXEMPTION FOR FUEL TRANSITION TO FRAMATOME GAIA FUEL (EPID L-2022-LLA-0150 AND EPID L-2022-LLE-0030)

Dear Mr. Diya:

By application dated October 12, 2022 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML22285A115), as supplemented by letter dated December 1, 2022 (ML22335A497), Union Electric Company, doing business as Ameren Missouri (the licensee), submitted a license amendment request (LAR) and an exemption request for Callaway Plant, Unit No. 1.

The proposed amendment would revise the technical specifications to allow use of Framatome GAIA fuel with M5 as a fuel cladding material. Since the Framatome GAIA fuel will use M5 fuel rod cladding, a Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and 10 CFR Part 50 Appendix K exemption request is included as part of this LAR.

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the licensee's LAR and exemption request and determined that a regulatory audit would assist in the timely completion of the review. The NRC staff will conduct a regulatory audit to support its review in accordance with the enclosed audit plan. A regulatory audit is a planned activity that includes the examination and evaluation of primarily non-docketed information.

To improve the efficiency of the NRC staff reviews, the licensee's representatives and the NRC staff have discussed having an audit using an online reference portal. The NRC staff plans to initially conduct a review of the documentation provided on the portal. The audit meeting with the licensee is scheduled from April 3-5, 2023. The online reference portal would allow the NRC staff to audit documents referenced in the request to determine whether the information included in the documents is necessary to reach a safety conclusion on the proposed

Enclosure 1 to this letter contains proprietary information. When separated from Enclosure 1, this document is DECONTROLLED.

amendment. The licensee will be formally requested to submit information needed to reach a safety conclusion 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 NRC 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 attachment to 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 via email. Please provide the NRC staff with access to the portal and send me the information needed for access, 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 audit. Please provide written confirmation that Ameren Missouri agrees to the terms and conditions set forth in this letter.

The audit plan is enclosed. The NRC staff has determined that the audit plan contains proprietary information pursuant to 10 CFR 2.390, "Public inspections, exemptions, request for withholding." Proprietary information, which is provided as enclosure 1, is indicated by **bold** text enclosed with **[[double brackets]]**. Accordingly, the NRC staff has prepared a redacted publicly available non-proprietary version of the audit plan, which is provided as enclosure 2.

If you have any questions, please contact me at 301-415-8371 or via email 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:
1. Regulatory Audit Plan (proprietary)
2. Regulatory Audit Plan (non-proprietary)

cc: Listserv without Enclosure 1

ENCLOSURE 2

(NON-PROPRIETARY)

REGULATORY AUDIT PLAN

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT NO. 1.

DOCKET NO. 50-483

Proprietary information pursuant to Section 2.390 of title 10 of the *Code of Federal Regulations* has been redacted from this document.

Redacted information is identified by blank space enclosed within [[double brackets]].

REGULATORY AUDIT PLAN

TO SUPPORT REVIEW OF LICENSE AMENDMENT/EXEMPTION REQUEST

FOR FUEL TRANSITION TO FRAMATOME GAIA FUEL

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

1.0 BACKGROUND

By application dated October 12, 2022 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML22285A115), as supplemented by letter dated December 1, 2022 (ML22335A497), Union Electric Company, doing business as Ameren Missouri (the licensee), submitted a license amendment request (LAR) and exemption request for Callaway Plant, Unit No. 1 (Callaway). The proposed amendment would revise the technical specifications (TSs) to allow use of Framatome GAIA fuel with M5 as a fuel cladding material. Since the Framatome GAIA fuel will use M5 fuel rod cladding, a Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and 10 CFR Part 50 Appendix K, "ECCS [Emergency Core Cooling System] Evaluation Models," exemption request is included as part of this LAR. Callaway is proposing to load a limited number of Framatome GAIA fuel assemblies starting in operating cycle 27 to obtain in-core performance data and acquire operational experience associated with the GAIA fuel design.

2.0 REGULATORY AUDIT BASES, SCOPE, AND METHODOLOGY

A regulatory audit is a planned license or regulation-related activity that includes the examination and evaluation of primarily non-docketed information. The audit is conducted to further identify material the U.S. Nuclear Regulatory Commission (NRC) staff may determine is warranted on the docket to support the basis of a licensing or regulatory decision. Performing a regulatory audit is expected to assist the NRC staff in efficiently conducting its review of the LAR and to gain insights on the licensee's processes and procedures. Information that the NRC staff relies upon to make the safety determination must be submitted on the docket. The basis of this remote audit is to support the NRC staff review of the licensee's LAR and exemption request for Callaway.

