# ENCLOSURE 3

## RESPONSE TO AUDIT QUESTIONS
## (NON-PROPRIETARY VERSION)

The following pages provide the non-proprietary version of Callaway's response to the audit questions presented in References 1, 28 and 29. References are provided at the end of this enclosure.

The proprietary content was provided by Framatome, Inc. The request to withhold from public disclosure is addressed in the cover letter. This request is supported by the Affidavit included as Enclosure 2. Criteria (c) and (d) defined in the associated affidavit for this document apply to all bracketed material in this enclosure.

The proprietary information in the NRC Question is identified by text in bold font and enclosed within square brackets. **[ This is an example. ]** For ease of reading and presentation, the proprietary information in the response to the questions is identified by non-bolded text enclosed within square brackets. **[** This is an example. **]**

## Acronyms

| Acronym | Definition |
|---|---|
| AFW | Auxiliary Feedwater |
| AOR | Analysis of Record |
| AREA | ARCADIA® Rod Ejection Analysis |
| ASME | American Society of Mechanical Engineers |
| BOC | Beginning of Cycle |
| CE | Combustion Engineering |
| CFR | Code of Federal Regulations |
| CHF | Critical Heat Flux |
| COLR | Core Operating Limits Report |
| CR | Condition Report |
| CVCS | Chemical and Volume Control System |
| DNB | Departure from Nucleate Boiling |
| DNBR | Departure from Nucleate Boiling Ratio |
| ECCS | Emergency Core Cooling System |
| EM | Evaluation Model |
| FANP | Framatome ANP |
| FSAR | Final Safety Analysis Report |
| FSAR SP | Final Safety Analysis Report – Standard Plant |
| GWd/MTU | Gigawatt days per Metric Ton Uranium |
| HHSI | High Head Safety Injection |
| HZP | Hot Zero Power |
| ICRR | Inverse Count Rate Ratio |
| LAR | License Amendment Request |
| LBLOCA | Large Break LOCA |
| LCO | Limiting Condition for Operation |
| LFA | Lead Fuel Assembly |
| LHSI | Low Head Safety Injection |
| LTA | Lead Test Assembly |
| LOCA | Loss of Coolant Accident |

| Acronym | Definition |
|---|---|
| LOOP | Loss of Off-Site Power |
| M&E | Mass and Energy |
| MLO | Maximum Local Oxidation |
| MRR | Most Reactive Rod |
| MSSV | Main Steam Safety Valve |
| MTU | Metric Tons Uranium |
| NPSH | Net Positive Suction Head |
| NRC | Nuclear Regulatory Commission |
| OTDT | Overtemperature Delta Temperature |
| PSV | Pressurizer Safety Valve |
| PWR | Pressurized Water Reactor |
| RCCA | Rod Control Cluster Assembly |
| RCS | Reactor Coolant System |
| RG | Regulatory Guide |
| RIA | Reactivity Initiated Accident |
| RLBLOCA | Realistic Large Break Loss of Coolant Accident |
| RPS | Reactor Protection System |
| RTP | Rated Thermal Power |
| SBLOCA | Small Break LOCA |
| SE | Safety Evaluation |
| SER | Safety Evaluation Report |
| SG | Steam Generator |
| SGSV | Steam Generator Safety Valve |
| SGTP | Steam Generator Tube Plugging |
| SIAS | Safety Injection Actuation Signal |
| SPC | Siemens Power Corporation |
| SR | Surveillance Requirement |
| SRM | Swelling and Rupture Model |
| SRP | Standard Review Plan |
| TH | Thermal-Hydraulics |

| **Acronym** | **Definition** |
|---|---|
| TR | Topical Report |
| TR | FSAR Chapter 16 Technical Requirement |
| TS | Technical Specification(s) |
| VQP | Vendor Qualification Program |

**Ameren Missouri Response to NRC Audit Questions**

By application dated October 12, 2022 (Reference 33), as supplemented by letter dated December 1, 2022 (Reference 34), 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.

Subsequent to NRC acceptance of the LAR, the NRC performed an audit which was conducted April 3 through 20, 2023. The NRC questions/requests to be addressed during the audit were provided in a letter dated March 20, 2023 (Reference 1). In addition, the NRC requested via Emails dated April 17, 2023, (Reference 28) and April 24, 2023, (Reference 29) that responses be provided to additional questions discussed during a call held on April 20, 2023. At the conclusion of the audit activities, the NRC requested that Ameren Missouri's responses to selected questions from the audit be transmitted via a letter to the NRC. The responses (with their associated audit questions) are provided on the following pages of this enclosure.

Not all of the NRC audit questions required docketed responses. Those that did not are listed below.

SNSB Question 2
SNSB Question 3
SNSB Question 5

**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.

**Ameren Missouri Response:**

The changes proposed under this license amendment request and associated exemption regarding the use of Framatome GAIA fuel with M5 cladding are located within the restricted area associated with Callaway Plant, Unit 1 as defined in 10 CFR Part 20 and described in the Final Safety Analysis Report (FSAR) Standard Plant (SP) Chapter 12, "Radiation Protection." In addition, the changes proposed under this license amendment request and associated exemption regarding the use of Framatome GAIA fuel with M5 cladding involve revision to inspections and surveillance requirements associated with the nuclear fuel. This satisfies the entry conditions for application of exemption described in 10 CFR 51.22(c)(9).

As discussed in section 4.3, "No Significant Hazards Consideration Determination," of Enclosure 1 to ULNRC-06768, the evaluation of the changes described in the license amendment request against the criteria in 10 CFR 50.92(c) concluded that this license amendment request does not involve a significant hazards consideration. This satisfies the categorical exclusion criteria stated in 10 CFR 51.22(c)(9)(i).

