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

October 5, 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 – ISSUANCE OF AMENDMENT NO. 235 TO REVISE TECHNICAL SPECIFICATIONS TO USE FRAMATOME GAIA FUEL (EPID L-2022-LLA-0150)

Dear Mr. Diya:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 235 to Renewed Facility Operating License No. NPF-30 for the Callaway Plant, Unit No. 1 (Callaway). The amendment consists of changes to the technical specifications (TSs) in response to your application dated October 12, 2022, as supplemented by letters dated December 1, 2022, May 9, 2023, June 21, 2023, and August 3, 2023.

The proposed amendment would revise the Callaway TSs to allow loading of a limited number of Framatome, Inc GAIA fuel with M5® as a fuel cladding material in operating cycle 27 to obtain incore performance data and acquire operational experience associated with the GAIA fuel design. Since the GAIA fuel uses M5® fuel rod cladding, the licensee included a Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.46 and 10 CFR Part 50, Appendix K exemption request as a part of the license amendment request. The staff reviewed the exemption request in a separate safety evaluation (SE), dated October 5, 2023.

The NRC staff has determined that the related SE contains proprietary information pursuant to 10 CFR 2.390, "Public inspections, exemptions, request for withholding." The proprietary information is indicated by bold text enclosed with **[[double brackets]]**. The proprietary version of the SE is provided as enclosure 2. Accordingly, the NRC staff has also prepared a non-proprietary version of the SE, which is provided as enclosure 3.

Enclosure 2 to this letter contains proprietary information. When separated from Enclosure 2, this document is DECONTROLLED.

A copy of the related SE is also enclosed. Notice of Issuance will be included in the Commission's monthly *Federal Register* notice.

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

Enclosures:
1. Amendment No. 235 to NPF-30
2. Safety Evaluation (Proprietary)
3. Safety Evaluation (Non-Proprietary)

cc: Listserv without enclosure 2

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

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT NO. 1

DOCKET NO. 50-483

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 235
License No. NPF-30

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Union Electric Company (UE, the licensee), dated October 12, 2022, as supplemented by letters dated December 1, 2022, May 9, 2023, June 21, 2023, and August 3, 2023, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-30 is hereby amended to read as follows:

    (2) Technical Specifications and Environmental Protection Plan*

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 235 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of its date of issuance, and shall be implemented prior to commencement of operating cycle 27.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Jennifer L. Dixon-Herrity, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. NPF-30 and
the Technical Specifications

Date of Issuance: October 5, 2023

ATTACHMENT TO LICENSE AMENDMENT NO. 235

CALLAWAY PLANT, UNIT NO. 1

RENEWED FACILITY OPERATING LICENSE NO. NPF-30

DOCKET NO. 50-483

Replace the following pages of the Renewed Facility Operating License No. NPF-30 and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Renewed Facility Operating License

REMOVE	INSERT
-3-	-3-

Technical Specifications

REMOVE	INSERT
2.0-1	2.0-1
4.0-1	4.0-1

(3) UE, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) UE, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source of special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) UE, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

UE is authorized to operate the facility at reactor core power levels not in excess of 3565 megawatts thermal (100% power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan*

The Technical Specifications contained in Appendix A, as revised through Amendment No. 235 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Environmental Qualification (Section 3.11, SSER #3)**

Deleted per Amendment No. 169.

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* Amendments 133, 134, & 135 were effective as of April 30, 2000 however these amendments were implemented on April 1, 2000.

** The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:

2.1.1.1 For Westinghouse fuel, the departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.17 for the WRB-2 DNB correlation.

2.1.1.2 For Westinghouse fuel, the peak fuel centerline temperature shall be maintained < 5080ºF, decreasing by 58ºF per 10,000 MWd/MTU of burnup.

2.1.1.3 For Framatome GAIA fuel, the DNBR shall be maintained ≥ 1.12 for the ORFEO-GAIA DNB correlation.

2.1.1.4 For Framatome GAIA fuel, the peak fuel centerline temperature shall be maintained < 4901°F, decreasing linearly by 13.7°F per 10,000 MWd/MTU of burnup.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

4.0 DESIGN FEATURES

4.1 Site Location

The Callaway Plant site consists of approximately 2,767 acres of rural land 10 miles southeast of the city of Fulton in Callaway County, Missouri, and 80 miles west of the St. Louis metropolitan area.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of zircaloy, ZIRLO™ or M5® clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitution of fuel rods by zirconium alloy or stainless steel filler rods may be used in accordance with approved applications of fuel rod configurations. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Rod Assemblies

The reactor core shall contain 53 control rod assemblies. The control rod material shall be silver indium cadmium, hafnium metal, or a mixture of both types, as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum nominal U-235 enrichment of 5.0 weight percent;

(continued)

ENCLOSURE 3

(NON-PROPRIETARY)

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 235 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-30

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]].

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 235 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-30

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT NO. 1

DOCKET NO. 50-483

1.0 INTRODUCTION

By letter dated October 12, 2022 (Reference 1), as supplemented by letters dated December 1, 2022, May 9, 2023, June 21, 2023, and August 3, 2023 (References 2, 3, 4, and 5), Union Electric Company, doing business as (dba) Ameren Missouri (the licensee), pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.90, "Application for amendment of license, construction permit, or early site permit," submitted a license amendment request (LAR) to the U.S. Nuclear Regulatory Commission (NRC, the Commission) for Callaway Plant, Unit No. 1 (Callaway). The proposed amendment would revise the technical specifications (TSs) to allow loading of a limited number of Framatome Inc (Framatome) GAIA fuel with M5® as a fuel cladding material in operating cycle 27 to obtain incore performance data and acquire operational experience associated with the GAIA fuel design. Since the GAIA fuel uses M5® fuel rod cladding, the licensee has included a 10 CFR 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 as enclosure 2 to the LAR. The staff reviewed the exemption request in a separate safety evaluation (SE) dated October 5, 2023 (Reference 6).

The licensee is implementing a plan that provides the option of transitioning from the use of fuel manufactured by Westinghouse Electric Company (Westinghouse), as currently used in the Callaway reactor core, to the use of fuel manufactured by Framatome. The GAIA fuel assemblies manufactured by Framatome were placed in Callaway's core in non-limiting locations during operating cycle 25. The licensee contracted Framatome to establish a Vendor Qualification Program (VQP) for the use of GAIA fuel at Callaway and Framatome's evaluation methodologies for application at Callaway.

In support of this LAR, the licensee intends to license the GAIA fuel design and its evaluation methodologies. Framatome developed a series of technical reports and evaluations that demonstrate compliance with Callaway's licensing basis and regulatory acceptance criteria. The technical reports are contained in the LAR as attachments 9 through 12 (attachments 5 through 8 are the publicly available versions) in support of a full transition to Framatome GAIA fuel including intermediate batch load quantities of fuel.

This SE is only intended for the loading of eight Framatome GAIA fuel assemblies with M5® cladding to Callaway.

The supplemental letters dated May 9, 2023, June 21, 2023, and August 3, 2023, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* (FR) on March 7, 2023 (88 FR 14184).

2.0 REGULATORY EVALUATION

2.1 Proposed TS Changes

The licensee's proposed TS changes are as follows:

TS 2.1.1, "SLs" [Safety Limits]

Current TSs 2.1.1.1 and 2.1.1.2 states:

> In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR [Core Operating Limits Report]; and the following SLs shall not be exceeded:
>
> 2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.17 for the WRB-2 DNB [departure from nucleate boiling] correlation.
>
> 2.1.1.2: The peak fuel centerline temperature shall be maintained < 5080°F degrees Fahrenheit], decreasing by 58°F per 10,000 MWd/MTU [megawatt day per metric ton of uranium] of burnup.

Proposed TSs 2.1.1.1 and 2.1.1.2 with the addition of new TSs 2.1.1.3 and 2.1.1.4 would state:

> In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR, and the following SLs shall not be exceeded:
>
> 2.1.1.1 For Westinghouse fuel, the departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.17 for the WRB-2 DNB correlation.
>
> 2.1.1.2 For Westinghouse fuel, the peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWd/MTU of burnup.

2.1.1.3 For Framatome GAIA fuel, the DNBR shall be maintained ≥ 1.12 for the ORFEO-GAIA DNB correlation.

2.1.1.4 For Framatome GAIA fuel, the peak fuel centerline temperature shall be maintained < 4901°F, decreasing linearly by 13.7°F per 10,000 MWd/MTU of burnup.

Current TS 4.2.1, “Fuel Assemblies,” states, in part:

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitution of fuel rods by zirconium alloy or stainless-steel filler rods….

Proposed TS 4.2.1 would state, in part:

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of zircaloy, ZIRLO™ or M5® clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitution of fuel rods by zirconium alloy or stainless steel filler rods….

2.2 Applicable Regulations and Guidance

The NRC staff considered the following regulatory requirements and guidance during its review of the LAR.

Regulatory Requirements

The regulations under 10 CFR 50.36, "Technical specifications," provide regulatory requirements related to the content of TSs. Section 50.36(b) of 10 CFR requires that each license authorizing the operation of a facility will include TSs and that the TSs will be derived from the safety analysis. Section 50.36(c) of 10 CFR specifies the categories that are to be included in the TSs including (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. Sections 50.36(c)(1), (c)(4), and (c)(5) of 10 CFR require the following:

- The regulation under 10 CFR 50.36(c)(1)(A), “Safety limits, limiting safety system settings and limiting control settings,” states, in part, that “[s]afety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity.”

- The regulation under 10 CFR 50.36(c)(2), “Limiting conditions for operation,” states, in part, “When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met."

- The regulation under 10 CFR 50.36(c)(4), "Design features" states that "[d]esign features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c) (1), (2), and (3) of this section."

- The regulation under 10 CFR 50.36(c)(5), "Administrative controls," states, in part, that "[a]dministrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure the operation of the facility in a safe manner."

Key regulatory requirements specified in 10 CFR 50.46(a) that are relevant to the proposed license amendment include:

- Each pressurized light-water reactor (LWR) fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must perform analysis of core cooling performance under postulated loss-of-coolant accident (LOCA) conditions using an acceptable evaluation model (EM).

- An acceptable LOCA EM must be used that either applies realistic methods with an explicit accounting for uncertainties or follows the prescriptive, conservative requirements of Appendix K to 10 CFR Part 50.

- Core cooling performance must be analyzed for a number of postulated LOCAs of different sizes, locations, and other characteristics to ensure that the most severe event is calculated.

The following ECCS acceptance criteria of 10 CFR 50.46(b)(1) though (b)(5) state in part:

(1) *Peak cladding temperature*. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

(2) *Maximum cladding oxidation*. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

(3) *Maximum hydrogen generation*. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) *Coolable geometry*. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

(5) *Long-term cooling*. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The licensee referred to acceptance criteria of 10 CFR 50.46(b)(1) through (5) as the peak cladding temperature (PCT) criterion, the maximum local oxidation (MLO) criterion, the hydrogen generation (or core wide oxidation (CWO)) criterion, the coolable geometry criterion, and the long-term cooling criterion respectively.

Final Safety Analysis Report (Standard Plant) (FSAR SP) section 3.1 (Reference 7), discusses the extent to which the design criteria for Westinghouse standardized nuclear unit power plant system plant structures, systems, and components important to safety comply with 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

The 10 CFR Part 50, Appendix A, General Design Criteria (GDCs) applicable to this LAR are as follows:

- GDC 4, "Environmental and dynamic effects design bases," states, in part, that structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated accidents, including loss-of-coolant accidents and shall be appropriately protected against dynamic effects, including the effects of discharging fluids, that may result from equipment failures.

- GDC 10, "Reactor design," states that "[t]he reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences."

- GDC 11, "Reactor inherent protection," states that "[t]he reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity."

- GDC 12, "Suppression of reactor power oscillations," states that "[t]he reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed."

- GDC 16, "Containment design," states that "[r]eactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require."

- GDC 20, "Protection system functions," states that "[t]he protection system be designed (1) to initiate, automatically, the operation of appropriate systems, including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety."

- GDC 25, "Protection system requirements for reactivity control malfunctions," states that "[t]he protection system be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods."

- GDC 26, "Reactivity control system redundancy and capability," states, in part, that two independent reactivity control systems of different design principles be provided, one of which can hold the reactor core subcritical under cold conditions.

- GDC 27, "Combined reactivity control system capability," states, in part, that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions.

- GDC 35, "Emergency core cooling," states:

> A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

- GDC 38, Containment heat removal," states, in part that "[a] system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

- GDC 50, "Containment design basis," states, in part, that "[t]he reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident.

Appendix K to 10 CFR Part 50 establishes required and acceptable features of EMs for heat removal by the ECCS after the blowdown phase of a LOCA. It consists of the following two parts:

- required and acceptable features of LOCA EMs and
- documentation required for LOCA EMs.

The first part specifies modeling requirements and acceptable methods for simulating significant physical phenomena throughout all phases of a design-basis LOCA event, including relevant heat sources, fuel rod performance, and thermal-hydraulic (T-H) behavior.

The second part specifies requirements for the documentation of LOCA EMs, including a complete description, a code listing, sensitivity studies, and comparisons against experimental data.

Regulatory Guidance

The NRC staff relied on the following sections of NUREG-0800, Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP) in its review of this LAR:

- Section 3.7.1, "Seismic Design Parameters,' Revision 4, December 2014 (Reference 8.a)
- Section 4.2, "Fuel System Design," Revision 3, March 2007 (Reference 8.b)
- Section 4.4, "Thermal and Hydraulic Design," Revision 2, March 2007 (Reference 8.c)
- Chapter 15, "Introduction – Transient and Accident Analyses," Revision 3, March 2007 (Reference 8.d)

Regulatory Guide (RG) 1.236, "Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents," June 2020 (Reference 9), details acceptable methods and procedures to use when analyzing a postulated control rod drop accident.

RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," Revision 3, October 2012 (Reference 10).

NRC Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," dated September 13, 2004 (Reference 11).

NRC Information Notice (IN) 2012-09, "Irradiation Effects on Fuel Assembly Spacer Grid Strength," dated June 28, 2012 (Reference 12).

2.3 Description

As described in report ANP-3944P (attachments 6 (publicly available) and attachment 10 (not publicly available) of enclosure 1 to the LAR dated October 12, 2022), the GAIA fuel design with M5® cladding consists of a 17x17 array with GAIA and intermediate GAIA mixing (IGM) grids, a lower high mechanical performance (HMP) grid, and an upper HMP grid. The fuel assembly includes a MONOBLOC guide tube design, M5 fuel rod design, and a GRIP lower nozzle. The fuel is standard UO_2 fuel with 2, 4, 6, and 8 weight-percent Gadolinia rods included. The GAIA fuel design is described in NRC-approved topical report (TR) ANP-10342P-A, Revision 0, “GAIA Fuel Assembly Mechanical Design,” September 2019 (Reference 13).

3.0 TECHNICAL EVALUATION

3.1 TS Changes

During operating cycle 27, the licensee plans to load up to eight GAIA fuel assemblies in the core, which will consist of four GAIA assemblies acquired under the VQP and four GAIA lead fuel assemblies previously present in the core during operating cycle 25. The licensee stated that due to core reload analysis considerations and limitations, no GAIA fuel was present in the core during operating cycle 26.

The licensee’s evaluation of the proposed changes in TS 2.1.1 is supported by the evaluations presented in reports ANP-3947P, ANP-3944P, ANP-3943P, and ANP-3969P (attachments 9 through 12, respectively, of enclosure 1 to the LAR (proprietary) (attachments 5 through 8, respectively are the non-proprietary versions). Attachment 9 of enclosure 1 to the LAR, provides a detailed description of the GAIA fuel, supporting bases for these proposed TS changes, and a summary of the analyses performed to support the acceptability of the use of GAIA fuel at Callaway. Attachments 10 through 12 of enclosure 1 to the LAR describes the licensee’s accident analyses and results of the FSAR SP Chapter 15 events to demonstrate continued compliance with the NRC regulatory requirements.