The remote audit will be performed consistent with NRC Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, Revision 1, "Regulatory Audits," dated October 31, 2019 (ML19226A274). The NRC staff plans to initially conduct a desk audit to review the documentation provided on the portal. The audit meeting with the licensee is scheduled for April 3-5, 2023. An audit was determined to be the most efficient approach toward a timely resolution of issues associated with this LAR review, since the NRC staff will have an opportunity to minimize the potential for multiple rounds of requests for additional information (RAIs) and ensure that no unnecessary burden will be imposed by requiring the licensee to address issues that are no longer necessary to make a safety determination. Audit review items include the following:

- Gain a better understanding of the detailed calculations, analyses, and bases underlying the LAR and confirm the NRC staff's understanding of the LAR.

- Gain a better understanding of plant design features and their implications for the LAR.

- Identify any information needed to enable the NRC staff's evaluation of the technical acceptability of the probabilistic risk assessment used for this application.

- Identify any information needed to enable the NRC staff's evaluation of whether the proposed changes challenge design-basis functions or adversely affect the capability or capacity of plant equipment to perform design-basis functions.

- Identify questions and requests that may become formal RAIs per NRR Office Instruction LIC-115, "Processing Requests for Additional Information," Revision 1, dated August 9, 2021 (ML21141A238).

3.0 INFORMATION AND OTHER MATERIALS NECESSARY FOR THE REMOTE REGULATORY AUDIT

The NRC staff will request information as needed. The NRC staff will use an "audit items list" to identify the information (e.g., methodology, process information, and calculations) to be audited and the subjects of requested interviews and meetings. The NRC staff requests the licensee to have the requested audit information listed in the audit document request to be readily available and accessible for the NRC staff's review via a web-based portal.

The NRC staff requests the licensee to have the information referenced in the attachment to this audit plan and any supplemental information be available and accessible for the NRC staff's review via an online reference portal as soon as feasible within a few days of the issuance of this audit plan.

4.0 AUDIT TEAM

DIVISION	Branch	NRR Staff/	Title	Email
DORL[1]	LPL4[2]	Mahesh Chawla	Project Manager	Mahesh.Chawla@nrc.gov
		Dennis Galvin	Project Manager (Backup)	Dennis.Galvin@nrc.gov
DSS[3]	SNSB[4]	Ahsan Sallman	Senior Nuclear Engineer	Ahsan.Sallman@nrc.gov
		Santosh Bhatt	Nuclear Engineer	Santosh.Bhatt@nrc.gov
	SFNB[5]	Mathew Panicker	Nuclear Engineer	Mathew.Panicker@nrc.gov
		Jeremy Dean	Senior Nuclear Engineer	Jeremy.Dean@nrc.gov
	STSB[6]	Robert Elliott	Senior Safety and Plant System Engineer	Robert.Elliott@nrc.gov
		Ravi Grover	Safety and Plant System Engineer	Ravinder.Grover@nrc.gov
DRO[7]	IQVB[8]	Andrea Keim	Reactor Operations Engineer	Andrea.Keim@nrc.gov
NMSS[9]	ENRB[10]	Gerry Stirewalt	Senior Environmental Project Engineer	Gerry.Stirewalt@nrc.gov
		Donald Palmrose	Senior Reactor Engineer	Donald.Palmrose@nrc.gov

5.0 SPECIAL REQUESTS

The NRC requests access to requested documents and information through a web-based portal that allows the NRC staff to access documents over the internet. The following conditions associated with the online portal must be maintained while the NRC staff has access to the online portal:

- The online portal will be password-protected. A separate password will be assigned to each member of the NRC staff participating in the audit.

- The online portal will prevent the NRC participants from printing, saving, downloading, or collecting any information directly from the online portal.

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[1] Division of Operating Reactor Licensing
[2] Plant Licensing Branch IV
[3] Division of Safety Systems
[4] Nuclear Systems Performance Branch
[5] Nuclear Methods and Fuel Analysis Branch
[6] Technical Specifications Branch
[7] Division of Reactor Oversight
[8] Quality Assurance and Vendor Inspection Branch
[9] Office of Nuclear Material Safety and Safeguards
[10] Environmental Review License Renewal Branch

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

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

6.0 DELIVERABLES

The NRC staff access to the online portal should be terminated by July 31, 2023. The NRC staff will issue an audit summary report within 90 days of the completion of the audit. The NRC staff will develop any RAIs, as needed, consistent with the guidance in NRR Office Instruction LIC-115, Revision 1, and issue such RAIs separately from audit-related correspondence.

7.0 AUDIT QUESTIONS

ENRB (Environmental Protection) Question

Regulatory Basis:

NRC staff needs a discussion for each criterion under 10 CFR 51.22(c)(9) with appropriate references to the LAR dated October 12, 2022, and appropriate past license renewal documents. This need is related to whether the changes proposed in the LAR would result in significant changes to hazards, effluent releases (with consequent environmental monitoring), and individual or cumulative occupational radiation exposure, and thus, require an environmental review with an EIS or EA.