The use of the Framatome GAIA fuel will be nearly indistinguishable from the use of the Westinghouse fuel currently used. No changes are proposed that alter unit operation, the operational characteristics of systems that interface with the reactor core containing the Framatome GAIA fuel, the reactor coolant system, the chemical and volume control system, nor are changes proposed that would alter the operational characteristics of the of the waste processing systems such that an increase in liquid or gaseous effluents would occur. Other than the physical presence of the eight GAIA assemblies, there are no changes to the design of any structure, system, or component. Given the design similarities with the co-resident fuel, the presence of the eight GAIA assemblies will not increase the available source term by a greater than negligible value. Therefore, there are no proposed changes that would cause a significant change in the types or significant increase in the amounts of any effluents that may be released offsite. This satisfies the categorical exclusion criteria stated in 10 CFR 51.22(c)(9)(ii).

No changes are proposed that alter unit operation and the operational characteristics of systems such that an increase in radiological source term would exist. The Framatome GAIA fuel was selected based on its demonstrated ability to operate free of fuel pin defects. For these reasons, no increase in occupational exposure to plant staff that operate, perform maintenance activities, and perform radiological monitoring activities are postulated. Similarly, the handling practices for irradiated GAIA fuel will not change from those associated with handling of irradiated Westinghouse fuel, and the corresponding occupational exposures during the handling of irradiated fuel is expected to be unchanged from that experienced during the handling of irradiated Westinghouse fuel. For these reasons, there is no significant increase in individual or cumulative occupational radiation exposure as a result of the changes described within this license amendment request. This satisfies the categorical exclusion criteria stated in 10 CFR 51.22(c)(9)(iii).

Based on the review above, Ameren Missouri has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

**SNSB (Nuclear Systems Performance Branch) 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)

**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)**

**Question 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.

**Ameren Missouri Response:**

The parameters and values listed in Table 3-1 of ANP-3943P are presented in the left-most column in the Table below. The TS limits, where applicable, are specified in the middle column, and the additional notes or comments, when warranted, are presented in the right-most column. Values chosen for analysis are either the TS values or conservative with respect to TS values. In some cases, the limit is provided in the Core Operating Limits Report (COLR), and in a few cases, the analysis value was derived from the safety analysis of record (AOR) as given in the FSAR or supporting calculation.

| Table 3-1 of ANP-3943P | Technical Specification | Comment |
|---|---|---|
| Reactor Power (MWt) 3636[1] | TS 1.1, DEFINITIONS: RATED THERMAL POWER (RTP) - RTP shall be a total reactor core heat | Measurement uncertainty of 2%. |

| Table 3-1 of ANP-3943P | Technical Specification | Comment |
|---|---|---|
| [1] Includes measurement uncertainty | transfer rate to the reactor coolant of 3565 MWt | |
| Axial Power Shape – Figure 3-4 | No explicit TS limit | Governed by COLR development process |
| Radial Peaking Factor ($F_{\Delta H}$) 1.65 (Includes measurement uncertainty) | 1.65 at RTP* | *Radial Peaking Factor (FDH): TS 3.2.2 – COLR specified - a power dependent limit with uncertainty included in the limit. Value is typical value |
| Maximum-Allowed Total Power Peaking Factor ($F_Q$) – 2.5[1]<br><br>[1] Includes uncertainties and k(z) set to 1.0. | 2.5* | *Heat Flux Hot Channel Factor/ Total Peaking Factor(F-Q): TS 3.2.1 – COLR specified - a power dependent limit with uncertainty included in the limit and including an axial dependent penalty factor K(z). Value is typical value. |
| Total RCS Flow Rate (gpm) 374,400 gpm | 382,630 gpm | TS 3.4.1.c (DNB limits) |
| Pressurizer Pressure (psia) 2249.3 psia | > 2195 psig* | *TS 3.4.1.a (DNB limits) is a COLR specified DNB value. |
| RCS Operating Temperature, $T_{avg}$- 590.1 °F | < 590.1 °F* | * TS. 3.4.1.b. (DNB limits) is a COLR specified DNB value. |
| SG Tube Plugging per SG (%) – 5% | No explicit TS limit | |
| SG Secondary Pressure (psia) – 998.9 psia | No explicit TS limit | |
| MFW Temperature – 446 °F | No explicit TS limit | |
| RPS Low Pressurizer Pressure for Reactor Trip (psia) - 1859.3 psia | ≥ 1874 psig | TS Table 3.3.1-1 item 8.a |
| RPS Low Pressurizer Trip delay (sec) – 2 sec | No explicit TS limit | FSAR TR 16.3.1.1;<br>Table 16.3-1 Function 9 |
| RPS Scram Delay (sec) 0 sec | No explicit TS limit | FSAR TR 16.3.1.1;<br>Table 16.3-1 Note (4) |