The licensee stated that the proposed change for the use of M5® zirconium alloy as fuel rod cladding material is necessary to support the transition to a limited number of GAIA fuel assemblies. This change requires a regulatory exemption from 10 CFR 50.46(a)(1)(i) and 10 CFR Part 50, Appendix K. Both regulations either explicitly or implicitly, state or assume that either zircaloy or ZIRLO™ is to be used as the fuel rod cladding material. The licensee’s justification of this exemption request is given in enclosure 2 to the LAR. Based on the NRC staff SE of the exemption request, the NRC staff found that the exemption request is acceptable.

In the licensee’s proposed TS 4.2.1 change, the word “Zircalloy” is changed to “zircaloy.” The NRC staff finds this change acceptable because it is an editorial change and is consistent with the zircaloy spelling in 10 CFR 50.46(a)(1)(i).

As specified in Callaway TS 5.6.5, “Core Operating Limits Report (COLR),” the licensee proposes to maintain the COLR in accordance with the administrative controls governing core reload design control and coordination. The licensee will continue using Westinghouse methods consistent with the methods specified in TS 5.6.5, with confirmatory analyses performed by Framatome. For this reason, the NRC staff agrees that no changes are necessary for the COLR to support operating cycles containing the limited number of GAIA fuel assemblies.

3.2 Small Break LOCA (SBLOCA) Analysis

The licensee performed an SBLOCA analysis in report ANP-3943P, “Callaway Small Break LOCA Analysis (attachments 7 and 11 of enclosure 1 to the LAR), to support the planned transition to a limited number of GAIA fuel assemblies. The licensee’s analysis verifies the applicability of certain regulatory requirements noted below following the transition to a limited number of GAIA fuel assemblies. NRC regulations require that licensees of operating LWRs analyze a spectrum of accidents involving the LOCA to assure adequate core cooling under the most limiting set of postulated design-basis conditions. LOCAs are postulated accidents that would result in the loss of reactor coolant from piping breaks in the RCS primary boundary at a rate in excess of the reactor coolant make-up system to replenish it. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core unless the water is replenished.

3.2.1 SBLOCA Description

The postulated SBLOCA is defined as a break in the RCS pressure boundary with an area less than or equal to 10 percent of the cold leg pipe area. The RPS and ECCS are provided to mitigate these accidents. The most limiting break location for SBLOCA analysis performed is in the cold leg pipe on the discharge side of the reactor coolant pump (RCP). This break location results in the largest amount of RCS inventory loss, the largest fraction of ECCS fluid discharged out the break, and the largest pressure drop between the core exit and the top of the downcomer. The SBLOCA event progression develops in the following distinct phases: (1) subcooled depressurization (also known as blowdown), (2) natural circulation, (3) loop seal clearing, (4) core boil-off, and (5) core recovery and long-term cooling. The duration of each of these phases is break size and system dependent. A detailed description of each of the phases is provided in report ANP-3943P.

The licensee performed SBLOCA analysis to support plant operation at a core power level of 3636 megawatt thermal (MWt) (including measurement uncertainty), a maximum-allowed local peaking factor (F_Q) of 2.5 (with uncertainties applied and an axial-dependent factor k(z) set to 1.0), a radial peaking factor of ($F_{\Delta H}$) of 1.65 (including measurement uncertainty), and up to 5 percent steam generator (SG) tube plugging per SG.

3.2.2 Methodology

The licensee performed the SBLOCA analysis using the NRC-approved SBLOCA methodology documented in TR EMF-2328(P)(A), Revision 0, “PWR [Pressurized-Water Reactor] Small Break LOCA Evaluation Model, S-RELAP5 Based, March 2001 (Reference 14) and Supplement 1 to TR EMF-2328(P)(A), “Revision 0, “PWR Small Break LOCA Evaluation Model, S-RELAP5 Based, December 2016 (Reference 15).

The licensee used the evaluation model for event response of the primary and secondary systems, and the hot fuel rod is based on the use of the following two computer codes:

- The RODEX2-2A code to determine the burnup dependent initial fuel rod conditions for the system calculations.

- The S-RELAP5 code to predict the primary and secondary system T-H and hot rod transient response.

The S-RELAP5 code was used in the NRC-approved SBLOCA methodology and documented in TR EMF-2328(P)(A). The use of S-RELAP5 and RODEX2A is required for SBLOCA analysis using Supplement 1 to TR EMF-2328(P)(A). In the supplemental letter dated May 9, 2023, the licensee responded to the NRC staff’s question regarding version control of these codes and that the Framatome computer codes and the code maintenance process are controlled by Framatome software procedures. The Framatome procedures are compliant with American Society of Mechanical Engineers (ASME) Nuclear Quality Assurance (NQA)-1, “Quality Assurance Requirements for Nuclear Facility Applications” (version 2008/2009). The licensee confirmed that the code versions used in the Callaway SBLOCA analysis are verified under the Framatome software procedures. The NRC staff therefore finds the codes appropriate for use with the applied methods.

3.2.3 Analysis

The licensee performed SBLOCA analysis consistent with the NRC-approved SBLOCA methodology documented in TR EMF-2328(P)(A) and Supplement 1 to TR EMF-2328(P)(A). The goal of the analysis is to demonstrate that the ECCS, while operating with GAIA fuel in the core, will continue to satisfy the ECCS acceptance criteria given in 10 CFR 50.46(b)(1) through (b)(4). A break spectrum analysis for SBLOCA was performed for breaks of varying diameters of up to 10 percent of the flow area for the cold leg pump discharge. The spectrum analyzed included a break size range from 1.00 to 8.70 inches in diameter, with a break size interval sufficient to establish a PCT trend.

In addition to the cold leg pump discharge break spectrum analysis, the licensee performed sensitivity studies for a delayed RCP trip, a break in an attached pipe, and a different ECCS temperature. For the delayed RCP trip, a trip time of 10 minutes following event initiation is analyzed to evaluate the adequacy of the specified trip criteria and demonstrate compliance to 10 CFR 50.46(b)(1) through (b)(4) criteria. The licensee performed an analysis of the ruptures in attached piping that compromise the ability to inject emergency coolant into the RCS. The attached piping study analyzed breaks in the accumulator line and high head safety injection (HHSI) line. The ECCS temperature sensitivity study analyzed the sensitivity to ECCS fluid temperatures different from those used in the break spectrum analysis.

The NRC staff finds the SBLOCA analysis performed by the licensee acceptable as it uses an NRC-approved methodology, and the results show compliance to the requirements in 10 CFR 50.46(b)(1) through (b)(4) acceptance criteria.

3.2.4 Results

The licensee’s SBLOCA break spectrum analysis resulted in a limiting PCT of 1618°F for the limiting 8.70-inch diameter cold leg pump discharge break. The same break produced the limiting MLO of 3.38 percent, including the **[[ ]]** The CWO is < 0.01 percent. Therefore, the NRC staff finds the analysis results demonstrate the adequacy of the ECCS to satisfy the criteria given in 10 CFR 50.46(b)(1) to (b)(3). Further, maintaining compliance to 10 CFR 50.46(b)(1) to (b)(3) criteria also ensures the 10 CFR 50.46(b)(4) criteria on maintaining the core amenable to cooling will be satisfied. The NRC staff evaluation on maintaining the coolable geometry of the fuel under the seismic and LOCA load combination to satisfy the 10 CFR 50.46(b)(4) criteria is provided in section 3.7.3 of this SE.

The results of the delayed RCP trip study performed by the licensee demonstrated that there is at least 10 minutes for operators to trip all four RCPs after the specified trip criteria being met with considerable margin to the 10 CFR 50.46(b)(1) to (b)(4) acceptance criteria. The results from analysis of the ruptures in attached piping that compromise the ability to inject emergency coolant into the RCS showed to be less limiting than those of the break spectrum analysis. **[[**

]] Hence, the acceptability of 10 CFR 50.46(b)(1) to (b)(4) criteria from the SBLOCA break spectrum analysis remains applicable to the attached piping ruptures study and the ECCS sensitivity study.

3.2.5 Compliance with NRC Staff Imposed Limitations and Conditions

Section 3.5 of ANP-3943P states that the NRC-approved supplemented EMF-2328(P)(A) method contains no restrictions. While the Supplement 1 to EMF-2328(P)(A) does not contain any direct limitations, it addresses compliance to the 10 percent cold leg break size limitation that is applied to the use of S-RELAP5 code for the SBLOCA analysis. This limitation is discussed by the licensee in attachment 13, “Response to Insufficiency Items from First License Amendment Request,” of the enclosure to the LAR. The licensee stated that it performed calculations for the SBLOCA in a consistent manner with this break size limitation. Based on its review, the NRC staff finds this limitation is satisfied.

3.2.6 Conclusions

The NRC staff reviewed the information in the licensee’s submittal pertaining to the analysis of the SBLOCA event for GAIA fuel with M5® cladding in the core to support plant operation at a core power level of 3636 MWt (includes measurement uncertainty), a maximum-allowed local peaking factor (F_Q) of 2.5 (with uncertainties applied and an axial-dependent factor k(z) set to 1.0), a radial peaking factor ($F_{\Delta H}$) of 1.65 (including measurement uncertainty), and up to 5 percent SG tube plugging per SG.

The NRC staff’s review verified that SBLOCA break spectrum analysis results meet the limiting PCT limits and the total MLO and CWO limits set by 10 CFR.50.46(b)(1) through (b)(3). The NRC staff finds the delayed RCP trip study performed by the licensee to be acceptable as it shows that there is at least 10 minutes for operators to trip all four RCPs after the trip criteria is being met. The NRC staff finds the results from analysis of the ruptures in attached piping to be acceptable as they are less limiting than the limiting break spectrum case.

The NRC staff’s review confirmed that the licensee has processes to assure that the Callaway specific input parameter values and operator action times (where appropriate) that were used to conduct the analyses will assure that 10 CFR.50.46(b)(1) through (b)(4) limits are not exceeded following a SBLOCA. Based on its review, the NRC staff also finds that the licensee presented evaluations for the heat removal by the ECCS after the blowdown phase of a LOCA to be acceptable. In addition, the NRC staff finds that the licensee’s analysis showed it will continue to meet GDCs 4, 27, and 35 of 10 CFR Part 50, Appendix A and 10 CFR Part 50, Appendix K requirements.

3.3 Large Break LOCA (LBLOCA) Analysis Report (ANP-3944P (Attachment 10 to LAR))

To support the planned transition to a limited number of GAIA fuel assemblies, the licensee analyzed the LBLOCA event to verify applicable regulatory requirements are satisfied. NRC regulations require the licensee to analyze a spectrum of LOCAs to ensure adequate core cooling under the most limiting set of postulated design-basis conditions. The postulated spectrum of LOCAs range from scenarios with leakage rates just exceeding the capacity of normal makeup systems up through those involving rapid coolant loss from the complete severance of the largest pipe in the RCS.

3.3.1 LBLOCA Description

During normal plant operation at full power, a LBLOCA is initiated by a postulated rupture of the RCS primary piping. The most limiting break is an instantaneously occurring break in the cold leg piping between the RCP and the reactor vessel. A worst-case single failure is also assumed to occur during the accident. The single failure for this analysis, as defined in the EM, is the loss of one ECCS injection train without the loss of containment spray.

The LBLOCA is described in three phases: the blowdown phase, the refill phase, and the reflood phase. The licensee described these phases in section 3.2, “Description of LBLOCA Event,” of report ANP-3944P, “Callaway Realistic Large Break LOCA Analysis with GAIA Fuel Design,” Revision 1 (attachment 6 (non-proprietary) and attachment 10 (proprietary) of enclosure 1 to the LAR).

3.3.2 Methodology

The NRC-approved TR EMF-2103(P)(A), “Realistic Large Break LOCA Methodology for Pressurized Water Reactors,” June 2016 (Reference 16), describes the Framatome methodology developed for the realistic evaluation of a LBLOCA for PWRs with recirculation (U-tube) SGs. It covers specifically the Westinghouse 3-loop and 4-loop plant designs; and Combustion Engineering (CE) plants, all with fuel assembly lengths of 14 feet or less and ECCS injection to the cold legs. Since Callaway is a 4-Loop Westinghouse designed PWR with recirculation SGs, this methodology is applicable to Callaway for the LBLOCA analysis. The EM in TR EMF-2103(P)(A) for the LBLOCA response of the RCS, secondary system, and the fuel rod used in the analysis is based on the use of the following computer codes:

- COPERNIC code provides initial conditions for the S-RELAP5 code including computation of the initial fuel stored energy, fission gas release, and the transient fuel-cladding gap conductance.

- S-RELAP5 code for the thermal-hydraulic system calculations, which includes ICECON code for containment response.

The licensee identified a difference in the previous cladding swelling and rupture model (SRM) used in the NRC-approved S-RELAP5 model in TR EMF-2103(P)(A) and the model applied for the LBLOCA analysis. In the supplement dated May 9, 2023, the licensee stated that a
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The NRC staff finds the difference in the cladding SRM model used in the LBLOCA analysis from the previous model in the NRC-approved TR EMF-2103(P)(A) methodology acceptable because **[[**

]] This difference in the SRM model has been presented in recent NRC-approved LBLOCA analyses in license amendments issued for Palo Verde Nuclear Generating Station and Shearon Harris Nuclear Power Plant (References 18 and 19, respectively).

3.3.3 Analysis

The licensee's LBLOCA analysis is based on a statistical realistic LOCA EM in accordance with the methodology in TR EMF-2103(P)(A) instead of conservative EMs specified by 10 CFR Part 50, Appendix K. As described in Callaway FSAR SP section 15.6.5, the LBLOCA analysis of record (AOR) was performed in accordance with 10 CFR Part 50, Appendix K models. For performing the statistical analysis, the licensee created **[[**

]] the licensee sampled each key input parameter over a range established through code uncertainty assessment or expected operating limits provided either by TSs or plant data. The licensee considered the key LOCA parameters listed in TR EMF-2103(P)(A), table A-6, and the uncertainty range associated with each of these parameters given in TR EMF-2103(P)(A), table A-7.

The nodalization details used in the S-RELAP5 code is shown in report ANP-3944P, figures 3-1 through 3-3. The key features of the licensee's LBLOCA model are as follows:

- Explicitly modeled reactor vessel, pressurizer, RCS, and ECCS.

- For each RCS loop, the ECCS model includes an injection connection to the cold leg for the accumulator, a connection for HHSI, and a connection for low head safety injection (LHSI).

- Intermediate head safety injection (IHSI) and HHSI are modeled as a combined system and identified as HHSI.

- ECCS injection connections to the cold leg pipes are downstream of the RCP discharge.

- ECCS injection is modeled as a table of flow versus backpressure.

- Model includes isolation of the SG secondary side by instantaneously closing the main steam isolation valve and feedwater trip at the time of the break.

- Following a steady-state condition, the transient analysis is initiated by introducing a break into one of the loops.

- The LOCA blowdown, refill, and reflood transient is analyzed using the S-RELAP5 and the containment pressure is calculated by the ICECON module which is within the S-RELAP5 code.