Deficiency:

In section 5.0, "Environmental Evaluation," of enclosure 1, "Description and Assessment of the Proposed Change," to the letter dated October 12, 2022, the licensee states that the proposed change related to the use of Framatome GAIA fuel with M5 cladding for Callaway does not violate either of the following three criteria defined in 10 CFR 51.22(c)(9): (i) The change does not involve a significant hazards consideration; (ii) There is no significant change in the types or a significant increase in the amounts of any effluent that may be released offsite; and (iii) There is no significant increase in individual or cumulative occupational radiation exposure. The licensee's citation of these three criteria appears to be the sole basis for its conclusion that the change allows a categorical exclusion from an environmental review pursuant to 10 CFR 51.22(b) and no EIS or EA need be prepared in connection with the proposed change. The licensee does not provide information to qualify how the three criteria cited are met to adequately justify that the change falls under a categorical exclusion and no environmental review with an EIS or EA is required.

Information Needed:

The licensee should discuss each criterion set forth under 10 CFR 51.22(c)(9) to document the basis for its conclusion that the proposed change related to the use of

Framatome GAIA fuel with M5 cladding for Callaway allows a categorical exclusion pursuant to 10 CFR 51.22(b) and no environmental impact statement (EIS) or environmental assessment (EA) is needed.

SNSB Questions

The review of the Callaway LAR for Framatome GAIA fuel transition is mainly based on the following NRC guidance, as well as appropriate sections of the General Design Criteria (GDC) listed in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," (SRP) including:
  - Section 4.2, "Fuel System Design," Revision 3, dated March 2007 (ML070740002)
  - Section 4.3, "Nuclear Design," Revision 3, dated March 2007 (ML070740003)
  - Section 4.4, "Thermal and Hydraulic Design," Revision 2, dated March 2007 (ML070550060)

The proprietary information in the audit items is identified by text in bold font and enclosed within double square brackets. **[[This is an example.]]**

ANP-3943P, "Callaway Small Break LOCA Analysis with GAIA Fuel Design," Revision 1, Audit Questions (attachments 11 (non-public) and 7 (public) of enclosure 1 to the letter dated October 12, 2022)

1. Table 3-1, "System Parameters and Initial Conditions," in ANP 3943P provides system parameters and initial conditions used in the small break (SB) loss-of-coolant accident (LOCA) (SBLOCA) analysis. Specify how these compare to the TS limits, where applicable.

2.. Section 3.3, "Description of Analytical Methods," of ANP 3943P, states, in part, that "two modeling deviations from approved SBLOCA methodology were included in this analysis…." The subsequent description provides description on the **[[**

]] and provides further explanation that **[[**

]].

Provide any additional details on the implementation of the logic scheme developed as well as examples for calculations performed with the new logic.

3. Section 3.5, "Safety Evaluation Compliance," of ANP 3943P states, in part, that the "NRC-approved supplemented EMF-2328 … method contains no restrictions." The 10 percent cold leg break size limitation applied to ANF-RELAP code that still applies to S-RELAP5 is discussed in attachment 13 to the LAR.

State how the limitations from section 4 of the safety evaluation (SE) for topical report (TR) BAW-10240(P)(A), "Incorporation of M5TM Properties in Framatome ANP Approved Methods," Revision 0, apply to this analysis and how they are dispositioned.

4. Section 3.3 of ANP 3943P states, in part, that "RODEX2-2A code was used to determine the burnup dependent initial fuel rod conditions for the system calculations" and "S-RELAP5 code was used to predict the primary and secondary system thermal-hydraulic and hot rod transient response."

   Confirm the versions of the codes used above are the same as the approved version in TR EMF-2328(P)(A) Supplement 1, "PWR [Pressurized-Water Reactor] Small Break LOCA Evaluation Model S-RELAP5 Based."

5. The analysis presented, and the subsequent results used to demonstrate compliance with the 10 CFR 50.46(b) criteria for SBLOCA are for the GAIA fuel design. Provide any considerations for mixed core design with co-resident fuel assemblies that have different form loss coefficients for the grid spacers and the upper and lower tie plates. Justify the SBLOCA analysis performed using the GAIA fuel data would be valid for the mixed core.

<u>ANP-3944P, "Callaway Realistic Large Break LOCA Analysis with GAIA Fuel Design," Revision 1 Audit Questions (attachments 10 (non-public) and 6 (public) of enclosure 1 to the letter dated October 12, 2022)</u>

1. Section 1.0, "Introduction," of ANP-3944P states:

   a. The analysis also addresses typical operational ranges or TS limits (whichever is applicable) with regard to **[[**

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   Specify parameters for which TS limits are used and for those for which operational ranges are used in the analysis.

2. Section 1.0 of ANP-3944P states, "The analysis explicitly analyzes fresh and once-burned fuel assemblies."

   Specify which fuel assemblies in the core are fresh and which ones are once-burned.

3. Section 1.0 of ANP-3944P states:

   > The analysis uses the Fuel Swelling, Rupture, and Relocation (FSRR) model to determine if cladding rupture occurs and evaluate the consequences of FSRR on the transient response.

   Describe the consequences of FSRR on the large break LOCA (LBLOCA) transient response.