| Table 3-1 of ANP-3943P | Technical Specification | Comment |
|---|---|---|
| SIAS Low Pressurizer Pressure Activation Setpoint (psia) - 1714.3 psia | ≥ 1834 psig | TS Table 3.3.2-1 item 1.d |
| Accumulator Pressure (psia) - 616.3 psia | ≥ 602 psig and ≤ 648 psig. | SR 3.5.1.3 - Verify nitrogen cover pressure in each accumulator is ≥ 602 psig and ≤ 648 psig. |
| Accumulator Fluid Temperature (°F) – 120 °F | No explicit TS Limit | |
| Accumulator Water Volume per Accumulator ($ft^3$) – 850 $ft^3$ | ≥ 6061 gallons and ≤ 6655 gallons. | SR 3.5.1.2 Verify borated water volume in each accumulator is ≥ 6061 gallons and ≤ 6655 gallons. Average = 6358 gallons = 850 $ft^3$ |
| AFW Temperature (°F) – 120 °F | No explicit TS Limit | |
| Total AFW Flow Rate (gpm) – 400 gpm | No explicit TS limit | |
| AFW Initiation on Low-Low SG Narrow Range Level Setpoint (% Narrow Range Span) – 0% | ≥ 20.6%(s) of Narrow Range Instrument Span<br><br>≥ 16.6%(s) of Narrow Range Instrument Span | TS Table 3.3.2-1 function 6 - Values are for adverse containment and normal containment environment respectively |
| AFW Injection Delay (sec) – 60 sec | No explicit TS limit | FSAR TR 16.3.2.1; Table 16.3-2 |
| ECCS Pumped Injection Temperature (°F) – 100°F | ≥ 37°F and ≤ 100°F | SR 3.5.4.1 RWST borated water temperature (during the injection phase) |
| HHSI Injection Delay Time on SIAS (sec) – 29 sec | No explicit TS limit | FSAR TR 16.3.2.1; Table 16.3-2 |
| LHIS Injection Delay Time on SIAS (sec) – 44 sec | No explicit TS limit | FSAR TR 16.3.2.1; Table 16.3-2 |
| MSSV Lift Pressure and Accumulation Nominal + 3% Accumulation (nominal) | TS TABLE 3.7.1-2 MAIN STEAM SAFETY VALVE LIFT SETTINGS 1185, 1197, 1210, 1222, 1234 psig | Lift Settings are nominal (psig +3%/-1%) |

**Question 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."

**Ameren Missouri Response:**

The RODEX2-2A and S-RELAP5 codes used in the Callaway SBLOCA analysis are the same codes that are required for application of Reference 3, but the code versions are not. Framatome computer codes and the code maintenance process are controlled by Framatome software procedures which are compliant with ASME NQA-1 (version 2008/2009). Under these procedures, the new code version is verified to stay within the terms, conditions, and limitations of the approved methods for which the code supports. The code versions used in the Callaway SBLOCA analysis are verified under the Framatome software procedures, and therefore, are appropriate for use with the applied methods.

**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)**

**Question 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 [

]

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

**Ameren Missouri Response:**

The RLBLOCA analysis plant parameter range table is below.

The Technical Specification values are presented in the following table:

| RLBLOCA analysis parameter ranges | Technical Specification | Comment |
|---|---|---|
| | < 590.1 °F* | * TS. 3.4.1.b (DNB limits) is a COLR specified DNB value. |
| | LCO 3.4.1 - RCS total flow rate ≥ 382,630 gpm | |
| | > 2195 psig* | *T.S. 3.4.1.a (DNB limits) is a COLR specified DNB value. |
| | LCO 3.4.9 The pressurizer shall be OPERABLE with:<br>a. Pressurizer water level ≤ 92%; | |
| | None | |
| | LCO 3.6.5 Containment average air temperature shall be ≤ 120°F | |
| | SR 3.5.1.3 - Verify nitrogen cover pressure in each accumulator is<br>≥ 602 psig and ≤ 648 psig. | |
| | SR 3.5.1.2 - Verify borated water volume in each accumulator is ≥ 6061 gallons and ≤ 6655 gallons. | TS 3.5.1 has 6061 - 6655 gal. Average = 6358 gal = 850 $ft^3$ |

**Question 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.

**Ameren Missouri Response:**

Specific to the GAIA fuel in operating cycle 27, the four GAIA lead fuel assemblies have one cycle of burnup (from operating cycle 25) and the four VQP assemblies will be fresh fuel. The balance of the co-resident fuel (185 assemblies) is a traditional combination of fresh, once- and twice-burned fuel assemblies.

[

]

**Question 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.

**Ameren Missouri Response:**

The FSRR model evaluates the fuel rod rupture during LOCA and is discussed in the RLBLOCA EM (section 7.9.3.3 of Reference 5). The consequences of FSRR on the Callaway LBLOCA transient analysis is documented in the supporting analyses of

Reference 4. [ ] See RLBLOCA audit question response #5 below.

**Question 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.

**Ameren Missouri Response:**

Consistent with 10 CFR 50.46(b)(4) and (b)(5), two potential impacts of the addition of GAIA fuel were considered: debris blockage of fuel and boron precipitation. The first potential impact is addressed in the License Amendment Request (Reference 33) discussion regarding resolution of the issues documented in Generic Letter (GL) 2004-02 (e.g., GSI-191). That discussion may be found in Enclosure 1 of Reference 33, section 2.4, Page 7 of 24. The evaluation concluded that the presence of the eight GAIA fuel assemblies would not adversely affect the ability to maintain a coolable geometry and provide long term core cooling. Callaway's method for resolution of the issues documented in GL 2004-02 was approved by the NRC in Operating License amendment 228 (reference Safety Evaluation; ADAMS Accession No. ML22220A132).

Callaway FSAR SP Section 6.3A.1.2.1 and ULNRC-06526 Enclosure 2.1 (ADAMS Accession No. ML21090A184) state that in-vessel performance criteria for boron precipitation was addressed for Callaway in accordance with WCAP-17788 and the existing 13-hour time for hot leg switchover was confirmed to remain valid. Earlier bases for the hot leg recirculation switchover time are discussed in Westinghouse proprietary letters. The results of these evaluations are not sensitive to the very minor increase in fuel volume associated with the addition of 8 GAIA fuel assemblies (< 0.4 $ft^3$ per assembly).