Report ANP-3944P, table 4-1 shows the plant parameters and ranges used for the analysis. The analysis assumes full-power operation at a core power level of 3636 MWt (including 2 percent power measurement uncertainty), a maximum-allowed local peaking factor (F_Q) of 2.50 (represents total peaking with an axial-dependent factor k(z) set to 1.0), an $F_{\Delta H}$ of 1.65 (includes uncertainty), and up to 5 percent SG tube plugging per SG. This analysis also addresses typical operational ranges or TS limits (whichever is applicable) with regard to **[[**

]] The analysis explicitly analyzes fresh and once-burned fuel assemblies. The summary of the major parameters for the demonstration case analysis are identified in report ANP-3944P, table 4-5. 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. TR EMF-2103(P)(A), section 7.9.3.3, "Clad Ballooning, Rupture and Area Adjustment Models," provides a discussion and consequences of FSRR and is documented in the supporting analyses in the TR. **[[**

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In the supplement dated May 9, 2023, the licensee provided a table showing the upper and lower limits of plant parameters and a table showing the ranges of plant operating parameters used in the LBLOCA statistical analysis and their TS limits.

In the supplement dated May 9, 2023, the licensee stated that **[[**

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3.3.4 Results

Report ANP-3944P, table 4-4 provides the results of the licensee's analysis for compliance with 10 CFR 50.46(b)(1), (b)(2), and (b)(3). Table 3-1 below (extracted from report ANP-3944P, table 4.4) shows the upper tolerance limit (UTL) for 95/95 simultaneous coverage/confidence PCT, MLO, and CWO results for **[[** **]]** cases.

Table 3-1: Results

Parameter	Value [[ ]]	Value [[ ]]	10 CFR 50.46(b)(1), (b)(2), and (b)(3) Acceptance Criteria
PCT (°F)	1561	**[[ ]]**	≤ 2,200
MLO (%)	2.35	**[[ ]]**	≤ 17
CWO (%)	0.028	**[[ ]]**	≤ 1

The results in table 3-1 above shows the limiting **[[ ]]** results for 95/95 simultaneous coverage/confidence meet the 10 CFR 50.46(b) criteria with a PCT of 1561°F, MLO of 2.35 percent and a total CWO of 0.028 percent. The PCT of 1561°F occurred in a once-burned 2 weight-percent Gadolinia rod with an assembly burnup of 26.2 GWd/MTU. Therefore, the NRC staff finds that the results of the licensee's LBLOCA analysis demonstrate that the ECCS is adequate to support the 10 CFR 50.46(b)(1), (b)(2), and (b)(3) acceptance criteria.

The licensee stated that the results used to demonstrate compliance with the 10 CFR 50.46(b) criteria are only applicable to the GAIA fuel. However, the analysis includes considerations for the mixed core scenario. **[[**

]] Consistent with the modeling features presented in section A.1.3.6.2.4 of TR EMF-2103(P)(A) and to justify that the **[[**

]] The supporting analysis documented in TR EMF-2103(P)(A) implements the mixed core modeling. Therefore, the NRC staff finds it acceptable that the **[[**

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The 10 CFR 50.46(b)(4) and (b)(5) acceptance criteria require: (a) to maintain coolable geometry of the fuel and (b) the ability to provide long term core cooling respectively. Related to GL 2004-02 (Generic Safety Issue (GSI)-191), the potential impacts of adding GAIA fuel on items (a) and (b) would be debris blockage of the fuel and excessive boron concentration in the core. By letter dated October 21, 2022 (Reference 20), the NRC staff approved the resolution of the GL 2004-02 issues documented in the Callaway Amendment No. 228. For operation with eight GAIA fuel assemblies, by a qualitative evaluation in accordance with the "NRC Staff Review Guidance for In-Vessel Downstream Effects Supporting Review of Generic Letter 2004-02 Responses" (Reference 21), the licensee concluded that the GAIA fuel assemblies would not adversely affect the ability to maintain a coolable geometry and provide long term core cooling.

In enclosure 2, "License Amendment Request for Callaway Risk-Informed Approach to Resolution of Generic Letter 2004-02," of the LAR for Amendment No. 228, the licensee's evaluation of in-vessel performance criteria for boron precipitation in accordance with TR WCAP-17788-NP, "Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090) (Reference 22) shows that the current hot leg switchover timing is appropriate with debris effects considered.

The NRC staff finds it acceptable that for the transition to a limited number of GAIA fuel assemblies, the impact of debris blockage and increase in boron concentration would not affect items (a) and (b) because the current evaluations are not sensitive to a minor increase (less than 0.4 cubic feet (ft^3) per assembly) in fuel volume associated with the addition of eight GAIA fuel assemblies. Therefore, from the standpoint of GL 2004-02, the NRC staff finds that the 10 CFR 50.46(b)(4) and (b)(5) criteria are satisfied.

The NRC staff evaluation of 10 CFR 50.46(b)(4) criteria on maintaining the coolable geometry of the fuel under the seismic and LOCA load combination is provided in section 3.7.3 of this SE.

3.3.5 Compliance with NRC Imposed Limitations and Conditions (L&Cs) for TR EMF-2103(P)(A)

For the application of the EMF-2103(P)(A) methodology, there are 11 L&Cs listed in section 4.0 of the NRC staff's SE for TR EMF-2103(P)(A). The licensee's compliance statements for these L&Cs are provided in report ANP-3944P, section 3.7. As discussed below, the NRC staff evaluated the licensee's compliance with each of the L&Cs provided in ANP-3944P, section 3.7, and finds that the L&C are satisfied.

(1) This EM was specifically reviewed in accordance with statements in TR EMF-2103, Revision 3. The NRC staff determined that the EM is acceptable for determining whether plant-specific results comply with the acceptance criteria set forth in 10 CFR 50.46(b), paragraphs (1) through (3). AREVA did not request, and the NRC staff did not consider, whether this EM would be considered applicable if used to determine whether the requirements of 10 CFR 50.46(b)(4), regarding coolable geometry, or (b)(5), regarding long-term core cooling, are satisfied. Thus, this approval does not apply to the use of S-RELAP5- based methods of evaluating the effects of grid deformation due to seismic of LOCA blowdown loads, or for evaluating the effects of reactor coolant system boric acid transport. Such evaluations would be considered separate methods.

The NRC staff finds L&C (1) is satisfied because the LBLOCA analysis presented based on the TR EMF-2103(P)(A) EM satisfies the acceptance criteria set forth in 10 CFR 50.46(b), paragraphs (1) through (3).

(2) TR EMF-2103, Revision 3, approval is limited to application for 3-loop and 4-loop Westinghouse-designed nuclear steam supply systems (NSSSs), and to Combustion Engineering-designed NSSSs with cold leg ECCS injection, only. The NRC staff did not consider model applicability to other NSSS designs in its review.

TR EMF-2103(P)(A) is applicable because Callaway is a 4-loop Westinghouse designed reactor with cold leg ECCS injection. Therefore, NRC staff finds L&C (2) is satisfied.

(3) EM is approved based on models that are specific to AREVA proprietary M5 fuel cladding. The application of the model to other cladding types has not been reviewed.

The NRC staff finds L&C (3) is satisfied because the LBLOCA analysis is based on M5® cladding.

(4) Plant-specific applications will generally be considered acceptable if they follow the modeling guidelines contained in Appendix A to EMF 2103, Revision 3. Plant-specific licensing actions referencing EMF 2103, Revision 3, analyses should include a statement summarizing the extent to which the guidelines were followed, and justification for any departures.

L&C (4) that the licensee completely followed the modeling guidelines in TR EMF-2103(P)(A), Revision 3, Appendix A.

(5) The response to RAI [request for additional information] 15 indicates that the fuel pellet relocation packing factor is derived from data that extend to currently licensed fuel burnup limits (i.e., rod average burnup of **[[ ]]**). Thus, the approval of this method is limited to fuel burnup below this value. Extension beyond rod average burnup of **[[ ]]** would require a revision or supplement to TR EMF-2103, Revision 3, or plant-specific justification.

The NRC staff finds L&C (5) is satisfied because as the licensee stated, the LBLOCA analysis did not exceed the rod average burnup of **[[ ]].**

(6) The response to RAI 15 indicates that the fuel pellet relocation packing factor is derived from currently available data. Should new data become available to suggest that fuel pellet fragmentation behavior is other than that suggested by the currently available database, the NRC may request AREVA to update its model to reflect such new data.

As stated in TR EMF-2103(P)(A), section 7.9.3.3.1, **[[ ]]** The licensee stated that the LBLOCA analysis used the NRC-approved TR EMF-2103(P)(A) relocation packing factor application, and **[[ ]]** Therefore, the NRC staff finds L&C (6) is satisfied.

(7) The regulatory limit contained in 10 CFR 50.46(b)(2), requiring cladding oxidation not to exceed 17 percent of the initial cladding thickness prior to oxidation, is based on the use of the Baker-Just oxidation correlation. To account for the use of the C-P [Cathcart-Pawel] correlation, this limit shall be reduced to 13 percent, inclusive of pre-transient oxide layer thickness.

As discussed in TR EMF-2103(P)(A), section 8.4.9, the licensee used the Cathcart-Pawel correlation for MLO analysis to satisfy the 10 CFR 50.46(b)(2) acceptance criteria. In the supplement dated May 9, 2023, the licensee stated that the analysis results confirm that the MLO **[[ ]]** for fresh and once-burned UO_2 and fresh and once-burned rods with gadolinium is less than 13 percent, which includes the pre-transient oxide layer thickness. **[[** **]]** The NRC staff finds L&C (7) is satisfied because the licensee performed the MLO analysis according to the NRC-approved methodology with acceptable results satisfying the 10 CFR 50.46(b)(2) acceptance criterion.

(8) In conjunction with Limitation 7 above, C-P oxidation results will be considered acceptable, provided plant-specific **[[** **]]** If second-cycle fuel is identified in a plant specific analysis, whose **[[** **]]** the NRC staff reviewing the plant-specific analysis may request technical justification or quantitative assessment, demonstrating that **[[** **]]**

According to the results presented in report ANP-3944P, table 4-4, the MLO UTL **[[** **]]** Therefore, the NRC staff finds L&C (8) is satisfied.

(9) The response to RAI 13 states that all operating ranges used in a plant-specific analysis are supplied for review by the NRC in a table like Table B-8 of TR EMF-2103, Revision 3. In plant-specific reviews, the uncertainty treatment for plant parameters will be considered acceptable if plant parameters are **[[** **]]** as appropriate. Alternative approaches may be used, provided they are supported with appropriate justification.

As shown in report ANP-3944P, **[[** **]]** Therefore, the NRC staff finds L&C (9) is satisfied.

(10) **[[** **]]**

The NRC staff finds L&C (10) is satisfied because as stated by the licensee, **[[** **]]** were not used in this analysis.

(11) Any plant submittal to the NRC using TR EMF-2103, Revision 3, which is not based on the first statistical calculation intended to be the analysis of record must state that a re-analysis has been performed and must identify the changes that were made to the evaluation model and/or input in order to obtain the results in the submitted analysis.

The NRC staff finds L&C (11) is satisfied because as stated by the licensee, the LBLOCA analysis presented in report ANP-3944P is the first statistical application of TR EMF-2103(P)(A).

3.3.6 Conclusions

Based on the above evaluation of the licensee's information presented in the LAR and the supplement dated May 9, 2023, the NRC staff conclusions are as follows:

- The licensee used NRC-approved methods for the analyses and demonstrated conformance to the acceptance criteria contained in 10 CFR 50.46(b).

- The nodalization scheme is consistent with the code input guidelines in TR EMF-2103(P)(A), appendix A.

- The analyses reflect the batch introduction of the GAIA fuel assembly design. The NRC staff also confirmed that the GAIA fuel assembly design introduction will be accomplished acceptably.

- The L&Cs specified in the NRC staff SE for the TR EMF-2103(P)(A) are satisfied as evaluated by the licensee in report ANP-3944P, table 3-1, and as evaluated above by the NRC staff.

Based on the above technical conclusions, NRC staff conclusions on the regulatory evaluation of the LBLOCA analysis are as follows:

- Section 50.46 of 10 CFR relevant requirements are satisfied based on the following:

  - The licensee used an NRC-approved EM to perform the LBLOCA analysis and demonstrated an acceptable ECCS performance by applying a realistic method explicitly accounting for uncertainties.

  - The NRC staff has approved the licensee exemption requesting to apply M5® cladding material instead of zircaloy or ZIRLO™ cladding to the GAIA fuel rods.

  - The licensee analyzed ECCS performance for several postulated LOCAs of different sizes, locations, and other characteristics to ensure that the most severe event is calculated.

  - The licensee demonstrated that the acceptance criteria in 10 CFR 50.46(b)(1) through (b)(5) are satisfied.

  - ECCS transfers heat from the reactor core following any LOCA at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

  - Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities are available assuring that in the

LOOP or no-LOOP condition, the ECCS safety function is accomplished, assuming a single failure.

- The NRC staff finds GDC 4 is satisfied because the licensee demonstrated that the coolable geometry of the fuel assemblies is maintained under the dynamic effects of seismic plus LOCA load combination.

- The NRC staff finds GDC 27 is satisfied because the licensee's analysis demonstrated that the reactivity control system along with boron addition from the ECCS will maintain the reactor sub-critical during a LBLOCA.

- The NRC staff finds GDC 35 is satisfied because while assuming the most limiting single failure and LOOP or no-LOOP conditions, the licensee's LBLOCA analysis, demonstrated that the ECCS performed it intended functions of satisfying 10 CFR 50.46(b) criteria so that the core cooling is continued and the metal-water reaction that causes fuel oxidation is far below the acceptable limits.

On the basis of the NRC staff's technical and regulatory conclusions described above, the staff finds the analysis and results of the LBLOCA for the transition to a limited number of GAIA fuel assemblies acceptable.

3.4 Containment Analysis

Based on possible 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 LOCA containment AORs. In the supplement dated May 9, 2023, the licensee provided evaluations of the following AORs for NRC staff review:

- M&E [Mass and Energy] release analyses for LBLOCA (Callaway FSAR SP Section 6.2.1.3).

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

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

- Available NPSH [net positive suction head] for containment spray pumps, and residual heat removal pumps (Callaway FSAR SP Table 6.2.2-7)

3.4.1 LOCA M&E Release and Containment Response

In the supplement dated May 9, 2023, the licensee stated that the short and long term LBLOCA M&E releases and long term SBLOCA M&E release AOR remain applicable to the Callaway VQP with up to eight GAIA fuel assemblies in the core. Since there is no impact on the AOR M&E releases, there would be no impact on the containment pressure and temperature response AOR. The NRC staff finds it acceptable that the containment response is not impacted because the AOR M&E release is not affected by the addition of the eight GAIA fuel assemblies.

3.4.2 Minimum Containment Pressure for LBLOCA Analysis

Report ANP-3944P, section 3.3 states that the containment pressure is calculated by the ICECON code module in parallel within the S-RELAP5 code. The ICECON code runs concurrently with the S-RELAP5 and **[[** **]]** The three modeling factors described in TR EMF-2103(P)(A), section 3.1.3.4.1 that are applied, assure that the containment back pressure in the LBLOCA calculation is conservatively minimized. As stated in report ANP-3944P, section 3.2, the single-failure for this analysis is the loss of one ECCS injection train without the loss of containment spray. **[[** **]]** Report ANP-3944P, table 4-1 shows the initial conditions for the containment sprays, and figure 4-16 shows the containment pressure response for the demonstration case.

The NRC staff finds the analysis for containment back pressure as an input to the LBLOCA analysis acceptable because the licensee used conservative inputs to minimize the pressure to conservatively maximize the PCT.

3.4.3 NPSH Analysis

FSAR SP, section 6.2.1.5 states that the LOCA containment pressure is conservatively minimized in the NPSH analysis for the pumps that draw water from the sump in the LOCA recirculation phase. In its LAR submitted by letter dated March 31, 2021 (Reference 23), enclosure 2, attachment 2-5, section 6.3A.1.1, the licensee stated that containment accident pressure (CAP) of 1.7 pounds per square inch (psi) (approximately 10 percent of the developed containment pressure) is credited for available NPSH during the LOCA phase when containment temperature is above 212°F to assure no flashing of sump water occurs in the debris bed. The licensee states that since the LOCA containment M&Es release input for the NPSH analysis is not impacted by the eight GAIA assemblies, the 1.7 psi CAP credit in the analysis for the GSI-191 resolution is not affected. The licensee's LAR was approved by the NRC staff in an SE for Amendment No. 228, by letter dated October 21, 2022, addressing GSI-191 and the response to GL 2004-02).