4.. Section 3.1, “Acceptance Criteria,” of ANP-3944P states:

> The final two criteria [10 CFR 50.46(b)(4) and (b)(5)], coolable geometry and long-term cooling, are treated in separate plant-specific evaluations.

Provide the plant specific evaluations demonstrating the 10 CFR 50.46(b)(4) and (5) acceptance criteria are met.

5. Section 3.3, “Description of Analytical Models,” of ANP-3944P mentions some differences from the NRC-approved EMF-2103(P)(A) methodology included in the LBLOCA analysis. Provide responses to the following:

(a) [[

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6. Section 3.6, “Plant Description,” of ANP-3944P states:

> The results used to demonstrate compliance with the 10 CFR 50.46(b) criteria are only applicable to the Framatome fuel product. However, the analysis includes considerations for the mixed core scenario. **[[**
>
> **]]**

Provide responses to the following:

(a) The following two statements in the above paragraph appear to conflict with each other. Explain or revise the statements to remove the conflict.

> The results used to demonstrate compliance with the 10 CFR 50.46(b) criteria are only applicable to the Framatome fuel product.

[[

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(b) Explain why the analysis results for the GAIA fuel in ANP-3944P, Revision 1, table 4-4, "Compliance with 10 CFR 50.46(b)," would be considered limiting for the mixed core.

7. Refer to ANP-3944P, table 3-1, "EMF-2103(P)(A), Revision 3, SE Limitations Evaluation," in response to Limitation No. 7. It states: **[[**

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(a) Explain which correlation for MLO is used and provide further discussion on how the limitation is met in the demonstration case and case set.

(b) Explain how this limitation will be satisfied in the reload analysis case.

8. The realistic LBLOCA (RLBLOCA) demonstration analysis is based on the GAIA fuel data given in ANP-3944P, table 4.1, "RLBLOCA Analysis – Plant Parameter Values and Ranges," section 1.1. Also, as stated in section 3.6, the results used to demonstrate compliance with 10 CFR 50.46(b) are only applicable to Framatome GAIA fuel. However, the mixed core in the first reload cycle will be mostly Westinghouse Vantage 5 fuel with fuel data different from the GAIA fuel data. Justify that the RLBLOCA analysis is valid for a mixed core. Also, if an analysis similar to the demonstration RLBLOCA analysis is performed for the mixed core reload cycle using the GAIA fuel data, justify that the analysis would be valid for the mixed core.

9. Refer to ANP-3944P, table 4-1, sections 2.0 and 3.0, and provide justification if any of the plant operating and accident boundary conditions are different from their values in the analysis of record (AOR). Also, justify if the conservatism in any of these parameters is reduced.

10. Due to differences in the fuel decay heat and the stored sensible energy in the reactor internals (for example in fuel assemblies and other components), the fuel transition from a full Westinghouse core to a mixed Westinghouse and Framatome GAIA core may impact the following LBLOCA mass and energy (M&E) release, containment pressure and temperature response, and net positive suction head (NPSH) AORs:

(i). M&E release analyses for LBLOCA (Callaway Final Safety Analysis Report-Standard Plant (SP) FSAR section 6.2.1.3).

(ii). LBLOCA containment pressure and temperature response (Callaway FSAR-SP section 6.2.1.1.3).

(iii). Minimum containment pressure analysis for performance capability studies on ECCS (Callaway FSAR-SP section 6.2.1.5).

(iv). Available NPSH for containment spray pumps, and residual heat removal pumps (Callaway FSAR-SP table 6.2.2-7)

For the transition from full Westinghouse core to a mixed Westinghouse and GAIA core, provide a response to the following:

(a) Justify the above (i) through (iv) AOR remain bounding.

(b) In case any of the above (i) through (iii) AORs are not bounding, provide a discussion of the impact, the AOR results, and revised results along with changes in the method(s) of analysis, inputs, and assumptions.

<u>ANP-3969P, "Callaway Non-LOCA Summary Report," Revision 2, Audit Questions (attachments 12 (nonpublic) and 8 (public) of enclosure 1 to the letter dated October 12, 2022)</u>

1. In ANP-3969P, the following statement appears in sections 5.13, 5.14, and 5.15:

   > The consequences of this event primarily depend on initial operating conditions, plant-related systems and capacities, and decay heat.

   Confirm that the GAIA fuel decay heat is bounded by the AOR fuel decay heat. If it is not bounded, provide the impact of these events on the figures of merit because of different GAIA fuel decay heat.

2. ANP-3969P, section 5.14, "Loss of Normal Feedwater Flow (FSAR-SP 15.2.7)," states:

   > If an alternative supply of feedwater is not supplied to the plant, core residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer would occur, resulting in a substantial loss of water from the RCS [reactor coolant system].

   Would the pressurizer become "water solid" during FSAR-SP 15.2.7 loss of normal feedwater flow event, and through which path would the RCS water be lost?