**Question 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)** [

]

**Ameren Missouri Response:**

[

]

Table 4-8 of Reference 4 provides the fuel rod rupture ranges of parameters for the Callaway RLBLOCA analysis.

**Question 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.
>
> [
>
> ]

(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.

**Ameren Missouri Response:**

(a) The results within Reference 4 are only applicable to the Framatome fuel product, however mixed core configurations are considered. [

] The implementation of the mixed core modeling is provided in the supporting analyses of Reference 4.

(b) [

]

**Question 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: [

]

(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.

**Ameren Missouri Response:**

(a) The Cathcart-Pawel correlation is used as discussed in section 8.4.9 of the RLBLOCA EM (Reference 5). The analysis verifies that the MLO [ ] (fresh and once-burned UO2, and fresh and once-burned rods with gadolinium) is less than 13 percent. [ ]

(b) [ ]

**Question 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.

**Ameren Missouri Response:**

The Reference 4 fuel analysis which considers a full core of Framatome GAIA fuel assemblies and applies the mixed core modeling as described in the response to RLBLOCA audit question #6 [ ]

**Question 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.

**Ameren Missouri Response:**

The RLBLOCA analysis documented in Reference 4 makes use of a statistical realistic LOCA evaluation model (Reference 5) instead of conservative evaluation models specified by 10 CFR 50 Appendix K.

The Callaway AOR was performed by Westinghouse using evaluation models consistent with 10 CFR 50 Appendix K as described in FSAR SP Section 15.6.5. These models were performed with the 1981 Version of the Westinghouse ECCS Evaluation Model using BASH (Reference 29), including the changes in the methodology for execution of the model which are described in References 31 and 32 as well as various updates handled under the 10 CFR 50.46 reporting process.

As a result of the differences in the evaluation models, a direct comparison between values and comparison of conservatism is not an effective method of comparison.

**Question 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) AORs 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.

**Ameren Missouri Response:**

The following paragraphs provide the responses to questions (a) and (b) as they relate to each of the analysis areas (i) through (iv).

(i) and (ii) The vendor holding the M&E release AOR has performed an evaluation of the addition of eight GAIA fuel assemblies and concluded "This evaluation concludes that the licensing basis analyses related to the short and long term LOCA M&E releases and long term SLB M&E releases remain applicable to the Callaway VQP with up to eight Framatome GAIA fuel assemblies in the core. Westinghouse does not hold the containment analysis scope; however, since there is no impact on the M&E releases there is no expected impact on the downstream containment response analyses."

(iii) For the RLBLOCA analyses in Reference 4 the S-RELAP5/ICECON code interface allows ICECON to be run concurrently with S-RELAP5, [

] The three modeling factors described in section 3.1.3.4.1 of the RLBLOCA EM (Reference 5) provides assurance that the containment pressure applied in the RLBLOCA calculation is conservative. Additionally, section 3.2 of Reference 4 states the single-failure for this analysis, as defined in the EM, is the loss of one ECCS pumped injection train without the loss of containment spray. [

] The boundary conditions for the containment sprays are shown in Table 4-1 of Reference 4. For the demonstration case in Reference 4 the containment pressure response is shown in Figure 4-16.

(iii) and (iv) Regarding that available NPSH for containment spray pumps, and residual heat removal pumps (Callaway FSAR-SP Table 6.2.2-7), as noted in FSAR SP section 6.2.1.5, the containment backpressure is calculated, using the methods and assumptions described in WCAP-8339 Appendix A to ensure a conservative low value for containment pressure is used for ECCS performance. Core stored energy is not listed as an input in this Appendix. It was confirmed with Westinghouse that the containment response code, COCO (WCAP-8327) does not contain an input for core stored energy (CSE) or a decay heat calculation in the code. Therefore, the presence of the GAIA fuel will have no impact on the results.

Regarding the available NPSH from the containment sump to ECCS pumps as a function of containment pressure, FSAR SP Section 6.3.1.1 (GSI-191 discussion) states the following:

> Containment accident pressure of 1.7 psi is credited for available NPSH during the LBLOCA phase when containment temperature is above 212°F to assure no flashing to steam occurs in the debris bed (approximately 10% of available containment pressure). In general, for a sump temperature of 212°F and higher, the NPSH available calculation assumes that the containment pressure is equal to the vapor pressure. For sump temperatures lower than 212°F, the containment pressure is assumed to be equal to atmospheric pressure of 14.7 psia, as if there is loss of containment. Overpressure credit is not needed to meet the NPSH, air release, or strainer buckling strainer performance criteria.

The 1.7 psi of credit in the GSI-191 calculations is a very conservative credit. The "approximately 10% of available containment pressure" statement is based on the containment response analysis of record. Since the containment analysis M&Es are not impacted by eight GAIA assemblies, there is no impact to the validity of this ~10% (1.7psi) credit in the GSI-191 analysis. The overpressure credit is not specifically related to NPSH; rather, it is credited to prevent boiling within the debris bed.

Aside from GSI-191, the FSAR SP Section 6.3.2.2 NPSH discussion documents no credit for containment overpressure:

> Available and required net positive suction head (NPSH) for ECCS pumps are shown in Table 6.3-1. Table 6.2.2-7 provides the assumptions and results of the NPSH analyses for the containment spray and RHR pumps. The safety intent of Regulatory Guide 1.1 is met by the design of the ECCS so that adequate NPSH is provided to system pumps. In addition to considering the static head and suction line pressure drop, the calculation of available NPSH in the recirculation mode assumes that the vapor pressure of the liquid in the sump is equal to the containment ambient pressure. This ensures that the actual available NPSH is always greater than the calculated NPSH. To ensure that the required NPSH is available during the recirculation phase of ECCS operation, restriction orifices are provided in the four discharge lines into the RCS cold legs and in the two discharge lines into the RCS hot legs.