The FSAR SP, section 6.3.2.2 NPSH discussion does not credit CAP for containment spray pumps and RHR pumps that draw water from the sump in the LOCA recirculation phase. FSAR SP, table 6.2.2-7 provides assumptions and results of the NPSH analyses for these pumps, and FSAR SP table 6.3-1 provides the available and required NPSH for the ECCS pumps. In addition to considering the static head and suction line pressure loss, the licensee's calculation of available NPSH in the LBLOCA recirculation phase assumes that the vapor pressure of the liquid in the sump is equal to the containment ambient pressure. The NRC staff finds that the licensee's NPSH AOR is not impacted by including GAIA fuel assemblies in the core because (a) the 1.7 psi credit is not specifically related to NPSH; rather, it is credited to prevent boiling within the debris bed, and (b) the assumptions and the results in the NPSH AOR provided in the FSAR SP are not affected since the licensee determined that the M&E release is not affected by the addition of GAIA fuel assemblies in the core.

3.4.4 Conclusions

Based on the licensee's information presented in the LAR supplement dated May 9, 2023, and the above NRC staff evaluations, the NRC staff's conclusions are as follows: .

- The AOR for the LOCA M&E release and containment response is not affected by the addition of GAIA fuel assemblies.

- The licensee performed the minimum containment pressure analysis using the NRC-accepted ICECON code module running in parallel within the S-RELAP5 code and conservatively minimized the containment pressure as a boundary condition for the ECCS analysis.

- The current GSI-191 resolution is not affected by the addition of the GAIA assemblies because the 1.7 psi CAP credit in the AOR is not affected.

- The AOR available NPSH analysis for the pumps that draw water from the sump during the LOCA recirculation phase is not affected by the addition of the GAIA fuel assemblies.

Based on the above technical conclusions, the NRC staff finds the following 10 CFR Part 50 Appendix A, GDC requirements: GDCs 16, 38, and 50 are satisfied.

Accordingly, based on the technical and regulatory conclusions described above, the NRC staff finds the licensee's evaluation of containment AOR for transition to a limited number of GAIA fuel assemblies acceptable.

3.5 Non-LOCA Events Analysis (report ANP-3969P (Attachment 12 to LAR))

The licensee analyzed FSAR SP Chapter 15 non-LOCA events affected by the fuel design parameters based on transition to a limited number of GAIA fuel assemblies. The analysis of these events is affected because of changes in thermal-hydraulic performance and neutronics inputs to the safety analyses as they potentially challenge the DNBR and fuel centerline melt (FCM), specified acceptable fuel design limits (SAFDLs), as well as other event-specific criteria such as time-to-criticality for the boron dilution event. The licensee did not analyze events that are not affected by the change in fuel design parameters because they would remain bounded by their AOR.

3.5.1 Methodology and Computer Codes

The licensee used the NRC-approved TR EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Revision 1 and Supplement 1, Revision 0, (Reference 24) methodology for evaluating the non-LOCA events. For each event, the licensee used conservatively biased inputs and a nodalization scheme in compliance with the SE for TR EMF-2310(P)(A).

The licensee made the following changes in the non-LOCA system transient analyses and downstream analyses. These changes are within the scope of the EMF-2310(P)(A) methodology.

- Replaced RODEX2 code with the COPERNIC code for the purpose of generating the fuel thermal-conductivity, heat capacity and fuel pellet-to-clad gap coefficient inputs for the average core and hot spot models in the S-RELAP5 code to account for the effects of thermal conductivity degradation. The COPERNIC fuel properties and gap coefficients are conservatively implemented in the S-RELAP5 model as approved in TR EMF-2310(P)(A). Instead of the EMF-92-081(P)(A) (Reference 25) methodology and to explicitly account for the thermal conductivity degradation, the licensee used COPERNIC code for calculating the linear heat generation rate (LHGR) that corresponds to the FCM temperature limit for each fuel rod type.

- **[[**

  ]] This flexibility of modeling for MSLB is permissible as stated in section 5.4 of TR EMF-2310(P)(A).

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- The NRC staff finds this change acceptable because the **[[**

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- For the “Feedwater System Malfunctions that Result in an Increase in Feedwater Flow” and “Steam System Piping Failure” , the licensee used COBRA-FLX code in place of the XCOBRA-IIIC code for the DNB analyses as indicated in the SE included in TR ANP-10311P-A, “COBRA-FLX: A Core Thermal-Hydraulic Analysis Code” (Reference 26). The licensee described the following difference in COBRA-FLX (standalone and within ARTEMIS) from the COBRA-FLX approved in TR ANP-10311P-A:

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The NRC staff finds the change in using the COBRA-FLX code instead of the XCOBRA-IIIC code and the change within the COBRA-FLX from its NRC-approved version in TR ANP-10311P-A, while using the appropriate DNB correlation, acceptable based on the following:

- COBRA-FLX is approved for DNB analysis in TR ANP-10311P-A.

- The licensee justified allowing the **[[** **]]**

The licensee used the following computer codes for the non-LOCA event analysis for the GAIA fuel transition:

- S-RELAP5 code is a Framatome modification of the RELAP5/MOD2 code documented in NRC-approved TR EMF-2310(P)(A). The S-RELAP5 code is used for simulation of the system response to non-LOCA transient events. TR BAW-10240(P)(A), "Incorporation of M5® Properties in Framatome ANP Approved Methods" (Reference 27), incorporates M5® cladding properties into the S-RELAP5 code.

- COPERNIC code documented in NRC-approved TR BAW-10231-NP-A, "COPERNIC Fuel Rod Design Computer Code]," Revision 1 (Reference 28) is used to perform thermal-mechanical calculations for a fuel rod under normal operating conditions. It is also used to establish the FCM LHGR limit as a function of exposure.

- XCOBRA-IIIC code documented in NRC-approved TR XN-NF-75-21(P)(A), Revision 2 (Reference 29) is a steady-state thermal-hydraulics code used for calculating the axial and radial flow and enthalpy distributions within assemblies and sub-channels for non-LOCA events.

- COBRA-FLX code documented in NRC-approved TR ANP-10311P-A is a steady-state and transient thermal-hydraulics code used for calculating (a) the axial and lateral flow, pressure, and enthalpy distribution within assemblies and subchannels, (b) minimum departure from nucleate boiling ratio (MDNBR) when used in conjunction with boundary conditions provided from S-RELAP5 transient analysis and the ORFEO-GAIA and ORFEO-NMGRID DNB correlations documented in TR ANP-10341(P)(A), "The ORFEO-GAIA and ORFEO-NMGRID Critical Heat Flux Correlations" (Reference 30).

Note: In response to Question 4 in the supplement dated June 21, 2023, the licensee stated that terms "CHF" [critical heat flux] and "DNB" and their associated limits are used interchangeably.

- ARTEMIS - Framatome's PWR neutronics methodology, which uses the ARCADIA code suite documented in NRC-approved TRs ANP-10297P-A, "The ARCADIA Reactor Analysis System for PWRs Methodology Description and Benchmarking Results Revision 0," (Reference 31) and NRC-approved TR ANP-10297, "The ARCADIA Reactor Analysis System for PWRs Methodology Description and Benchmarking Results," Revision 0, Supplement 1PA (Reference 32). The ARTEMIS code is used to calculate the core reactivity, nodal power distribution, pin power distribution, incore and excore detector responses, and to simulate fuel shuffling, insertion, and discharge.

3.5.2 Methodology Changes

The licensee described the differences in the method used for the transient analyses from that in the approved TR EMF-2310(P)(A) as given below.

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  The NRC finds the changes acceptable because **[[**

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- **[[**

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  The NRC finds this change acceptable because **[[**

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- For the "Feedwater System Malfunctions that Result in an Increase in Feedwater Flow" and the "Steam System Piping Failure" events, the ARTEMIS code is used to calculate the radial and axial power distribution and the reactivity verification. The ARTEMIS and COBRA-FLX codes are internally coupled. The ARTEMIS code feeds the conditions within the core to the COBRA-FLX code. The COBRA-FLX code calculates moderator densities and temperatures to transfer back to the ARTEMIS code.

The NRC staff finds this change acceptable because the coupling process to obtain the power distribution and core reactivity is similar to that described in TR EMF-2310(P)(A) section 5.4.4.1.

3.5.3 Key Parameters

For non-LOCA events, the licensee used the plant operating conditions given below and included measurement uncertainties for performing conservative analyses for these events. For pressurizer pressure and reactor vessel average temperature, the licensee used nominal values to predict more realistic protective system responses. Initial condition measurement uncertainties are either treated deterministically or statistically in the DNB calculations.

- Rated core power (100 percent rated thermal power (RTP)) = 3565 MWt, measurement uncertainty = ±2 percent of RTP.
- Nominal HFP average reactor coolant temperature = 588.4°F.
- Nominal HZP reactor vessel average temperature = 556.8°F, measurement uncertainty and allowance for steady-state fluctuations = +4.3/-3.5°F
- TS minimum RCS flow rate = 374,400 gallons per minute (gpm).
- Maximum SG tube plugging = 5 percent.
- Nominal pressurizer pressure = 2235 pounds per square inch gauge along with measurement uncertainty and dead-band = +30/-60 psi.
- TS//COLR $F_{\Delta H}$ limit = 1.65 at HFP.
- TS/COLR F_Q limit = 2.5.
- Maximum core bypass flow = 8.6 percent.

The following tables of ANP-3969P, "Callaway Non-LOCA Summary Report," Revision 2 (attachments 8 (non-proprietary) and 12 (proprietary) of enclosure 1 to the LAR) list the parameters used for the transient analysis of the events:

- Table 3-1 lists the level of SG tube plugging.
- Table 3-2 shows the key component setpoints and capacities.
- Table 3-3 lists the plant operational modes.
- Table 3-4 list the RPS functions and response times (i.e., time delay for the trip breakers open and the rod cluster control assemblies (RCCAs) to start to insert into the core.
- Tables 3.5 lists the RPS trip functions credited in the analysis.
- Table 3-6 lists the engineered safety feature actuation system (ESFAS) setpoints and response times.

- Table 3-7 summarizes the fuel mechanical design parameters.
- Table 3-8 shows the core power distribution parameters $F_{\Delta H}$ and F_Q.
- Table 3-9 shows the key core kinetics parameters and reactivity feedback coefficients supported by the transient analyses.
- Table 3-10 provides the MDNBR design limits of correlations ORFEO-GAIA and ORFEO-NMGRID used in the thermal-hydraulic codes XCOBRA-IIIC and COBRA-FLX.

3.5.4 DNB and FCM Analysis

The key features of the licensee's DNB and FCM analysis are as follows:

- The DNB and FCM analyses are performed in accordance with the TR EMF-2310(P)(A) and their statistical analyses using TR EMF-92-081(P)(A) methodologies.
- The DNB calculation for a mixed core using XCOBRA-IIIC code (TR XN-NF-75-21(P)(A). The application of this methodology is described in TR XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations" (Reference 33). A mixed core penalty of 2 percent is applied to the DNB correlation limit in accordance with the SE included in TR XN-NF-82-21(P)(A).
- DNB calculations performed with the XCOBRA-IIIC code are in accordance with TR XN-NF-75-21(P)(A). The MDNBR calculations are performed using a steady-state XCOBRA-IIIC model with core boundary conditions at the time of MDNBR from the S-RELAP5 transient analyses.
- DNB calculations performed with the COBRA-FLX are in accordance with TR ANP-10311P-A using the same methodology as XCOBRA-IIIC. To be consistent with the methodology used for XCOBRA-IIIC, MDNBR calculations are performed using a steady-state COBRA-FLX model with core boundary conditions at the time of MDNBR from the S-RELAP5 transient analyses.
- The DNB calculations are performed using NRC-approved ORFEO-NMGRID and ORFEO-GAIA DNB correlations. The fuel design parameters for the GAIA assembly are within the applicable range for the ORFEO-NMGRID and ORFEO-GAIA DNB correlations. The GAIA fuel transition operating conditions are within the applicable range of coolant conditions of the ORFEO-NMGIRD and ORFEO-GAIA DNB correlations.
- According to TR EMF-92-081(P)(A), the protection against FCM is expressed as a limit on LHGR allowed in the core. An FCM limit is established for (UO_2) fuel rods such that the FCM is prevented for all fuel rod types.
- For slow evolving transients, power peaking factors are combined to determine peak LHGR (PLHGR), which is compared to a LHGR corresponding to the FCM temperature.

- For fast transients such as uncontrolled RCCA bank withdrawal from HZP that challenge FCM, a hot spot model in the S-RELAP5 code is used with event specific power peaking factors to calculate peak fuel centerline temperature, which is compared to the fuel melt temperature.

3.5.5 Classification of Plant Conditions

In report ANP-3969P section 3.11, consistent with FSAR SP section 15.0.1, the licensee provided classification of FSAR SP Chapter 15 non-LOCA events adopted by the American Nuclear Society (ANS)/American National Standard Institute (ANSI) -N18.2, “Nuclear Safety Criteria for the Design of Stationary PWR Plants, 1973 (Reference 34). ANS/ANSI-N 18-2 places each event into one of four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows:

- ANS Condition I: Normal operation and operational transients
- ANS Condition II: Faults of moderate frequency
- ANS Condition III: Infrequent faults
- ANS Condition IV: Limiting faults.

Report ANP-3969P, table 3-11 summarizes the non-LOCA event classifications and acceptance criteria based on the frequency of occurrence and consequences.

3.5.6 FSAR SP Chapter 15 Events Disposition and Analysis

The licensee’s analysis of the FSAR SP Chapter 15 events based on the proposed transition to a limited number of GAIA fuel assemblies, and the NRC staff evaluation of the licensee’s evaluation is given below.

<u>Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature (FSAR SP 15.1.1)</u>

Report ANP-3969P section 5.1, provides the event description, analysis method, assumptions, and results for the reduction in feedwater temperature event which causes an increase in core power by decreasing the reactor coolant temperature. This event is classified as ANS Condition II event. The following are the key features of the analysis for this event:

- Methodology: S-RELAP5 and XCOBRA-IIIC
- DNB correlation: ORFEO-GAIA
- Sequence of events: report ANP-3969P, table 5-1
- Result graphs: report ANP-3969P, figures 5-1 through 5-4.

<u>Results (Report ANP-3969P, Table 5-2)</u>

Criterion	Result	Acceptable Limit
MDNBR	1.440	1.142 minimum
PLHGR (kW/ft*)	19.5	[[ ]]

* <u>kilowatt/foot</u>

The NRC staff finds that the results are acceptable because they are based on the use of the approved methodology and DNB correlation. In addition, the NRC staff finds the results are acceptable because the calculated MDNBR is greater than the minimum acceptable limit and the calculated PLHGR is less than the maximum acceptable limit.