3. ANP-3969P, section 5.15, "Feedwater System Pipe Break (FSAR-SP 15.2.8) states:

   > Depending upon the size of the break and the plant operating conditions at the time of the break, the event could cause an RCS cooldown which is evaluated in Section 5.5 and Section 5.7.

   The above statement refers to ANP-3969P, sections 5.5 and 5.7 which provide an evaluation of steam system piping breaks that can cause RCS cooldown. Explain how the feedwater line break FSAR SP 15.2.8 event could result in RCS cooldown.

4. ANP-3969P, section 3.9.2, "Methodology Changes," second bullet, last sentence states:

   > **[[**
   > **]]**

   Provide reference to the applications in which the approach was approved.

5. ANP-3969P, section 3.9.1, "Methodology Description," states:

   > For each non-LOCA transient event analysis, the nodalization, chosen parameters, conservative input and sensitivity studies are reviewed for applicability to the Framatome VQP [Vendor Qualification Program] in compliance with the safety evaluation report (SER) for Revision 0 of the non-LOCA topical report (Reference 1).

   The NRC-approved version of the non-LOCA topical report EMF-2310(P)(A) is Revision 1. What are the differences between Revision 0 and 1?

6. Refer to ANP-3969P, table 3-1, "Summary of Initial Conditions and Computer Codes Used," and list the differences between the initial conditions in this table from the initial condition used in the AOR as listed in FSAR-SP table 15.0.2. Justify the differences.

7. Explain how the following conditions which are number 2 through 9 listed in section 4.0 of the SER for EMF-2310(P)(A), Revision 1, are met:

   1. The boron dilution is assumed to occur at the maximum possible rate.

   2. The core burnup and corresponding boron concentration are selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

   3. All fuel assemblies are installed in the core.

   4. A conservatively low value is assumed for the reactor coolant volume.

   5. For analyses during refueling, all control rods are withdrawn from the core.

   6. For analyses during power operation, the minimum shutdown margin allowed by the technical specifications is assumed to exist prior to the initiation of boron dilution.

   7. For each event analyzed, a conservatively high reactivity addition rate is assumed taking into account the effect of increasing boron worth with dilution.

   8. Conservative scram characteristics are assumed (i.e., maximum delay time with the most reactive rod held out of the core).

8. Explain how the following limitations and conditions listed in section 4.0 of the SER for EMF-92-081(P)(A), "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," Revision 1, are met:

   1. The methodology includes a statistical treatment of specific variables in the analysis; therefore if additional variables are treated statistically,

Siemens Power Corporation (SPC) should re-evaluate the methodology and document the changes in the treatment of the variables. The documentation will be maintained by SPC and will be available for NRC audit.

2. The steam generator safety valve (SGSV) limit line provides an upper limit on the temperature range for setpoint verification. The upper limit on the temperature range should be adjusted to reflect the steam generator plugging level.

9. Explain how the following limitations and conditions listed in section 4.1 of SER for ANP-10341P-A, “The ORFEO-GAI and ORFEO-NMGRID Critical Heat Flux Correlations,” Revision 0, are met:

<u>Conditions</u>

1. The inlet subcooling must be greater than 0 degrees. This is to ensure that the burnout length is limited to the fuel region.

2. For ORFEM-NMGRID, Framatome should confirm that the reload calculation performed for set points, AOOs [anticipated operational occurrences], and accidents are far removed from the **[[ ]]** subregion. If the calculations are not far removed from this region, then Framatome must quantify the additional uncertainty of the region and apply that increased uncertainty in the analysis.

3. While both ORFEO-GAIA and ORFEO-NMGRID are approved over their entire application domain, this approval is given under the assumption that their use in the low quality region (i.e., equilibrium qualities below -0.1) has minimal impact on the limiting minimum DNBR [departure from nucleate boiling ratio] values. Limiting minimum DNBR is defined as the scenario in which the event is approaching the design limit. Application of the ORFEO-GAIA and ORFEO-NMGRID CHF [critical heat flux] correlations for events in which the limiting DNBR is sufficiently far from the design limit is not subject to this condition regardless of the local quality. Should this assumption no longer be true and should the low quality domain become a limiting domain, Framatome would need to provide additional analysis in quantifying the uncertainty in this domain.

<u>Limitations</u>

1. ORFEO-GAIA is approved for use in predicting the CHF downstream of GAIA and IGM [intermediate GAIA mixing grid] mixing grids in GAIA fuel. This prediction must be made in the subchannel code COBRA-FLX with the modeling option as specified in Table 5.1 of the TR with a design limit of 1.12 over the application domain specified in Table 2-2 of the initial submittal of the TR. The approved design limit contains a bias of 0.01 which the NRC staff believed was necessary to account for variations between the tested fuel assembly and the production fuel assembly which will be used in the reactor.