**<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>**

**Question 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.

**Ameren Missouri Response:**

The GAIA fuel decay heat is bounded by the AOR. Through confirmation with the vendor that performed the AOR, the decay heat models used are the "1971 model, ANS-1979 model, and the ANS-1979+2σ model." This decay heat model bounds all variations of core designs, with exceptions including competitor cores, transition cores containing competitor fuel, and fuel containing mixed oxide fuel. Since the operating cycle 25 and 27 cores contain a limited number of non-limiting competitor assemblies (not full regions as would be expected in a transition core), it is considered that the decay heat model bounds the operating cycle 25 and 27 cores.

Core decay heat magnitude is confirmed for each reload design. As part of the existing reload design process, the individual parameters influencing this value were evaluated and considered met in the reload safety analyses and design confirmations for operating cycles 25 and 27. Future core designs with a larger number of GAIA assemblies (full region) would require an evaluation for impact on decay heat.

**Question 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?

**Ameren Missouri Response:**

No. As discussed in FSAR SP Section 15.2.7, the pressurizer does not become water-solid for this event. There is no water relieved from the pressurizer via either the safety valves or power operated relief valves.

As described in FSAR SP Sections 15.2.6.1 and 15.2.7.1, the safety-related Auxiliary Feedwater System (AFW) actuates and delivers water to the SGs. This ensures that a heat sink is available to remove heat from the reactor coolant system via the SGs.

The paragraph cited in the question resides in FSAR SP Section 15.2.7.1 and simply presents the hypothetical condition that would result if an alternative supply of feedwater to the SGs were not available beyond that provided by the normal feedwater system. The safety-related AFW system with its design redundancy ensures that this condition does not occur.

Key parameters for this event include:

- Initial power
- Initial vessel average temperature
- Initial pressurizer pressure.
- Pressurizer pressure control (PORVs and sprays)
- PSV setpoint
- Injection flow rate
- Operator action time

No aspect of Framatome fuel significantly affects the key parameters for this analysis, and therefore no reanalysis for pressurizer overfill is required with Framatome fuel.

**Question 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 Section 15.2.8 event could result in RCS cooldown.

**Ameren Missouri Response:**

This event can be considered a heat-up event, a cool-down event, or a combination of both. There can be an initial, short heat-up transient when the feedwater flow stops. This phase is terminated by reactor trip. Following reactor trip, the primary and secondary systems begin to cool down as a result of the heat removal from the affected SG via excessive discharge through the feedwater line break. The cool-down portion of the transient is terminated by dryout of the affected steam generator, which dramatically reduces the heat removal from the primary system.

**Question 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.

**Ameren Missouri Response:**

This flow penalty map has been applied in both the Calvert Cliffs (Reference 7, p. 90) and St. Lucie Unit 2 (Reference 14, p. 180/272) fuel transitions to HTP fuel. Note that these examples are for CE units with HTP fuel. The plant type and fuel design will have a negligible effect on the choice of inlet flow asymmetry used in the subchannel analysis.

**Question 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?

**Ameren Missouri Response:**

Revision 1 of Reference 8 is only relevant for the boron dilution event. The SER for Revision 1 is similarly limited in scope to boron dilution and therefore does not contain these requirements. These requirements are only contained in the SER for Revision 0 of Reference 8.

The language used in Reference 6 is consistent with similar previous applications of the methodology (e.g., Reference 14, section 2.7).)

**Question 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.

**Ameren Missouri Response:**

Specific differences between individual parameter values in FSAR Table 15.0-2 and Reference 6 Table 3-1 are due to the application of vendor-specific methods (e.g., application of an uncertainty to initial conditions vs. design limits). The Framatome analyses support the Technical Specification (TS) Limiting Conditions for Operation (LCO), and the LCOs support plant operating conditions. The difference in input parameters is consistent with the approved Framatome methodology. Approval of the method via its SER provides justification for the selection of inputs and the application of uncertainties.

**Question 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).

**Ameren Missouri Response:**

These conditions are addressed as follows:

1. CVCS dilution flow is based on conservative values provided in the FSAR SP Section 15.4.

2. [

   ]

3. All neutronics calculations are performed utilizing a full fuel core. This only applies to Mode 6 analysis, which was not analyzed for the VQP.

4. RCS volume is based on conservative values provided in the FSAR SP Section 15.4.

5. Mode 6 was not analyzed for the VQP.

6. In Mode 1, the results of the boron dilution event are bounded by the range of reactivity insertion rates considered for the uncontrolled bank withdrawal at power analysis.

7. [

   ]

8. This only applies to Mode 1 and is bounded by uncontrolled bank withdrawal at power analysis.

**Question 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.

**Ameren Missouri Response:**

These conditions are addressed as follows:

1. For the statistical setpoint analyses, no additional variables to those explicitly mentioned in Reference 9 are treated statistically. The code packages used to verify the setpoints are hardwired to support the statistical treatment of the variables described in the topical report, and the analyst does not have flexibility in changing these. [
]

2. This restriction only applies to the OTDT verification analysis. Relevant inputs should correspond to the steam generator plugging level being analyzed to meet this condition. Due to lack of pressure vs. power data for 5% SGTP (the VQP cycle plugging level), the OTDT analysis used a conservative value of 10%.

**Question 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:

Conditions

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.

Limitations

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.

**Ameren Missouri Response:**

From Reference 6 section 3.9.1, the conditions were addressed as follows:

1. The subchannel TH code runs were verified to show subcooled coolant conditions at the first axial node.

2. The DNB calculations utilizing the ORFEO-NMGRID correlation were verified to remain far removed from the [ ] subregion.