Feedwater System Malfunctions that Result in an Increase in Feedwater Flow

Report ANP-3969P section 5.2, provides the event description, analysis method, assumptions and results for the addition of excessive feedwater event which causes an increase in core power by decreasing the reactor coolant temperature. This event is classified as an ANS Condition II event. The licensee performed the following analyses at HFP and HZP:

Analysis at HFP

- Methodology: S-RELAP5 and XCOBRA-IIIC
- DNB correlation: ORFEO-GAIA
- Sequence of events: report ANP-3969P, table 5-3
- Result graphs: report ANP-3969P, figures 5-5 through 5-9

Analysis at HZP

- Methodology: S-RELAP5, COBRA-FLX, and ARTEMIS
- DNB correlation: ORFEO-NMGRID
- Sequence of events: report ANP-3969P, table 5-3
- Result graphs: report ANP-3969P, figures 5-10 through 5-15

Results (Report ANP-3969P, Table 5-4)

Case	Criterion	Result	Acceptable Limit
HFP, end-of-cycle (EOC), symmetric	MDNBR	1.927	1.142 minimum
	PLHGR (kW/ft)	17.6	[[ ]]
HFP, EOC, asymmetric	MDNBR	1.806	1.142 minimum
	PLHGR (kW/ft)	Bounded by HFP symmetric case	[[ ]]
HZP, EOC, symmetric	MDNBR	3.299	1.173 minimum
	PLHGR (kW/ft)	11.79	[[ ]]
HZP, EOC, asymmetric	MDNBR	3.680	1.173 minimum
	PLHGR (kW/ft)	13.11	[[ ]]

Based on the use of approved methodologies and DNB correlations, the NRC staff finds the results are acceptable because the calculated MDNBR is greater than the minimum acceptable limit and the calculated PLHGR is less than the maximum acceptable limit.

Excessive Increase in Secondary Steam Flow (FSAR SP Section 15.1.3)

In report ANP-3969P section 5.3, the licensee provides the event description, analysis method, assumptions, and results for the event modeled to be initiated by a 10 percent step increase in steam demand. This increase in steam demand is within the limits which the reactor is designed to accommodate; therefore, a reactor trip is not expected to occur as a result of this event. This event is classified as an ANS Condition II event.

- Methodology: S-RELAP5 and COBRA-IIIC
- DNB correlation: ORFEO-GAIA
- Sequence of events: report ANP-3969P, table 5-5
- Result graphs: report ANP-3969P, figures 5-16 through 5-19

Results (Report ANP-3969P, Table 5-6)

Criterion	**Result**	**Acceptable Limit**
MDNBR	1.733	1.142 minimum
PLHGR (kW/ft)	18.3	**[[ ]]**

Based on the use of the approved methodology and DNB correlation, the NRC staff finds the results are acceptable because the calculated MDNBR is greater than the minimum acceptable limit and the calculated PLHGR is less than the maximum acceptable limit.

Inadvertent Opening of a Steam Generator Relief or Safety Valve (FSAR SP Section 15.1.4)

Report ANP-3969P section 5.4 provides the event description, analysis method, assumptions, and results for this event. This event is classified as an ANS Condition II event.

This event is like the FSAR SP section 15.1.5 event (discussed below) in that it meets the Condition II criteria and is more severe than the FSAR SP section 15.1.4 event. Therefore, the NRC staff finds it acceptable that the licensee did not reanalyze this event for the GAIA VQP.

Steam System Piping Failure

In report ANP-3969P section 5.5, the licensee provides the event description, analysis method, assumptions, and results of the analysis of the post-scram phase of a steam system piping failure or a MSLB event. This event is classified as ANS Condition III/IV event.

Analysis at HFP and HZP

Methodology: S-RELAP5, COBRA-FLX, and ARTEMIS
DNB correlation: ORFEO-NMGRID
Sequence of events: report ANP-3969P, Table 5-7
Result graphs: report ANP-3969P, Figures 5-20 through 5-31.

Results (Report ANP-3969P, Table 5-8)

Case	Criterion	Result	Acceptable Limit
HFP, EOC, offsite power available	MDNBR	3.212	1.173 minimum
	PLHGR (kW/ft)	15.00	[[ ]]
HFP, EOC, loss of offsite power	MDNBR	3.609	1.204 minimum
	PLHGR (kW/ft)	10.50	[[ ]]
HZP, EOC, offsite power available	MDNBR	2.816	1.173 minimum
	PLHGR (kW/ft)	15.32	[[ ]]
HZP, EOC, loss of offsite power	MDNBR	2.322	1.204 minimum
	PLHGR (kW/ft)	14.99	[[ ]]
Mode 3, EOC, minimum safety injection (SI), offsite power available	MDNBR	2.674	1.173 minimum
	PLHGR (kW/ft)	14.40	[[ ]]
Mode 3, EOC, maximum SI, offsite power available	MDNBR	2.505	1.173 minimum
	PLHGR (kW/ft)	16.10	[[ ]]

Based on the use of the approved methodology and DNB correlation, the NRC staff finds the results are acceptable because the calculated MDNBR is greater than the minimum acceptable limit and the calculated PLHGR is less than the maximum acceptable limit.

Steam Line Break with Coincidental RCCA Withdrawal at Power (FSAR SP Section 15.1.5.5)

In report ANP-3969P section 5.6, the licensee states that the event is not applicable to Callaway because the automatic rod control system has been disabled. The NRC staff agrees that this event is not required to be analyzed because the Callaway automatic rod control system has been disabled.

Steam System Piping Failure at Full Power (FSAR SP Section 15.1.5.6)

In report ANP-3969P section 5.7, the licensee provides the event description, analysis method, assumptions, and results of the pre-scram phase of a MSLB event. This event is classified as ANS Condition III/IV event.

- Methodology: S-RELAP5 and XCOBRA-IIIC

- DNB correlation: ORFEO-GAIA
- Sequence of events: report ANP-3969P, table 5-9
- Result graphs: report ANP-3969P, figures 5-32 through 5-40

Results (Report ANP-3969P, Table 5-10)

Criterion	Result	Limit
MDNBR	1.163	1.142 minimum
PLHGR (kW/ft)	21.5	[[ ]]

Based on the use of the approved methodology and DNB correlation, the NRC staff finds the results are acceptable because the calculated MDNBR is greater than the minimum acceptable limit and the calculated PLHGR is less than the maximum acceptable limit.

Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow (FSAR SP Section 15.2.1)

In report ANP-3969P, section 5.8, the licensee stated that there are no steam pressure regulators in Callaway whose failure or malfunction could cause a steam flow transient. The NRC staff agrees that this event is not required to be analyzed because Callaway does not have steam pressure regulators.

Loss of External Electrical Load (FSAR SP Section 15.2.2)

In report ANP-3969P section 5.9, the licensee stated that for a loss of external electrical load without subsequent turbine trip, no direct reactor trip signal would be generated, and the plant would be expected to trip from the RPS if a safety limit is approached. A loss of external load event results in a transient that is bounded by the turbine trip event. The NRC staff finds it acceptable that this event does not require reanalysis for the fuel transition because it is relatively less severe compared to the turbine trip event discussed below.

Turbine Trip (FSAR SP Section 15.2.3)

In report ANP-3969P section 5.10, the licensee stated that in a turbine trip, the reactor trips directly (unless below approximately 50 percent RTP) from a signal derived from the turbine stop emergency trip fluid pressure and turbine stop valves. This event is more limiting than a loss of external load, a loss of condenser vacuum, and other events which result in a turbine trip. No aspect of the GAIA fuel affects the relative severity of the loss of electrical load, loss of condenser vacuum, or other events which result in turbine trip, compared to the turbine trip event.

The key parameters driving the RCS heatup during a turbine trip are the closing speed of the turbine stop or control valves, main feedwater response, the level of SG tube plugging, RCP performance, and the RPS. These parameters are unrelated to the fuel type, therefore the MDNBR is not affected due to GAIA fuel transition during this event.

The licensee stated that the RCS overpressure criteria is not challenged because the parameters affecting the overpressure such as plant configuration, operating parameters, or

RPS or ESFAS are not affected by the fuel type. Global reactivity feedback is not a significant parameter for this event. No aspect of the GAIA fuel affects the power mismatch between the primary and secondary systems.

The NRC staff finds it acceptable that reanalysis is not required for this event because the RCS overpressure criteria is not affected.

Inadvertent Closure of Main Steam Isolation Valves (FSAR SP Section 15.2.4)

In report ANP-3969P section 5.11, the licensee stated that an inadvertent closure of the main steam isolation valves event results in a turbine trip with no credit taken for the turbine bypass system.

The NRC staff finds it acceptable that reanalysis of this event is not required for the GAIA fuel transition because it is less severe compared to the turbine trip event discussed above.

Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip (FSAR SP Section 15.2.5)

In report ANP-3969P section 5.12, the licensee stated that the loss of condenser vacuum prevents steam dump to the condenser. A steam dump to the condenser is not credited in the analysis of a turbine trip event, therefore there is no additional adverse effects result if a turbine trip is caused by a loss of condenser vacuum.

The NRC staff finds it acceptable that GAIA fuel transition does not require reanalysis of this event because it is bounded by the turbine trip event discussed above.

Loss of Nonemergency AC Power to the Plant Auxiliaries (FSAR SP Section 15.2.6)

In report ANP-3969P section 5.13, the licensee stated that "[a] complete loss of nonemergency AC [alternating current] power may result in the loss of all power to the plant auxiliaries, i.e., the RCPs, condensate pumps, etc." In addition to the initial operating conditions, auxiliary system design, and equipment capacities, the consequences of this event would depend on the core decay heat which may vary with the core design. In the supplement dated May 9, 2023, the licensee stated that the GAIA fuel decay heat is bounded by the AOR decay heat models. As part of the reload design process, for the limited number of GAIA fuel assemblies, the licensee evaluated the individual parameters influencing the decay heat and found them acceptable in the reload safety analyses for operating cycles 25 and 27. The licensee stated that future core designs with a larger number of GAIA assemblies would require an evaluation for impact on core decay heat.

The NRC staff finds the licensee's analysis for core decay heat for operating cycle 27 is acceptable because the GAIA fuel decay heat is bounded by the AOR decay heat model. The licensee stated that the core decay heat magnitude will be confirmed for each reload design. The GAIA fuel does not significantly impact any of these decay heat controlling parameters; therefore, this event does not require reanalysis to support the transition to a limited number of GAIA fuel assemblies.

Loss of Normal Feedwater Flow (FSAR SP Section 15.2.7)

In report ANP-3969P section 5.14, the licensee stated that "[a] loss of normal feedwater flow, caused by pump failures, valve malfunctions, or loss of offsite AC power or feedwater control system failure, results in a reduction in the capability of the secondary system to remove the [decay] heat generated in the reactor core." As stated in FSAR SP sections 15.2.6.1 and 15.2.7.1, the safety-related auxiliary feedwater (AFW) system initiates and delivers water to the SGs. The AFW system with its design redundancy ensures that a heat sink is available to remove decay heat from the RCS via the SGs and therefore the pressurizer will not overfill. In the supplement dated May 9, 2023, the licensee stated that the key parameters that effect this event include initial power, initial vessel average temperature, initial pressurizer pressure, pressurizer pressure control using power operated relief valves, pressurizer sprays, pressurizer safety valve (PSV) setpoint, AFW injection flow rate, and operator action time. Since the reactor is tripped well before the SG heat transfer capability is reduced and the AFW system initiates, the RCS variables would not approach a DNB condition.

The NRC staff finds it acceptable that this event does not require reanalysis to support the transition because the GAIA fuel parameters would not significantly affect the above-mentioned key parameters for this event.

Feedwater System Pipe Break (FSAR SP Sections 15.2.8)

Report ANP-3969P section 5.15 provides a description of this event. The feedwater line breaks can be upstream or downstream of the feedwater check valve. The upstream break would be equivalent to a loss of normal feedwater, which is covered in the FSAR SP sections 15.2.6 and 15.2.7 events evaluated above. A major feedwater line break would be a break in a feedwater line large enough to prevent the addition of sufficient feedwater to the SGs to maintain shell-side fluid inventory in the SGs. This would occur if the break were in the piping downstream of the feedwater check valve (i.e., between the check valve and the SG), so that fluid from the SG will also discharge through the break.

In the LAR supplement dated May 9, 2023, the licensee stated that:

> 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 a 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.

A break between the check valve and SG could prevent the subsequent addition of AFW to the affected SG, which could cause RCS heat-up and over pressurization of the RCS or loss of hot leg subcooling due to failure to remove decay heat.

The licensee stated and the NRC staff agrees that the consequences of this event primarily depend on initial operating conditions, plant-related systems and capacities, and decay heat. Therefore, the NRC staff finds it acceptable that this event does not require reanalysis to support the transition because the GAIA fuel parameters would not significantly affect the above-mentioned controlling parameters for this event.

Partial Loss of Forced Reactor Coolant Flow (FSAR SP Section 15.3.1)

In report ANP-3969P section 5.16, the licensee stated that:

> A partial loss of forced reactor coolant flow transient can result from a mechanical or electrical failure in an RCP or from a fault in the power supply to the pump or pumps supplied by an RCP [electrical] bus. If the reactor is at power at the time of the event, the immediate effect of the partial loss of forced reactor coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor does not trip promptly.

The licensee performed a sensitivity study demonstrating that the analysis of FSAR SP section 15.3.2, "Complete Loss of Forced Reactor Coolant Flow," event described below bounds this event.

Based on the licensee's sensitivity study, the NRC staff finds that the FSAR SP 15.3.2 event is more severe and would bound this event because for complete loss of power to all four RCPs, the Condition III criteria are met for the more severe event.

Complete Loss of Forced Reactor Coolant Flow

Report ANP-3969P section 5.17 provides the event description, analysis method, assumptions, and results of a complete loss of forced reactor coolant flow which may result from a simultaneous loss of electrical supplies to all RCPs. This event is classified as an ANS Condition III event.

In report ANP-3969P section 5.17, the licensee stated that

> A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical [power] to all RCPs. If the reactor is at power at the time of the event, the immediate effect of loss of coolant flow is a rapid… coolant heatup. This increase could result in a DNB with subsequent fuel damage if the reactor is not tripped promptly.

- Methodology: S-RELAP5 and XCOBRA-IIIC
- DNB correlation: ORFEO-GAIA
- Sequence of events: Report ANP-3969P table 5-11
- Result graphs: Report ANP-3969P figures 5-41 through 5-44

Results (Report ANP-3969P, Table 5-12)

Criterion	Result	Acceptable Limit
MDNBR	1.592	1.142 minimum
PLHGR (kW/ft)	18.1	[[ ]]

Based on the use of approved methodology and DNB correlation, the NRC staff finds the results are acceptable because the calculated MDNBR is greater than the minimum acceptable limit and the calculated PLHGR is less than the maximum acceptable limit.

Reactor Coolant Pump Shaft Seizure (Locked Rotor) (FSAR SP Section 15.3.3)

Report ANP-3969P section 5.18 provides the event description, analysis method, and results of an event which causes an instantaneous seizure of an RCP rotor. This event is classified as an ANS Condition IV event.

- Methodology: S-RELAP5 and XCOBRA-IIIC
- DNB correlation: ORFEO-GAIA
- Sequence of events: report ANP-3969P table 5-13
- Result graphs: report ANP-3969P figures 5-45 through 5-47

Results (Report ANP-3969P Table 5-14)

Criterion	**Result**	**Acceptable Limit**
MDNBR	1.204	1.142 minimum
PLHGR (kW/ft)	18.1	[[ ]]

Based on the use of the approved methodology and DNB correlation, the NRC staff finds the results are acceptable because the calculated MDNBR is greater than the minimum acceptable limit and the calculated PLHGR is less than the maximum acceptable limit.

Reactor Coolant Pump Shaft Break (FSAR SP Section 15.3.4)

In report ANP-3969P section 5.19, the licensee stated that:

> The event is postulated as an instantaneous failure of an RCP shaft. Flow through the affected reactor coolant loop is rapidly reduced, though the initial rate of reduction of coolant flow is greater for the RCP rotor seizure (locked rotor) event. A free spinning pump impeller is assumed in the faulted RCP for the locked rotor analysis ([FSAR SP] Section [15.3.3]) to address higher reverse flows that are characteristic of this event.

The licensee stated that reanalysis is not required for this event for the GAIA VQP.