2. ORFEO-NMGRID is approved for use in predicting the CHF downstream of W 17x17 HMP [high mechanical performance] non-mixing grids and GAIA and IGM mixing grids in GAIA fuel. This prediction must be made in the subchannel code COBRA-FLX with the modeling option as specified in Table 5.1 of the TR with a design limit of 1.15 over the application domain specified in Table 2-5 of the initial submittal of the TR.

10. Explain how the following limitations and condition that are implicitly stated in section 4.0 of the SER for BAW-10231P-A, “COPERNIC Fuel Rod Design Computer Code,” Revision 1, are met:

    COPERNIC code is acceptable for MOX [mixed oxide] fuel licensing applications up to a WG Pu [weapons grade plutonium] content of 6 wt% and a peak rod average burnup of 50 GWd/MTHm [gigawatt days per metric ton of initial heavy metal].

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11. Explain how the following limitations and condition that are implicitly stated in the SER for XN-NF-75-21(P)(A), “XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Core Operations,” Revision 2, are met:

    1. XCOBRA-IIIC code is applicable to all transients in which flow reversals and recirculation do not occur. This excludes LOCA calculations.

    2. The XNB CHF correlation is restricted to homogeneous models for two-phase flow.

    3. XCOBRA-IIIC code is acceptable for homogeneous models for those options pertaining to PWRs in conjunction with the XNB CHF correlation.

    4. XCOBRA-IIIC code is acceptable for calculating transient AOOs and postulated accidents as described using the “snapshot” mode in which a series of steady state calculations are made. The “full transient” mode should give less conservative results, and an extensive evaluation would be required to assure that the 95/95 DNBR acceptance criterion is satisfied.

12. Explain how the following limitations and conditions 1, 2, 4, 5, and 6 listed in Section 4.1 of SER for ANP-10297P-A, Revision 0, Supplement 1PA, “The ARCADIA® Reactor Analysis System for PWRS Methodology Description and Benchmarking Results,” are met:

    1. The range of applicability of the ARCADIA® methodology is restricted to the fuel data provided in the TR, as supplemented, unless additional analysis and benchmarking is conducted to validate ARCADIA® to a fuel type not mentioned in the TR, as supplemented. (This is Condition 1 of the SE for the original ARCADIA® TR (ANP-10297P-A, Revision 0). It remains

applicable to Supplement 1, and it has been updated to include the expanded range of fuel data presented within Supplement 1).

2. The benchmarks provided in the ARCADIA® TR, as supplemented, include uncertainty verification for plants that use moveable incore, rhodium fixed incore, and Aeroball incore detectors. Framatome will evaluate at least three cycles of data relative to these criteria prior to licensing the first cycle with Framatome fuel with ARCADIA®. Additionally, application of ARCADIA® to a new uncertainty measurement system(s) would require review and approval by the NRC staff prior to implementation. (This is Condition 2 of the SE for the original ARCADIA® TR (ANP-10297P-A, Revision 0). It remains applicable to Supplement 1, and it has been updated to include the incore detector systems presented within Supplement 1).

3. Originally in ANP-10297P-A, Revision 0, and removed by its supplement 1PA.

4. For any changes made to the stand-alone version of COBRA-FLX™ that is implemented in ARCADIA® (the COBRA-FLX™ module), Framatome will revalidate ARCADIA® output using measured data from multiple plants and cycles. (This is Condition 4 of the SE for the original ARCADIA® TR (ANP-10297P-A, Revision 0). It remains applicable to Supplement 1).

5. The NRC staff finds ARTEMIS™ acceptably models the best estimate neutronic time dependent transient responses (e.g., power response to changes in Doppler, moderator, etc.), and that it is an acceptable tool for use in an evaluation model for non-LOCA SRP Chapter 15 events. However, use of ARTEMIS™ in an evaluation model for such events requires consideration of bounding conditions, inputs, limits, time-step sensitivities, etc., which are not included in Supplement 1. Therefore, as implied for ANP-10297P-A, Revision 0, this SE does not constitute approval of ARTEMIS™ as a stand-alone evaluation model for non-LOCA SRP Chapter 15 events. NRC review and approval of an associated evaluation methodology using ARTEMIS™ is required prior to its use in non-LOCA SRP Chapter 15 event licensing analyses.

6. Any changes made to the ARCADIA® code system must:

    a. ensure the validation suite acceptance criteria (Table 10-2 of Supplement 1) remain applicable,

    b. be consistent with the methodology described in ANP-10297P, as supplemented, and

    c. not invalidate the NRC staffs SE.

    In instances where it is unclear if a change is consistent with the approved methodology, Framatome may submit descriptions of a change to the NRC for confirmation that the change is within the scope of the approved methodology, as discussed in Section 3.9.3 of this SE.

13. ANP-3969P, Section 3.9.1, last paragraph states:

> Reference 11 [BAW-10240(P)(A)] incorporates M5 cladding properties into the S-RELAP5 based non-LOCA methodology. No restrictions or requirements are identified in the SER for the Reference 11 methodology relative to its application to S-RELAP5 non-LOCA analyses.