3. The DNB analyses with low margin to the design limits were verified to have equilibrium qualities greater than -0.1.

From Reference 6 section 3.9.1, the limitations were addressed as follows:

1. The ORFEO-GAIA correlation was validated for use within XCOBRA-IIIC, and a design limit was calculated. The modeling options used for the DNB calculations were consistent with the modeling options used for the validation within XCOBRA-IIIC. The DNB calculations were confirmed to be within the application domain for use with XCOBRA-IIIC.

2. The ORFEO-NMGRID correlation was validated for use within XCOBRA-IIIC and a design limit was calculated. The modeling options used for the DNB calculations were consistent with the modeling options used for the validation within XCOBRA-IIIC. The DNB calculations were confirmed to be within the application domain for use with XCOBRA-IIIC.

**Question 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].

**Ameren Missouri Response:**

Callaway does not utilize mixed oxide (MOX) fuel and is not licensed to do so. This satisfies the limitation and condition.

**Question 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.

**Ameren Missouri Response:**

From Reference 6 section 3.9.1, these limitations and conditions are addressed as follows:

1. XCOBRA-IIIC was not utilized for LOCA/ECCS calculations. Additionally, regardless of flow reversal, (i.e., locked rotor transient), the snapshot boundary conditions (from S-RELAP5) account for this, and as such the XCOBRA-IIIC code was not used to analyze flow reversals or recirculation.

2. The XNB CHF correlation was not used; therefore, this condition is not applicable.

3. The XNB CHF correlation was not used; therefore, this condition is not applicable.

4. XCOBRA-IIIC was utilized only using the “snapshot” mode.

**Question 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.

**Ameren Missouri Response:**

The ARCADIA L&Cs are addressed for the Callaway plant as follows:

1. The Callaway plant adds no new materials or geometries that are not already present in the topical report (Reference 12). Therefore, no new benchmarks are required per L&C 1 of the topical report. However, benchmarks that included more than three cycles were performed per Framatome standard practice.

2. As in the response to question 1, the Callaway plant falls within the benchmarks provided in the specified topical report and the uncertainty analysis remains applicable. However, benchmarks that include more than three cycles were performed, and it was deemed prudent to validate the uncertainty analysis because the Callaway monitoring system uses a different number of axial nodes than those used in the topical report. It was confirmed that the peaking uncertainties for Callaway remain bounded by the values in the topical report.

3. N/A.

4. The multi-cycle benchmarking and comparisons to operating data performed for Callaway cover the requirement of this L&C. These benchmarks provide confirmation that the COBRA-FLX code used in ARTEMIS is functioning as expected and all generated data remain consistent with the approved topical report.

5. ARTEMIS is not used as part of an evaluation model in the Callaway VQP analysis apart from the RCCA ejection analysis. Therefore, this L&C is only applicable to the SRP 15.4.8 event analysis. Use of ARTEMIS as the evaluation model in the RCCA ejection analysis is covered in the AREA topical report (Reference 13).

6. For each new release of the ARCADIA code system a review of the changes in the codes is performed. This review bins the changes in three categories: 1) changes allowed by the topical report, 2) changes that require discussion with the NRC to determine if they need additional review, and 3) changes that cannot be used until a supplement implementing the change has been approved by the NRC. This review is documented and provided to all users of the ARCADIA codes. The review documents clearly identifies which features and models are not allowed for licensing analyses.

**Question 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-l0240(P) or Reference 2 was performed.

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

**Ameren Missouri Response:**

These conditions are addressed as follows:

1. S-RELAP5 does not calculate corrosion; therefore, the limit is unaffected by the use of S-RELAP5.

2. Conditions of other methods or topical reports are checked in their respective sections.

3. The analyses presented in Reference 6 are within the range of applicability for $M5_{Framatome}$ as presented in Reference 15.

4. The burnup limit remains at 62 GWd/MTU (Reference 17, p. 48/98).

**Question 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.

**Ameren Missouri Response:**

From Reference 6 section 3.9.1, these limitations and conditions are addressed as follows:

1. COBRA-FLX model development guidance prescribes the use of these approved models. Additionally, NRC approved models are set by default and are the only allowed options. The code will terminate with an error message if the user attempts to over-ride them to an unapproved model.

2. No post-CHF calculations utilizing the rewetting model or the COBRA-FLX internal fuel rod model were used.

**Question 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.

**Ameren Missouri Response:**

XN-NF-82-06(P)(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup," Revision 1, Supplement 5, was made available for review in the electronic reading room (i.e., portal).

In response to the discussion during the audit regarding limitations and restrictions contained within the safety evaluation report (SER) associated with the topical report, as stated in Reference 6, section 3.9.1, no restrictions, limitations, and/or conditions are identified in the SER for Reference 23 relative to DNB propagation.

**SFNB (Nuclear Methods and Fuel Analysis Branch) Questions**

**Question 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.

**Ameren Missouri Response:**

The Reference 19 TR L&C has been addressed with the spectrum of RCCA Ejection Accidents (REAs) in Section 6.1 of Reference 17 by performing an explicit RIA analysis using the AREA methodology (Reference 13). The event analysis was performed in Reference 20 and was based on the Vendor Qualification Program (VQP) representative cycle design. The analysis is applicable to transition cycles (containing co-resident fuel with the GAIA fuel) and cycle designs containing a full core of GAIA provided the conditions of the cycle design are bounded by the requirement of the Callaway REA analysis described in Reference 20. This analysis report was submitted to the NRC in Ameren Missouri letter ULNRC-06783 dated November 16, 2022 (Reference 34).