The NRC staff agrees that reanalysis is not required for the GAIA VQP because the consequences of this event are bounded by the analytical assumptions made for the FSAR SP section 15.3.3 analysis.

Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low-Power Startup Condition (FSAR SP Section 15.4.1)

Report ANP-3969P section 5.20, provides the event description, analysis method, and results of a RCCA withdrawal event initiated by an addition of reactivity to the reactor core caused by the uncontrolled withdrawal of a sequential pair of RCCA banks resulting in a core power excursion. This event is classified as an ANS Condition II event.

- Methodology: S-RELAP5 and XCOBRA-IIIC
- DNB correlation: ORFEO-GAIA
- Sequence of events: report ANP-3969P table 5-15
- Result graphs: report ANP-3969P figures 5-48 through 5-50.

Results (Report ANP-3969P Table 5-16):

Criterion	Result	Acceptable Limit
MDNBR	1.371	1.142 minimum
Peak fuel centerline temperature (ºF)	2862.4	[[ ]]

Based on the use of the approved methodology and DNB correlation, the NRC staff finds the results are acceptable because the calculated MDNBR is greater than the minimum acceptable limit and the calculated PLHGR is less than the maximum acceptable limit.

Uncontrolled RCCA Bank Withdrawal at Power (FSAR SP Section 15.4.2)

Report ANP-3969P section 5.21, provides the event description, analysis method, assumptions, and results of an uncontrolled RCCA bank withdrawal at-power which results in an increase in the core heat flux. This event is classified as an ANS Condition II event.

Analysis at 100, 60, and 10 Percent RTP

- Methodology: S-RELAP5 and XCOBRA-IIIC
- DNB correlation: ORFEO-GAIA
- Sequence of events: report ANP-3969P table 5-17
- Result graphs: report ANP-3969P figures 5-51 through 5-61.

Results (Report ANP-3969P Table 5-18)

Limiting Case	Criterion	Result (Note 1)	Acceptable Limit
100% RTP Beginning-of-Cycle (BOC)	MDNBR	1,301	1.142 minimum
	PLHGR (kW/ft)	Note 2	**[[ ]]**
100% RTP EOC	MDNBR	1.316	1.142 minimum
	PLHGR (kW/ft)	Note 2	**[[ ]]**
60% RTP BOC	MDNBR	1,245	1.142 minimum
	PLHGR (kW/ft)	Note 2	**[[ ]]**
60% RTP EOC	MDNBR	1.348	1.142 minimum
	PLHGR (kW/ft)	Note 2	**[[ ]]**
10% RTP BOC	MDNBR	**[[ ]]**, 1.228 (Note 1)	1.142 minimum
	PLHGR (kW/ft)	21.8	**[[ ]]**
10% RTP EOC	MDNBR	1.204	1.142 minimum
	PLHGR (kW/ft)	Note 2	**[[ ]]**

Notes:
1. As stated in licensee's supplement dated June 21, 2023, for 10 percent RTP BOC case, the MDNBR of **[[ ]]** was calculated using deterministic method and MDNBR of 1.228 was calculated using statistical method. All other MDNBRs are calculated using deterministic method.
2. Bounded by the value for 10 percent RTP BOC since the overall peak transient kinetic power (supplement dated June 21, 2023, response to item 3) for the 10 percent RTP BOC case is higher than the other cases.

The overall limiting case for both MDNBR and PLHGR is the 10 percent RTP with BOC kinetics. Based on the use of approved methodology and DNB correlation, the NRC staff finds the results are acceptable because the calculated MDNBR is greater than the minimum acceptable limit and the calculated PLHGR is less than the maximum acceptable limit.

RCCA Misoperation (System Malfunction or Operator Error) (FSAR SP Section 15.4.3)

ANP-3969P section 5.22, provides the event descriptions, analysis method, assumptions, and results of the RCCA misoperation events which include the following: (a) one or more dropped

RCCAs within the same group, (b) a dropped RCCA bank, (c) statically misaligned RCCA, and (d) withdrawal of a single RCCA.

Events (a), (b), and (c) are classified as ANS Condition II events, and (d) is classified as an ANS Condition III event. For event (b), the return to power will be less than event (a) due to the greater worth of the entire bank. The consequences of (c) are bounded by event (a) analysis.

- Methodology: S-RELAP5 and XCOBRA-IIIC
- DNB correlation: ORFEO-GAIA
- Sequence of events: report ANP-3969P table 5-19
- Result graphs: report ANP-3969P figures 5-62 through 5-64.

Results for RCCA Drop (Report ANP-3969P Table 5-20)

Criterion	**Result**	**Acceptable Limit**
MDNBR	1.146	1.142 minimum
PLHGR (kW/ft)	20.8	**[[ ]]**

Results for Single RCCA Withdrawal (Report ANP-3969P Table 5-21)

Criterion	**Result**	**Acceptable Limit**
MDNBR	1.202	1.142 minimum
PLHGR (kW/ft)	18.4	**[[ ]]**

Based on the use of the approved methodology and DNB correlation, the NRC staff finds the results are acceptable because the calculated MDNBR is greater than the minimum acceptable limit and the calculated PLHGR is less than the maximum acceptable limit .

Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature (FSAR SP Section 15.4.4)

In report ANP-3969P section 5.23, the licensee stated that [t]he plant Technical Specifications do not permit operation in Modes 1 and 2 with fewer than four reactor coolant loops operating. Therefore, the NRC staff finds that no analysis is required for this event for the GAIA fuel VQP.

A Malfunction of Failure of the Flow Controller in a Boiling Water Reactor Loop that Results in an Increased Reactor Coolant Flow Rate (FSAR SP Section 15.4.5

The NRC staff finds that this event is not applicable to Callaway because it is a boiling-water reactor (BWR) event.

Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant (FSAR SP Section 15.4.6)

In report ANP-3969P section 5.25, the licensee provided the event description, analysis method, assumptions, and results. This event is classified as an ANS Condition II event.

The licensee stated that "[a] boron dilution event is caused by a malfunction or inadvertent operation of the chemical and volume control system (CVCS) that results in the reduction of the boron concentration in the RCS. The reduction of the boron concentration causes a positive reactivity insertion which could increase core power and challenge the DNBR and FCM." The licensee justified that this event reanalysis is not required in Mode 1 operation because the results of the event are bounded by the range of reactivity insertion rates considered for the uncontrolled bank withdrawal event at power. For Modes 2 through 5, the licensee analyzed the event "to assess the adequacy of allowed operator response times (Mode 2) or the boron dilution mitigation system (BDMS) (Modes 3, 4, and 5) to prevent core re-criticality. The time required for a return to power is based upon the dilution flow rate, the mixing volume, temperature, pressure, the initial boron concentration, and initial shutdown margin." The licensee stated that "Modes 2 through 5 do not involve system transient calculations but the time to re-criticality is analyzed for the [GAIA] VQP." For Mode 6, the licensee stated that the analysis is not required because an uncontrolled boron dilution event will not occur during this mode. The licensee stated that "[i]nadvertent dilution via unborated water sources is prevented by administrative controls described in the plant Technical Specifications… Section 3.9.2 which isolates the RCS from potential sources of unborated water."

<u>Results (Report ANP-3969P Table 5-22)</u>

Mode	**Critical Time – BDMS Initiation Time (minutes)**	**Delay Allowance (minutes)**	**Margin (minutes)**
5	19.5	6.6	12.9
4	13.2	4.5	8.7
3	12.6	4.5	8.1
Mode	**Critical Time (minutes)**	**Response Time (minutes)**	**Margin (minutes)**
2	41.5	40	1.5

The NRC staff finds the results are acceptable because the BDMS initiation times for Modes 3, 4, and 5 are bounded by the delay allowance times and for Mode 2, the response time is bounded by the critical time. For Mode 1, the NRC staff finds it acceptable that this event reanalysis is not required because the licensee justified that the results of the event in this mode are bounded by the range of reactivity insertion rates considered for the uncontrolled bank withdrawal event at power (FSAR SP 15.4.2). For Mode 6, the NRC staff finds it acceptable that this event analysis is not required because an uncontrolled boron dilution event will not occur as it is prevented by administrative controls described in the TS.

<u>Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (FSAR SP Section15.4.7)</u>

FSAR SP section 15.4.7 provides the event description, analysis, and conclusion. This event is classified as an ANS Condition III event. In section 2.2 of enclosure 1 to the letter dated October 12, 2022, the licensee stated that the FSAR SP Section 15.4.7 event description and conclusions bound the transition to a limited number of GAIA fuel assemblies. After consulting with the current fuel manufacturer (Westinghouse), the licensee concluded that the ability to detect significant power distribution anomalies due to fuel assembly loading errors before exceeding the SAFDLs remain valid for the GAIA fuel. The licensee stated and the NRC staff agrees that the current licensing basis analysis is independent of the presence of the GAIA fuel assemblies and therefore the FSAR SP section 15.4.7 conclusions on this event are not affected.

Spectrum of Rod Cluster Control Assembly Ejection Accidents (FSAR SP Section 15.4.8)

Report ANP-3947P section 6.1.1 provides the description of the accident, acceptance criteria, analysis method, and conclusions. This event is classified as an ANS Condition IV event. The licensee analyzed this event using the NRC-approved ARCADIA Rod Ejection Accident (AREA) methodology using the GALILEO fuel rod thermal mechanical methodology as against the acceptance criteria given in RG 1.236. Section 3.10 of this SE provides the results of the analysis of this event.

Inadvertent Operation of the Emergency Core Cooling System During Power Operation (FSAR SP Section 15.5.1)

Report ANP-3969P section 5.28 provides a description of this event. The licensee stated that an operator error or a false electrical signal could cause this event. Following the false signal, ECCS charging pumps would start and inject borated water into the cold leg of each loop of the reactor. The SI pumps would also start automatically but provide no flow if the RCS is at normal pressure. In case the reactor does not immediately trip on a spurious SI signal, it will experience a negative reactivity excursion due to the injection of the borated water causing a decrease in core power and core temperature and subsequent decrease in the RCS pressure. The overall result would be an increase in the margin to DNB. Subsequently the reactor trip will occur on low pressurizer pressure or would be manually tripped. The NRC staff finds it acceptable that reanalysis of this event is not required for the GAIA VQP because the ECCS flow rate or negative reactivity insertion by the ECCS would not affect the SAFDLs.

The licensee also stated that the RCS and main steam system pressure will remain below 110 percent of their design pressure in this event. The NRC staff finds it acceptable that the RCS and main steam pressure would remain below 110 percent of the design pressure because the GAIA fuel does not significantly affect the controlling parameters (i.e., initial conditions, system setpoints and capacities, or operator action times) for this aspect of the event, and therefore reanalysis of this event is not required for the GAIA VQP.

Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory (FSAR SP Section15.5.2)

Report ANP-3969P section 5.29 provides a description of this event. The licensee stated that an increase in reactor coolant inventory could have the injected fluid under any of the following three conditions: (a) unborated water, (b) borated water with a higher boron concentration than the RCS, and (c) water with a boron concentration the same as the RCS boron concentration. Conditions (a) and (b) are the FSAR SP sections15.4.6 and 15.5.1 events respectively evaluated above in this SE. The condition (c) event would not be a reactivity event and the core power and RCS temperature would change insignificantly because the CVCS malfunction is not causing changes in core reactivity. The NRC staff finds it acceptable that condition (c) would not challenge the SAFDLs because it does not affect the core reactivity and therefore does not require reanalysis for the GAIA fuel VQP.

A Number of BWR Transients (FSAR SP Section 15.5.3)

The NRC staff finds that these events are not applicable to Callaway as they are BWR events.

Inadvertent Opening of a Pressurizer Safety or Relief Valve (FSAR SP Section 15.6.1)

Report ANP-3969P section 5.31 provides the event description, analysis method, and results of an accidental depressurization of the RCS which could occur from an inadvertent opening of a pressurizer relief or safety valve. This event is classified as ANS Condition II/III event. An inadvertent opening of a pressurizer relief valve is classified as an ANS Condition II event, and the failure of a PSV is classified as an ANS Condition III event.

- Methodology: S-RELAP5 and XCOBRA-IIIC
- DNB correlation: ORFEO-GAIA
- Sequence of events: report ANP-3969P table 5-23
- Result graphs: report ANP-3969P figures 5-65 through 5-67

Results (Report ANP-3969P Table 5-24):

Criterion	**Result**	**Acceptable Limit**
MDNBR	1.463	1.142 minimum
PLHGR (kW/ft)	18.3	**[[ ]]**

Based on the use of approved methodology and DNB correlation, the NRC staff finds the results are acceptable because the calculated MDNBR is greater than the minimum acceptable limit and the calculated PLHGR is less than the maximum acceptable limit .

Break in Instrument Line or Other Lines from Reactor Coolant Pressure Boundary that Penetrate Containment (FSAR SP Section 15.6.2)

In the current licensing basis, only a radiological dose analysis is provided; transient response analysis is not required for this event. The radiological dose analysis is not within the scope of this LAR.

Steam Generator Tube Failure (FSAR SP Section 15.6.3)

Report ANP-3969P section 5.33 describes this event as a complete severance of a single SG tube assumed to occur at HFP with the RCS contaminated with fission products corresponding to continuous operation with a limited number of defective fuel rods. The licensee stated that the DNBR response due to the depressurization of the RCS from the ruptured tube is less severe than the FSAR SP Section 15.6.1 event. The licensee did not perform DNB analysis for this event. The NRC staff agrees that reanalysis is not required for this event for the GAIA VQP because it is bounded by the FSAR SP Section 15.6.1 event.

The licensee stated that the “AOR contains a system transient response analysis of this event for SG overfill and input to the radiological dose analysis. The consequences of this event primarily depend on the break flow rate, secondary side relief setpoints and capacity, charging and safety injection flow rates, and operator actions." The licensee stated and NRC staff agrees that the GAIA fuel transition does not impact any of these controlling parameters, therefore its reanalysis is not required for the GAIA VQP.

Spectrum of Boiling Water Reactor Steam System Piping Failures Outside of Containment (FSAR SP Section 15.6.4)

The NRC staff finds that this event is not applicable to Callaway because these are BWR events.

Loss-of-Coolant Accidents Resulting from a Spectrum of Postulation Piping Breaks Within the Reactor Coolant Pressure Boundary

The SBLOCA and LBLOCA events are evaluated in sections 3.2 and 3.3, respectively, of this SE.

A Number of BWR Transients (FSAR SP Section 15.6.6)

The NRC staff finds that these events are not applicable to Callaway because they are BWR events.

Radioactive Release from a Subsystem or Component (FSAR SP Section 15.7)

Report ANP-3969P section 5.37 describes that this event can be caused by radioactive gas waste system leak or failure, radioactive liquid waste system leak or failure, radioactive release due to liquid tank failures, and a fuel handling accident. The NRC staff finds it acceptable that the assessment of radiological doses is out of scope of this LAR.

Anticipated Transient Without Scram (FSAR SP Section 15.8)

In report ANP-3969P section 5.38, the licensee stated and the NRC staff agrees that "[t]he effects of an anticipated transient without scram are not considered as part of the design basis for transients analyzed in [FSAR SP] Chapter 15."

3.5.7 Compliance with NRC Staff Imposed L&Cs

The licensee's dispositions and NRC staff evaluation for the L&Cs identified in the NRC staff SEs for the TRs used in the non-LOCA events analysis is given below.

L&Cs in NRC staff SE for EMF-2310(P)(A), Revision 1

As discussed below, the NRC staff evaluated the licensee's compliance with each of the L&Cs provided in the SE for EMF-2310(P)(A), Revision 1, and finds that the L&C are satisfied.

(1) For analyses during power operation, the initial power level is rated output (licensed core thermal power) plus an allowance of 2 percent, or justified amount, to account for power-measurement uncertainty.