The above statement appears to be incorrect. Section 4.0 of the SER for Reference 11 (BAW-10240(P)(A)) lists four conditions given below. Explain how these conditions are met.

1. The corrosion limit, as predicted by the best-estimate model will remain below 100 microns for all locations of the fuel.

2. All of the conditions listed in the SEs for all FANP [Framatome, ANP} methodologies used for M5 fuel analysis will continue to be met, except that the use of M5 cladding in addition to Zircaloy-4 cladding is now approved.

3. All FANP methodologies will be used only within the range for which M5 data was acceptable and for which the verifications discussed in BAW-I0240(P) or Reference 2 was performed.

4. The burnup limit for this approval is 62 GWd/MTU.

14. Explain how the following limitations and conditions listed in ANP-10311P-A, Revision 1, “COBRA-FLX: A Core Thermal-Hydraulic Analysis Code,” are met:

1. The fuel rod model in COBRA-FLX and the rewetting model for post-CHF heat transfer will not be used for safety-related analysis and are specifically excluded from this review. Additional limitations are specified for the empirical correlations that will be used in licensing calculations. These empirical correlations are listed in Table 1-2 and Appendix A of the TR (Reference 1) and are summarized as the following:

   a) water properties (IAPWS-IF97 [International Association for the Properties of Water and Steam-Industrial Formulation-97])

   b) friction factor correlation constants
      i. Lehman friction factor (with or without Szablewski correction)
      ii. wall viscosity correction option

   c) two-phase friction multiplier - homogeneous model only

   d) bulk void correlation - Chexal-Lellouche (using the full curve fit routine or tables with interpolation)

e) subcooled void correlation - Saha-Zuber

f) subcooled boiling profile fit correlation - Zuber-Staub

g) nucleate boiling forced convection heat transfer correlation - Chen

h) post-departure from nucleate boiling (DNB) forced convection heat transfer correlation -Groeneveld 5. 7

i) single-phase convection heat transfer correlations

   i. Sieder-Tate for normal flow conditions
   ii. McAdams natural convection correlation for very low flow conditions

2. This review examined only the specific models and correlations requested by the applicant, as summarized in Section 2.0 of this SE. These are the only models and correlations that may be used in licensing calculations with the COBRA-FLX subchannel code. The fuel rod model in COBRA-FLX and the rewetting model for post-CHF heat transfer shall not be used for safety related analysis, and are specifically excluded from this review.

15. Make XN-NF-82-06(P)(A), “Qualification of Exxon Nuclear Fuel for Extended Burnup,” Revision 1, Supplement 5, available for review in the portal.

<u>SFNB Questions</u>

1. ANP-10342P-A “GAIA Fuel Assembly Mechanical Design,” TR section 4.0 “Limitations and Conditions” No. 5 states, “As part of the plant-specific LAR implementing GAIA, the licensee must demonstrate acceptable performance of GAIA under RIA [reactivity initiated accident] conditions, including fuel damage, coolable geometry, and radiological consequences, using approved methods.” This discussion should be based on current/updated guidance and analytical limits per SRP 4.2 appendix B. Provide the discussion or describe where it can be found in the LAR.

2. Discuss the hydraulic characterization comparison between Framatome GAIA fuel and co-resident Westinghouse fuel in the Callaway core including spacer loss coefficients, friction factors, and other pertinent hydraulic comparisons.

3. Discussion/presentation on Callaway GAIA fuel design rod ejection analysis as presented in ANP-4012P, “Callaway Rod Ejection Accident Analysis.” Please discuss whether this analysis is in line with the latest guidance as prescribed in Regulatory Guide (RG) 1.236 “Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Ejection Accidents,” dated June 2020 (ML20055F490). Discuss whether there is any deviation from the RG 1.236.

4. Section 2.1, “Technical Specification Changes,” of enclosure 1 to the letter dated October 12, 2022, states, in part, “Specifically, the proposed TS changes are driven

by the need to accommodate Framatome GAIA fuel assemblies starting with operating cycle 27, which affects TS 2.1.1 and TS 4.2.1. Westinghouse fuel will continue to constitute the vast majority of the fuel in the core."

Provide a chronology of the various fuel designs in the Callaway core design (GAIA and co-resident Westinghouse fuel design) starting from Callaway operating cycle 27, including the planned introduction of GAIA in batch sizes transition cycles and equilibrium cycle.

5. Fuel pellet diameter for GAIA fuel is about 7 percent larger than the co-resident Westinghouse fuel (0.3225 inch versus 0.3008 inch). Is there a corresponding change in the Uranium-235 (U-235) weight in GAIA fuel design. If yes, what is its impact on core design and energy requirements.

Attachment:
Requested Audit Materials List

REQUESTED AUDIT MATERIALS LIST

1. Section 2.2 of ANP-3947P discusses operational experience of Framatome GAIA fuel assemblies in Westinghouse 17x17 reactors. Also, section 2.2 states that post-irradiation examination included **[[**

   ]] The U.S. Nuclear Regulatory Commission (NRC) staff would like to review the detailed report on post-irradiation examination of the GAIA fuel with respect to the following characteristics.