The Reference 20 analysis is compliant with the criteria defined in Reference 21. Additional information is provided in the Question 3 Response.

**Question 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.

**Ameren Missouri Response:**

Both vendors have assessed the impact of hydraulic differences between the two fuel designs on the fuel mechanical and thermal hydraulic performance of their fuel. The results are summarized below:

Westinghouse

Westinghouse has evaluated the impact of the GAIA assemblies on the fuel mechanical design for the resident Westinghouse 17x17 Vantage+ fuel design. The evaluations address a number of different areas including fuel assembly lift forces and top nozzle holddown forces, Seismic/LOCA analyses, fuel handling, and the potential flow induced vibration and grid-to rod fretting wear concern. Based on the evaluations and analysis it is concluded that the resident Westinghouse 17x17 Vantage+ fuel design for fuel mechanical design considerations, such as top nozzle holddown forces, Seismic/LOCA analyses, fuel handling, the potential for flow induced vibration and grid-to rod fretting wear, will not be adversely affected by the presence of eight Framatome GAIA fuel assemblies.

Regarding thermal hydraulic analysis, DNB penalties for Cycle 27 and any cycle containing GAIA fuel are applied in accordance with WCAP-11837-P-A which accounts for the flow impact of hydraulic differences (FLCs/Friction factors) between fuel designs in transition cores.

Framatome

The differences in hydraulic characteristics between the resident and GAIA fuel assembly designs have been evaluated for impact on mechanical and thermal-hydraulic design criteria applicable to GAIA fuel. Regarding mechanical specified acceptable fuel design limits (SAFDLs), the pressure drop profile between the two assembly types has been calculated and cross-flow velocities affecting the Framatome GAIA fuel assemblies were analyzed (using COBRA-FLX) to assure satisfactory performance during transition. Several transition cores were assessed and the bounding configuration (highest cross flow velocity) was identified. This bounding core configuration was considered to cover all mixed-core configurations associated with the transition. Cross-flow velocities were provided as an input to a mechanical assessment. The SAFDLs for the fuel rod and fuel assembly for the LAR representative cycles are presented in ANP-3947P (Attachment 9 to Enclosure 1 of Reference 33) and demonstrate that the fuel design is acceptable to ensure mechanical SAFDL compliance. The specific reload designs are assessed to show continued SAFDL compliance. SAFDLs potentially affected by hydraulic differences between the fuel designs include fretting wear/cross-flow velocities and liftoff/top nozzle hold down forces.

Thermal hydraulic analyses show that Framatome GAIA results in an increase in the RCS loop flow due to the lower pressure drop in the GAIA fuel assembly. Therefore, the change in the Reactor Coolant System (RCS) loop flow does not impact the Technical Specification minimum loop flow rate. The driving force for bypass flow decreases and the total bypass flow fraction decreases transitioning from the resident fuel to Framatome GAIA fuel assemblies such that the bounding bypass flow value of 8.6% remains bounding.

The GAIA fuel assembly is associated with less overall flow resistance than the resident fuel. This results in flow transferring from the co-resident fuel to the GAIA fuel assembly, which improves GAIA DNB performance relative to a full GAIA core configuration. The conclusion is that a full core of GAIA fuel is limiting for DNB analysis relative to mixed core configurations at Callaway. The VQP DNB results presented in Enclosure 1 to ULNRC-06768 (Reference 33) are based on a full core of GAIA fuel and demonstrate that applicable limits are met. The VQP setpoint analyses confirm that neither DNB nor hot leg saturation are predicted to occur within the operating space permitted by the existing Core Safety Limits for the GAIA fuel assembly design and that the OPΔT [overpower delta temperature] trip protects the Framatome GAIA fuel assembly from Fuel Centerline Melt (FCM). Margin to the DNB acceptance criteria for event specific analysis is confirmed and documented in ANP-3969P (Attachment 12 to Enclosure 1 to ULNRC-06768 (Reference 33)). Note that a DNB assessment is performed for each specific reload design to confirm that the VQP analysis remains bounding.

**Question 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.

**Ameren Missouri Response:**

Reference 20 provides the Callaway REA analysis using Framatome's ARCADIA Rod Ejection Accident (AREA) methodology (Reference 13). This methodology is compliant with the criteria defined in Reference 21. The relevant acceptance criteria from Reference 21 are provided below:

Fuel Rod Cladding Failure Thresholds –

- High Temperature Cladding Failure Threshold – For prompt critical scenarios, the failure threshold provided in Reference 21, Figure 1, is used for the first 3 seconds. For non-prompt critical scenarios and **[** **]** are used as the threshold.
- Pellet Clad Mechanical Interaction (PCMI) Cladding Failure Threshold – Pellet Clad Mechanical Interaction failure thresholds (Δcal/g) from Reference 21, Figures 2 and 3.
- Molten Fuel Cladding Failure Threshold – Fuel cladding failure is presumed if predicted fuel temperature anywhere in the pellet exceeds incipient fuel melting conditions

Allowable Limits on Radiological Consequences – The Reference 13 methodology is used to determine the number of fuel rod failures but does not address the radiological consequences of the Rod Ejection Accident. RG 1.183 (Reference 24) and RG 1.195 (Reference 25) contain the accident dose radiological consequences criteria for control rod ejection accidents.

Allowable Limits on Reactor Coolant System Pressure – The reactor coolant system pressure was not evaluated using the Reference 13 methodology. No aspect of the Framatome fuel affects the severity of the rod ejection overpressure analysis, thus, no reanalysis is required.