The NRC staff finds L&C (1) is satisfied because section 3.1 of report ANP-3969P states the power measurement uncertainty of ±2 percent of the RTP.

(2) The boron dilution is assumed to occur at the maximum possible rate.

The NRC staff finds L&C (2) is satisfied because as stated in the licensee's supplement dated May 9, 2023, the "CVCS dilution flow is based on conservative values provided in the FSAR SP section 15.4."

(3) 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.

In the supplement dated May 9, 2023, the licensee stated the following:

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Based on the above statement, the NRC staff considers L&C (3) is satisfied because the licensee used conservative core burnup at BOC, boron concentration, and conservative input parameters in using the EMF-2310(P)(A) methodology for the non-LOCA event analysis.

(4) All fuel assemblies are installed in the core.

The NRC staff finds L&C (4) is satisfied because the licensee performed all neutronics calculations based on a full core.

(5) A conservatively low value is assumed for the reactor coolant volume.

NRC finds L&C (5) is satisfied because the analysis is based on the conservatively minimum RCS volume provided in the FSAR SP section 15.4.

(6) For analyses during refueling, all control rods are withdrawn from the core.

The licensee did not analyze refueling (Mode 6) for the VQP. The NRC staff finds it acceptable because during refueling operations all control rods are withdrawn and according to Callaway TS 3.9.1, "Boron Concentration," the "[b]oron concentrations of all filled portions of the Reactor Coolant System and the refueling pool that have direct access to the reactor vessel, shall be maintained sufficient to ensure that the more restrictive of the following reactivity conditions is met: k_{eff} **<** 0.95 or boron concentration of > 2000 ppm." L&C (6) is satisfied.

(7) 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.

In the May 9, 2023, LAR supplement, the licensee stated that the analysis for power operation (Mode 1), the results of the boron dilution event are bounded by the range of reactivity insertion rates considered for the FSAR SP section15.4.2 event analysis. Based on the licensee's statement, the NRC finds L&C (7) is satisfied.

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

In the May 9, 2023, LAR supplement the licensee stated:

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Based on the above statement, the NRC staff finds L&C (8) is satisfied because the licensee assumed conservatively high reactivity addition rate considering the effect of increasing boron worth with dilution.

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

L&C (9) applies to power operation (Mode 1) FSAR SP 15.4.2 event analysis. The NRC staff finds this L&C is satisfied because the reactor trip response time assumed is conservatively maximized by including delay for the trip signal actuation and scram system holding coil release.

<u>L&C in NRC Staff SE for XN-NF-82-21(P)(A), Revision 1</u>

As discussed below, the NRC staff evaluated the licensee's compliance with the L&C provided in the SE for XN-NF-82-21(P)(A), Revision 1, and finds that the L&C is satisfied.

> An adjustment of 2% on the minimum DNBR must be included for mixed cores containing hydraulically different fuel assemblies.

Report ANP-3969P, Section 3.9 states:

> A mixed core penalty of 2% is applied to the CHF [DNB] correlation limit in accordance with Reference 4 for transition analysis performed with

XCOBRA-IIIC. The mixed core penalty is only required for transition cycles containing hydraulically dissimilar fuel assembly types.

Based on the above statement the NRC staff finds this L&C satisfied.

L&Cs in NRC Staff SE for EMF-92-081(P)(A), Revision 1

As discussed below, the NRC staff evaluated the licensee's compliance with each of the L&Cs provided in the SE for EMF-92-081(P)(A), Revision 1, and finds that the L&C are satisfied.

(1) The methodology includes a statistical treatment of specific variables in the analysis; therefore, if additional variables are treated statistically SPC [Siemens Power Corporation] 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.

In the May 9, 2023, supplement the licensee stated to address L&C (1) that "no additional variables to those explicitly mentioned in EMF-92-081(P)(A) 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. **[[**

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Based on the above statement, the NRC staff considers L&C (1) is satisfied because no additional variables other than explicitly mentioned in EMF-92-081(P)(A) are treated statistically. **[[**

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(2) The steam generator safety valve 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.

In the May 9, 2023, supplement, the licensee stated to address L&C (2) that "[t]his restriction only applies to the OTΔT [overtemperature delta temperature] verification analysis." which requires the "[r]elevant inputs should correspond to the steam generator plugging level being analyzed to meet this condition. Due to lack of pressure vs. [versus] power data for 5% SGTP [SG tube plugging] (the VQP cycle plugging level), the licensee's OTΔT analysis used a conservative value of 10%."

Based on the above statement, the NRC staff considers L&C (2) is satisfied because the licensee OTΔT analysis is conservative.

L&Cs in NRC Staff SE for TR ANP-10341P-A, Revision 0.

As discussed below, the NRC staff evaluated the licensee's compliance with each of the L&Cs provided in the SE for TR ANP-10341P-A, Revision 0, and finds that the L&C are satisfied.

<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.

In the supplement dated May 9, 2023, the licensee stated that "[t]he subchannel TH [thermal-hydraulic] code runs were verified to show subcooled coolant conditions at the first axial node." The NRC staff, therefore, finds condition (1) is satisfied.

(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.

In the supplement dated May 9, 2023, the licensee stated that "[t]he DNB calculations utilizing the ORFEO-NMGRID correlation were verified to remain far removed from the **[[ ]]** subregion." The NRC staff, therefore, finds condition (2) is satisfied.

(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 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 [DNB] 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.

In the supplement dated May 9, 2023, the licensee stated that "[t]he DNB analyses with low margin to the design limits were verified to have equilibrium qualities greater than 0.1." The NRC staff, therefore, finds condition (3) is satisfied.

<u>Limitations</u>

(1) ORFEO-GAIA is approved for use in predicting the CHF [DNB] downstream of GAIA and IGM 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.

In the supplement dated May 9, 2023, the licensee stated that "[t]he 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." The NRC staff, therefore, finds limitation (1) is satisfied.

(2) ORFEO-NMGRID is approved for use in predicting the F [DNB] downstream of W [Westinghouse] 17x17 HMP non-mixing grids and GAIA and IGM 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.

In the supplement dated May 9, 2023, the licensee stated that "[t]he 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." The NRC staff, therefore, finds limitation (2) is satisfied.

<u>L&Cs in NRC Staff SE for TR BAW-10231P-A Revision 1</u>

As discussed below, the NRC staff evaluated the licensee's compliance with each of the L&Cs provided in the SE for TR BAW-10231P-A Revision 1 and finds that the L&C are satisfied.

> 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].

The NRC staff finds this L&C is satisfied because the licensee stated that Callaway does not use MOX fuel and is not licensed for its use in the core. However, besides MOX fuel licensing, as concluded in the NRC staff SE on TR BAW-10231P-A, the COPERNIC code is acceptable for fuel licensing applications up to a rod average burnup of 62 GWD/MTU.

<u>L&Cs in NRC Staff SE for TR XN-NF-75-21(P)(A), Revision 2:</u>

As discussed below, the NRC staff evaluated the licensee's compliance with each of the L&Cs provided in the SE for TR XN-NF-75-21(P)(A), Revision 2, and finds that the L&C are satisfied.

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

The NRC finds L&C (1) is satisfied based on the licensee's statement in the LAR supplement dated May 9, 2023, which states:

> 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 [DNB] correlation is restricted to homogeneous models for two-phase flow.

The licensee did not use the XNB DNB correlation, therefore, the NRC staff finds L&C (2) is satisfied.

(3) XCOBRA-IIIC code is acceptable for homogeneous models for those options pertaining to PWR reactors in conjunction with the XNB [DNB] correlation.

The licensee did not use the XNB DNB correlation, therefore, the NRC staff finds L&C (3) is satisfied.

(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.

As stated in report ANP-3969P section 3.9.1 the licensee used the XCOBRA-III in the “snapshot mode”, therefore, the NRC staff finds L&C (4) is satisfied.

<u>L&Cs in NRC Staff SE for TR 10297P-A, Revision 0</u>

The NRC staff notes that the L&Cs provided in the SE for TR 10297P-A, Revision 0, are the same as those listed below for ANP-10297P-A, Revision 0, Supplement 1PA, with the exception of L&C (3) which was removed by the SE for ANP-10297P-A, Revision 0, Supplement 1PA. The staff’s evaluation of these L&Cs is provided below under the evaluation of L&Cs for ANP-10297P-A, Revision 0, Supplement 1PA. L&C (3) was not reviewed because it was removed.

(1) The range applicability of the ARCADIA® code system Methodology is restricted to the fuel data provided in the TR, unless additional analysis and benchmarking is conducted to validate the ARCADIA® code system to a fuel type not mentioned in the TR (Reference 1).

This L&C is evaluated below in the evaluation of L&C (1) in SE for TR ANP-10297P-A, Revision 0, Supplement 1PA.

(2) The benchmarks provided in the TR (Reference 1) include uncertainty verification for plants that use moveable incore, Rh fixed incore, and Aeroball incore detectors. AREVA will continue to monitor its methods with respect to current cycle designs for its licensing applications. Prior to licensing a new contract, AREVA will evaluate at least three cycles of data relative to these criteria prior to licensing the first cycle with AREVA fuel with ARCADIA®. In addition, application of the ARCADIA® code system to a new uncertainty measurement system(s) would require review and approval by the NRC staff prior to implementation. This includes verification of their measurement uncertainties and/or calculation uncertainties by using the appropriate method presented in Section 12 of the TR.

This L&C is evaluated below in the evaluation of L&C (2) in the SE for TR ANP-10297P-A, Revision 0, Supplement 1PA.

(3) The ARCADIA® code system is limited to fuel types with non-Inconel grids unless additional verification of uncertainties is conducted to account for any peaking biases due to grid type or other plant effects. Verification of uncertainties must be quantified and accounted for in the uncertainties and/or peaking allowances in the licensing calculations on plant specific basis.

This L&C is evaluated below in the evaluation of L&C (3) in SE for TR ANP-10297P-A, Revision 0, Supplement 1PA.

(4) For any changes made to the stand-alone version of COBRA-FLX that is implemented in the ARCADIA® code system (the COBRA-FLX module), AREVA will revalidate the ARCADIA® code system output using measured data from multiple plants and cycles.

This L&C is evaluated below in the evaluation of L&C (4) in SE for TR ANP-10297P-A, Revision 0, Supplement 1PA.

<u>L&Cs in SE for TR ANP-10297P-A, Revision 0, Supplement 1PA.</u>

As discussed below, the NRC staff evaluated the licensee's compliance with each of the L&Cs provided in the SE for ANP-10297P-A, Revision 0, Supplement 1PA, and finds that the L&Cs are satisfied.

(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.)

In the LAR supplement dated May 9, 2023, the licensee stated that no new materials or geometries are added other than already present in the TR ANP-10297, Revision 0, Supplement 1(P)(A). The NRC staff therefore finds it acceptable that no new benchmarks are required, and L&C (1) is satisfied.

(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 1PA).

In the LAR supplement dated May 9, 2023, the licensee stated that Callaway falls within the benchmarks provided in the TR and the uncertainty analysis remains applicable. However, the licensee performed benchmarks that include more than three cycles and validated the uncertainty analysis. The analysis confirmed that the peaking uncertainties remain bounded by the values in the TR. Based on the above statement, the NRC staff finds L&C (2) is satisfied.

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

This L&C is not applicable, so NRC staff finds L&C (3) is satisfied.

(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).

The multi-cycle benchmarking mentioned in the evaluation of L&C (1) and comparison to the operating data performed cover the requirement of this L&C. The NRC staff finds L&C (4) is satisfied because the benchmarks provide confirmation that the COBRA-FLX code used in ARTEMIS™ is functioning as expected and all generated data remain consistent with the NRC-approved TR.

(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 TR -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. ...

In the LAR supplement dated May 9, 2023, the licensee stated that ARTEMIS™ is not used as part of an EM in the Callaway VQP analysis except for the RCCA ejection analysis. Therefore, this L&C is only applicable to the SRP 15.4.8, "Spectrum of Rod Ejection Accidents (PWR)," event analysis. Use of ARTEMIS as the EM in the RCCA ejection analysis is covered in the TR ANP-10338(P)(A), "AREA™ – ARCADIA® Rod Ejection Accident" (Reference 35), analyzed by the licensee in report ANP-3947P, and evaluated in section 3.10 of this SE. Based on the NRC staff evaluation presented in section 3.10 of this SE, the NRC staff finds L&C (5) is satisfied.

(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 TR 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.

In the LAR supplement dated May 9, 2023, the licensee stated that for each new release of the ARCADIA code system, a review of the changes in the codes is performed. The changes could fall into the following three categories:

- changes allowed by the TR,
- changes that require discussion with the NRC to determine if they need additional review,
- changes that cannot be used until a supplement implementing the change has been approved by the NRC.

The NRC staff finds L&C (6) is satisfied because the licensee documents all changes and provides them to all users of the ARCADIA codes while clearly identifying the features and models not allowed for licensing analyses.

<u>L&Cs in NRC Staff SE for BAW-10240(P)(A)</u>

As discussed below, the NRC staff evaluated the licensee's compliance with each of the L&Cs provided in the SE for BAW-10240(P)(A), and finds that the L&C are satisfied.

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

In the LAR supplement dated May 9, 2023, the licensee stated that the S-RELAP5 code does not calculate corrosion. The NRC finds L&C (1) is satisfied because the corrosion limit is not affected using S-RELAP5.

(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.

In the LAR supplement dated May 9, 2023, the licensee stated that conditions of other methods or TRs are checked in their respective sections of the attachments to this LAR. The NRC staff finds L&C (2) is satisfied.

(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-10240(P) or Reference 2 [BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel" (Reference 36)] was performed.

In the LAR supplement dated May 9, 2023, the licensee stated that the analyses presented in report ANP-3969P are within the range of applicability for M5 as presented in BAW-10240(P)(A). The NRC staff finds L&C (3) is satisfied.

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

As stated in report ANP-3947P section 2.4.3.1, the maximum fuel rod burnup limit of 62 GWd/MTU is not exceeded. The NRC staff finds L&C (4) is satisfied.

L&Cs in NRC Staff SE TR ANP-10311P-A, Revision 1

As discussed below, the NRC staff evaluated the licensee's compliance with each of the L&Cs provided in the SE TR ANP-10311P-A, Revision 1, and finds that the L&C are satisfied.

(1) The fuel rod model in COBRA-FLX and the rewetting model for post-CHF [DNB] 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 [ANP-10311P]) and are summarized as the following:

a) water properties (IAPWS-IF97)

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

As stated in report ANP-3969P section 3.9.1, the COBRA-FLX model development guidance prescribes the use of these NRC-approved models. In a supplement dated May 9, 2023, the licensee stated that these approved models "are set by default and are the only allowed options." The licensee also stated that "[t]he code will terminate with an error message if the user attempts to over-ride them to an unapproved model." Therefore, the NRC staff finds L&C (1) is satisfied.

(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-[DNB] heat transfer shall not be used for safety related analysis and are specifically excluded from this review.

In the supplement dated May 9, 2023, the licensee stated that "[n]o post-[DNB] calculations utilizing the rewetting model, or the COBRA-FLX internal fuel rod model were used." Therefore, the NRC staff finds L&C (2) is satisfied.

L&Cs in NRC Staff SE for TR XN-NF-82-06(P)(A), Revision 1 & Supplements 2, 4, and 5 (Reference 37)

> If a plant depressurization accident were to occur involving ENC [Exxon Nuclear Company, Inc] fuel at extended burnup levels, the licensee must address the extent of possible hydride reorientation in their fuel cladding before further irradiation of this fuel is allowed, see Section 2.0(h). This requirement is only in effect following a plant depressurization accident.