[[

]]

2. Section 2.3 of ANP-3947P summarizes. mechanical compatibility of GAIA fuel with core internals, control components, rod cluster control assemblies, co-resident fuel assemblies, and in-core instrumentation at Callaway Plant, Unit No. 1 (Callaway). Please provide the calculations done for the fuel assembly mechanical and fuel rod thermal-mechanical analyses considered for the mechanical design evaluation:

Fuel Assembly Analyses
- Mechanical Compatibility
- Normal Operation Component Stress and Load Limits
- Faulted Component Stress and Load Limits
- Fretting Wear
- Fuel Assembly / Fuel Rod Growth
- Fuel Assembly Bow
- Lift-Off

Fuel Rod Analyses
- Cladding Fatigue
- Cladding Oxidation
- Internal Pin Pressure
- Internal Hydriding
- Cladding Creep Collapse
- Fuel Centerline Melt
- Transient Cladding Strain
- Cladding Stress and Buckling

Attachment

3. The faulted condition analysis that evaluates structural response of the fuel assembly to externally applied forces such as earthquakes and postulated pipe breaks, is based on the criteria established in the recently approved topical report ANP-10377P-A. Provide calculation details of your analysis for the following situations:

   a. Seismic evaluation for a mixed core with Framatome GAIA and co-resident Westinghouse fuel designs in the Callaway core.

   b. Section 2.4.3 of ANP-3947P describes the fuel assembly structural analysis for normal operating and faulted condition. Please provide calculation details for structural analysis for normal operating conditions for a Callaway core with GAIA fuel assemblies.

   c. Section 2.4.3 of ANP-3947P states "The faulted condition analysis evaluates the structural response of the fuel assembly to externally applied forces such as earthquakes and postulated pipe breaks in the reactor coolant system" based on NRC-approved methodology in ANP-10337P-A for GAIA fuel design. Provide calculation details of this analysis.

4. Section 3.3 of ANP-3947P discusses nuclear design for Callaway core based on the ARCADIA code system for transition cycles, and future operation of Framatome GAIA fuel at Callaway. Provide calculation analysis details for the following Callaway nuclear design aspects:

   a. Callaway transition core designs and the loading patterns developed based on design parameters such as energy requirements, peaking factor, and pin burnup limits. Shutdown margin, and other parameters related to core design.

5. Chapter 5 of ANP-3947P discusses thermal and hydraulic analyses that supports the transition to Framatome GAIA fuel at Callaway. Provide the below-listed calculation documents for NRC staff review during the upcoming regulatory audit.

   a. Mechanical/Thermal-hydraulic compatibility of the GAIA and co-resident Westinghouse fuel with different cladding materials and two different critical heat flux correlations for departure from nucleate boiling ratio calculations. Also discuss the details of the thermal-hydraulic characterization and thermal margin analysis for the mixed core at Callaway after the potential fuel transition to Framatome fuel.

   b. Calculation details of core pressure drops for the two fuel types at Callaway.

   c. Calculation details of how the rod bow is determined.

d. Calculations that show how the GAIA fuel design will not experience thermos-hydrodynamic instability during normal operation and anticipated operational occurrences (AOOs).

e. Calculation details of Callaway core departure from nucleate boiling performance during the transition to GAIA fuel design.

6. Calculation details of rod ejection analysis as described in ANP-4012P Revision 1

7. Provide calculation details of Callaway setpoints analysis as summarized in section 5.5.6 of ANP-3947P including over-temperature delta-temperature (OTΔT) verification, over-power delta-temperature (opδt) verification, and core safety limits verification.

8. Provide calculation details of how the GAIA rod bow penalties listed in table 5-1 of ANP-3947P, Revision 3 are obtained.

9. For guide tube heating, an “analysis was performed using the COBRAFLX code to demonstrate that boiling will not occur within the guide tubes of the Framatome GAIA fuel assemblies” (section 5.5.4 of ANP-3947P) Provide details of this calculation for guide tube heating.

10. Section 5.5.2 of ANP-3947P Revision 3, states, “Framatome has evaluated the Callaway Unit 1 reactor for its susceptibility to a wide range of potential thermo-hydrodynamic instabilities. This evaluation demonstrates that Callaway Unit 1 with GAIA fuel assemblies will not experience thermo-hydrodynamic instabilities during normal operation and AOO’s.” Provide the details of this calculation.

SUBJECT: CALLAWAY PLANT, UNIT NO. 1 – REGULATORY AUDIT PLAN AND SETUP OF ONLINE REFERENCE PORTAL FOR LICENSE AMENDMENT REQUEST AND EXEMPTION FOR FUEL TRANSITION TO FRAMATOME GAIA FUEL (EPID L-2022-LLA-0150 AND EPID L-2022-LLE-0030) MARCH 20, 2023

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