Allowable Limits on Damage Core Coolability -

- Peak radial average fuel enthalpy must remain below 230 cal/g (Reference 21, Section 6.0).
- A limited amount of fuel melting is acceptable provided it is restricted to the fuel centerline region and is less than 10 percent of the pellet volume. The peak fuel temperature in the outer 90 percent of the pellet's volume must remain below incipient fuel melting conditions (Reference 21, Section 6.0).

The results of the Callaway AREA analysis (Reference 20) show that there is ample margin to limits for fuel temperature, fuel rim temperature, MDNBR, enthalpy, and enthalpy rise, which means there are no fuel failures associated with this event and, therefore, no dose consequences.

**Question 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.

**Ameren Missouri Response:**

Callaway intends to utilize eight (8) GAIA assemblies in operating cycle 27. These eight assemblies are comprised of four (4) once-burned assemblies which were used in operating cycle 25 (e.g., lead fuel assemblies) and four (4) feed assemblies referred to as the VQP assemblies. Future cycle designs are not developed at this time. While no core loading plans have been drafted for operating cycles beyond 27, the four (4) VQP assemblies are expected to be used in Cycle 28.

Westinghouse fuel continues to be the co-resident fuel.

Due to a change in the date for awarding the future fuel contract, no batch loading of GAIA fuel could occur before operating cycle 30 or 31. If Framatome is awarded the fuel contract, another LAR would be submitted to support fuel transition and adoption of the Framatome power distribution controls and COLR methods to support the power distribution TSs.

**Question 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.

**Ameren Missouri Response:**

It is correct that the GAIA fuel pellet is larger than the coresident fuel pellets, but this change is explicitly accounted for in the modeling of the GAIA assemblies and the establishment of fuel enrichment and burnable absorber requirements used to determine cycle length and ensuring peaking margins are maintained. Also, the Framatome VQP analysis models a full core of GAIA fuel accounting for this design change in the licensing analysis.

The core reload design process establishes a combination of enrichment and burnable neutron absorber content to ensure that the core contains the needed energy content for the operating cycle while meeting the TS and safety analysis limits. This is true for both Westinghouse and Framatome processes.

**Additional Questions provided by the NRC include the following. These questions were included in Emails dated April 17, 2023 (Reference 28) and April 24 (Reference 29).**

**Question 1 -** ANP-3947P, section 5.5.1.1, discusses the pressure drop associated with the GAIA fuel. Discuss how the pressure drop is determined for the GAIA fuel assemblies under the various described configurations.

**Ameren Missouri Response:**

The pressure drop calculations for the Callaway VQP and LFA programs were determined as follows: **[**

**]**

The two pressure values presented in ANP-3947P section 5.5.1.1 text, respectively, are the full-core GAIA pressure drop and the delta pressure drop between full-core GAIA and full-core resident fuel.

**Question 2 -** ANP-3947P, Section 5.5.3 lists the GAIA rod bow penalties for DNBR and LHGR. Provide a short summary of the procedure used to calculate these penalties for the GAIA fuel.

**Ameren Missouri Response:**

The penalties for GAIA fuel are calculated using the RODBOW code, utilizing calculational methodology from Reference 26, and the gap closure model from Reference 27. More detail is provided below:

**[**

**]**

**Question 3 -** ANP-10341PA, "The ORFEO-GAIA and ORFEO-NMGRID Critical Heat Flux Correlations," includes several limitations and conditions in section 4 of the associated Safety Evaluation. Discuss how the conditions and limitations are satisfied for the proposed Callaway application.

**Ameren Missouri Response:**

The Limitations and Conditions for ANP-10341 (Reference 10) are addressed in section 3.9.1 (pages 8 and 9) of Reference 6.

**Question 4 -** Table 28 of FS1-0042400, "Mechanical compatibility of GAIA with co-resident fuel (VANTAGE)," lists spacer and mixing grid centerline elevations for GAIA and resident fuel in the Callaway core. Discuss the impact on mechanical compatibility between the two fuel assemblies because of the non-alignment of spacer and mixing grids of two fuels as per Table 28**.**

**Ameren Missouri Response:**

The table in the cited reference presents a comparison of the nominal, BOL (cold) centerline elevations for the GAIA and resident (co-resident) fuel. Differences in centerline elevations between the two fuel assemblies were explicitly evaluated for mechanical (seismic) and thermal-hydraulic performance and found acceptable.

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3. Framatome Inc. Topical Report EMF-2328(P)(A), Revision 0, Supplement 1(P)(A), Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," March 2012.

4. Framatome Inc. Licensing Report ANP-3944(P), Revision 1, "Callaway Realistic Large Break LOCA Analysis with GAIA Fuel Design," October 2022.

5. Framatome Inc. Topical Report EMF-2103(P)(A), Revision 3, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," June 2016.

6. Framatome Inc. Licensing Report ANP-3969(P), Revision 2, "Callaway Non-LOCA Summary Report," October 2022.

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13. Framatome Inc. Topical Report ANP-10338(P)(A), Revision 0, "AREA™ - ARCADIA® Rod Ejection Accident," December 2017.

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20. Framatome Inc. Technical Report ANP-4012(P), Revision 1, "Callaway Rod Ejection Accident Analysis," November 2022.

21. Regulatory Guide RG 1.236, "Pressurized-Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents," ADAMS Accession No. ML20055F490, July 2020.

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28. NRC Letter from Mahesh Chawla (NRC) to Thomas Elwood (Ameren), "Regulatory Audit Questions - Callaway LAR and Exemption for Fuel Transition to Framatome GAIA Fuel - EPIDs - L-2022-LLA-0150 and L-2022-LLE-0030," ADAMS Accession No. ML23107A181, April 17, 2023.

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