This L&C is not applicable to the analysis presented in report ANP-3969P. The licensee used the TR XN-NF-82-06(P)(A), Supplement 5 method for analyzing the effects of DNB propagation. Report ANP-3969P, section 3.9.1 states, in part, that no restrictions, limitations, and/or conditions are identified in the SE for Supplement 5 of this TR relative to DNB propagation. Therefore, the NRC staff finds this L&C is satisfied.

3.5.8 Conclusions

Based on the above evaluation of the licensee's information presented in the LAR and the supplements dated May 9, and June 21, 2023, the NRC staff conclusions are as follows:

- The nodalization scheme used for the analysis to support the GAIA fuel transition is specific to Callaway and is consistent with the EMF-2310(P)(A) methodology.
- The parameters and equipment states in the analyses are conservatively chosen.
- The biasing and assumptions for key input parameters are consistent with or conservative to the approved TR EMF-2310(P)(A) methodology.
- The S-RELAP5 code assessments in TR EMF-2310(P)(A) methodology validated the ability of the code to predict the response of the primary and secondary systems to the non-LOCA events. No additional model sensitivity studies are needed for this application.
- The properties from the COPERNIC code are developed for BOC and EOC conditions in accordance with TR EMF-2310(P)(A). The COPERNIC fuel properties and gap coefficients are conservatively implemented in the S-RELAP5 model as approved in TR EMF-2310(P)(A).

- Only safety grade equipment is credited to mitigate the consequences of events.

- The setpoints and response times modeled in the transient analyses are conservatively applied to provide bounding simulations of the plant response.

- To the extent that the RPS and ESFAS are credited in the event analyses, the setpoints have been verified to adequately protect plant operation with the GAIA fuel.

- The current TS/COLR $F_{\Delta H}$ and F_Q limits are supported by the analyses.

- Depending on the event-specific characteristics (e.g., RCS heat-up or cooldown), for conservative analysis the licensee used either maximum or minimum reactivity coefficient values.

- Moderator reactivity feedback is based on the TS/COLR most-negative moderator temperature coefficient (MTC) limit. Therefore, the analysis supports the current TS/COLR limits on MTC.

- The doppler reactivity coefficients include biases according to the approved EMF-2310(P)(A) methodology with additional conservatism to bound potential cycle-to-cycle changes.

- The time delay for the trip breakers to open and the RCCA to start to insert into the core includes the time required to process the trip signal and for the magnetic flux of the RCCA holding coils to decay sufficiently to release the RCCAs.

- The maximum TS time for the RCCAs to reach the entrance of the guide tube dashpot is 2.7 seconds.

- For events initiated from HFP conditions, a conservative minimum HFP scram worth is used which accounts for the most reactive RCCA being fully withdrawn.

- For events initiated from HZP and part-power conditions, the scram worth is set to the TS/COLR minimum shutdown margin requirement.

- The shutdown margin requirements will be verified for each reload cycle.

The NRC staff finds the analysis and results of the non-LOCA events for the transition to a limited number of GAIA fuel assemblies is acceptable.

Based on the above technical conclusions, the NRC staff finds the following 10 CFR Part 50 Appendix A, GDC requirements are satisfied: GDCs 10, 11, 12, 20, 25, 26 and 27. Accordingly, based on the technical and regulatory conclusions described above, the NRC staff finds the analysis and results of the non-LOCA events for the transition to a limited number of GAIA fuel assemblies is acceptable.

3.6 Mechanical Design of GAIA Fuel Assembly

Chapter 2 of report ANP-3947P provides a summary of the mechanical design of Framatome GAIA fuel design that is intended for batch implementation at Callaway and its compatibility with the coresident fuel during the transition from mixed-fuel type core configurations to cores with only Framatome GAIA fuel.

The NRC staff reviewed the mechanical design of Framatome GAIA assembly design for Callaway as per NRC staff approved ANP-10342P-A Revision 0, “GAIA Fuel Assembly Mechanical Design.” Framatome performed a VQP in support of licensing the GAIA fuel design and related Framatome evaluation methodologies at Callaway for the GAIA fuel transition. The VQP establishes the applicability of Framatome’s reload licensing methodology for the use of GAIA at the Callaway station including the mechanical design methods in ANP-10342P-A Revision 0. The NRC staff determined that the transition to GAIA fuel design at Callaway has utilized the mechanical design aspects of GAIA fuel design and provides assurance the plant licensing basis will be met for the Callaway operation because the included analyses were found to be in accordance with those described in ANP-10342P-A Revision 0.

3.7 Mechanical Design Evaluations

3.7.1 Fuel Assembly and Fuel Rod Mechanical Design

The mechanical design evaluations are performed using the NRC staff approved design methods and design criteria per TR ANP-10342P-A, Revision 0. The design criteria are consistent with SRP section 4.2. The fuel system design analysis is broadly classified into fuel rod analyses and structural analyses consistent with SRP section 4.2.

The following fuel assembly mechanical evaluations are considered:

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

The following fuel-rod mechanical analyses are considered:

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

Table 2-4 of report ANP-3947 lists a summary of major mechanical design evaluation methods and their references of approved methodologies used in analyses performed for the GAIA assemblies in Callaway plant.

Table 2-6 of report ANP-3947P lists generic criteria and SAFDLs for the fuel rod and fuel assembly, with the section number from the criteria in TR ANP-10342P-A, Revision 0 and corresponding results.

3.7.2 Fuel Assembly Structural Analysis

The fuel assembly structural analysis consists of the normal operating analysis, shipping and handling analysis, and faulted condition analysis. The structural analysis during normal operating condition includes the fuel assembly stress state during startup, steady-state operation, shutdown, and AOOs), and compares it with the criteria established in NRC-approved TR ANP-10342P-A. The shipping and handling analysis evaluates the fuel assembly against handling limits established in TR ANP-10342P-A and shipping load limits established in Framatome shipping specifications in TRs ANP-10342P-A. The faulted condition analysis is performed per TR ANP-10337P-A, Revision 0, “PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations” (Reference 38), and is discussed in TR ANP-10342P-A. The analysis is performed based on structural models obtained by benchmarking to tests performed on prototypical fuel assemblies and components. The tests were conducted on spacer grids and other components of GAIA fuel assembly to determine the main dynamic characteristics of the fuel assembly for the lateral and vertical models. The evaluations address the operating basis earthquake (OBE), the safe shutdown earthquake (SSE), and two branch line breaks (one on the cold leg and one on the hot leg). Each event is evaluated independently with a large set of core row models (reflecting various core loading patterns) in the horizontal direction and models (BOL and EOL) for the vertical direction.

The fuel assembly faulted condition analysis considered NRC IN 2012-09. NRC IN-2012-09 states that the fuel assembly faulted condition analyses should include accounting of both the direct effects of irradiation on spacer grid dynamic characteristics and strength and the indirect effects of spacer grid relaxation on fuel assembly dynamic characteristics. To address the issue identified in IN 2012-09, the licensee tested two sets of fuel assemblies with spacer grids at BOL and simulated EOL in accordance with the TR ANP-10337P-A methodology. The EOL condition includes the co-resident fuel and spacer grid dynamic properties in the core row models. The SSE impacts are combined with the LOCA impacts via square root of sum of squares (SRSS).

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3.7.3 Seismic and LOCA Time History Generation

The faulted analysis solves the non-linear equations of motion in core row models for displacement, velocity, and acceleration at the interfaces between reactor internals and fuel assemblies with explicit boundary conditions and the CASAC code (Reference 38) for acceleration**. [[**

]]

For the lateral seismic cases, Framatome **[[**

]] This process endured a conservative level of seismic and LOCA loading.

Motions of the lower and upper core plates and the core barrel flange are not independent, therefore **[[**

]].

For the lateral LOCA case, time histories are generated as a **[[**

]]

The vertical seismic cases are very similar in approach to **[[**

]].

For the subsequent dynamic loading and component stress analysis, the TR ANP-10337P-A methodology requires the combination of loads from SSE and LOCA events and the combination of lateral and vertical loading. For LOCAs, **[[**

]] with full consideration of impact direction, grid type, and irradiated fuel condition.

3.7.3.1. Implementation of L&Cs of TR ANP-10337P A

The NRC staff imposed several limitations and conditions on the use of TR ANP-10337P-A. The following sections summarize how the licensee implemented the L&Cs and provide the NRC staff's evaluation for the L&Cs identified in the NRC staff SEs for ANP-10337P-A.

(1) Dynamic grid crush tests, must be conducted in accordance with Section 6.1.2.1 of TR ANP-10337P (as amended by RAI 16), and spacer grid behavior must satisfy the requirements in the TR, the key elements of which are:

(a) **[[**

]]

Response (1): The staff finds that *Framatome has acceptably conducted the necessary dynamic grid crush testing to Demonstrate the behavior defined in items a., b., and c. above for the GAIA spacer grids planned for use in the Callaway fuel design.*

(2) For fuel assembly designs where spacer grid applied loads are limited based on allowable grid permanent deformation (as opposed to buckling), the following limits from Table 4-1 of the topical report apply:

(a) For all OBE analyses, allowable spacer grid deformation is limited to design tolerances and **[[ ]]**,

(b) For SSE, LOCA, and combined SSE+LOCA analyses, **[[**

]]

Response (2): *The staff finds that Framatome has defined allowable spacer grid deformation limits that are in accordance with items a. and b. above for the GAIA spacer grids planned for use in the Callaway fuel design.*

(3) The modification or use of the codes CASAC and ANSYS (or other similar industry standard codes) are subject to the following limitations:

(a) CASAC computer code revisions, necessitated by errors discovered in the source code, needed to return the algorithms to those described in TR ANP-10337P… are acceptable,

(b) Changes to CASAC numerical methods to improve code convergence or speed of convergence, transfer of the code to a different computing platform to facilitate utilization, addition of features that support effective code input/output, and changes to details below the level described in TR ANP-10337P would not be considered to constitute a departure from a method of evaluation in the safety analysis. Such changes [can be implemented] without NRC staff review and approval. …

(c) ANSYS or other industry standard codes may be used if they are documented in an auditable manner to meet the quality assurance requirements of

10 CFR Part 50, Appendix B, including the appropriate verification and validation….

Response (3): *For the Callaway faulted analysis, Framatome has used CASAC exclusively. The staff finds that the CASAC versions used are fully consistent with requirements a. and b. above.*

(4) This methodology is limited to applications that are similar to the current operating fleet of PWR reactor and fuel designs. The core geometry should be comparable to the current fleet, in terms of dimensions, dimension tolerances, fuel assembly row lengths, and the gaps between fuel assemblies. Fuel designs should be comparable to the current fleet, in terms of materials, geometry, and dynamic behavior.

Response (4): *The staff finds that the Callaway Unit 1 reactor is part of the "current fleet" of PWR reactors in place at the time of approval of TR ANP-10337P.*

(5) TR ANP-10337P established generic fixed damping values intended to be used for all PWR designs. All applications of this methodology to new fuel assembly designs must consider the continued applicability of the fixed damping values of this methodology. If new materials, new geometry, or new design features of a new fuel assembly design may affect damping, additional testing and/or evaluation to determine appropriate damping values may be required.

Response (5): *This LAR addresses the application of a GAIA fuel design to an existing reactor that is part of the "current fleet" of PWRs. Hence, the staff finds that the application of the generic damping values from TR ANP-10337P falls within the range of intended application acceptable. Furthermore, the application of the damping values indicated in TR ANP-10337P-A to the GAIA fuel design is approved in TR ANP-10342P-A.*

(6) The TR ANP-10337P methodology includes the generation of fuel rod loads but does not provide a means to demonstrate compliance for fuel rod performance under externally applied loads (to applicable acceptance criteria). Applications of this methodology must provide an acceptable demonstration of fuel rod performance.

Response (6): *The staff finds that the performance of the Callaway GAIA fuel rods is evaluated in the same manner as demonstrated in the sample problem for TR ANP-10337P-A and therefore acceptable.*

(7) As indicated in TR ANP-10337P when orthogonal deflections from separate core locations are artificially superimposed to calculate component stresses, the component stresses must be compared against the design criteria associated with control rod positions.

Response (7): *The staff finds that the analysis performed for the Callaway Unit 1 GAIA fuel fully considers the actual core location and appropriately considers the guide tube criteria for control rod positions per the requirements of TR ANP-10337P-A and is therefore acceptable.*

(8) In accordance with RG 1.92, the combination of loads for non-grid component evaluation should ideally be based on three orthogonal components (two horizontal and one vertical). **[[**

]].

Response (8): *The staff finds that the analysis performed under TR ANP-10337P-A (Reference* 3*) is performed in accordance with RG 1.92 and combines load based on three-orthogonal components and therefore acceptable.*

(9) [[

]] and therefore acceptable.

Response (9): *The staff finds that the grid impact loads predicted by Framatome for the Callaway GAIA fuel design* **[[**
]] *and is, therefore, acceptable.*

3.7.4 Fuel Rod Analysis

The fuel rod thermal-mechanical performance evaluations are performed according to the NRC staff approved TR ANP-10342P-A, Revision 0. The fuel rod analyses listed in this section are verified and/or performed to demonstrate that the fuel rod design criteria defined within TR ANP-10342P-A are satisfied for the GAIA design up to a peak rod average burnup of 62 GWd/MTU for UO_2 rods and 55 GWd/MTU for gadolinia rods. The Framatome methodology requires these analyses to be verified and/or re-performed on a cycle-specific basis to ensure that the fuel will not result in SAFDL non-compliance. The fuel rod analyses are listed in sections 2.4.4.1 through 2.4.4.8 of TR ANP-3947P. Table 2-6 of TR ANP-3947P lists the following fuel mechanical evaluation criteria and results.

Fuel System Damage
- Stress, Strain Loading Limits
- Strain Fatigue
- Fretting Wear
- Oxidation, Hydriding and Crud
- Fuel Rod / Fuel Assembly Bow and Growth
- Fuel Rod Internal Pressure
- Fuel Assembly Lift-Off

Fuel Rod Failure
- Hydriding
- Cladding Collapse
- Overheating of Fuel Pellets

Fuel Coolability
- Structural Deformation

Additional Items

- Overheating of Cladding
- Excessive Fuel Enthalpy
- Bursting
- Cladding Embrittlement
- Violent Expulsion of Fuel
- Reactivity Coefficients
- Fuel Rod Ballooning

3.7.5 Fuel Mechanical Design Conclusion

The NRC staff reviewed the mechanical design evaluations for the fuel assembly and fuel rod for the Framatome GAIA fuel design for Callaway. The NRC staff finds the GAIA fuel design is compatible with core internals, control system components, co-resident fuel (section 3.9.2 of this SE) in-core instrumentation, and RCCAs at Callaway. In addition, the staff finds that all fuel mechanical design analyses were performed consistent with referenced NRC approved methodologies and that the licensee appropriately accounted for information not available to Framatome, as described in earlier sections. The staff determined that the SSE/LOCA spacer grid acceptance criteria for deformation have been met and do not challenge the "coolability" of the fuel assembly. Similarly, the NRC staff determined that the acceptance criteria for structural integrity of M5 cladded fuel rod have been met. The NRC staff reviewed the mechanical design evaluations and determined that the design criteria are met to the licensed peak fuel rod burnup of 62 GWd/MTU and peak Gadolinia bearing fuel rod burnup of 55 GWd/MTU under normal and faulted operating conditions.

3.8 Nuclear Design

This section evaluates the Callaway core design analysis to verify the cycle specific reload design and the key safety parameters in reload analysis. The effects of transition from the co-resident fuel to the Framatome GAIA fuel on the nuclear design bases and methodologies for the Callaway are evaluated in this section. Sections 2.1 and 2.3 of TR ANP-3947P describe the differences between the Framatome GAIA fuel and co-resident Westinghouse fuel design.

3.8.1 Methodology

The Callaway core design is based on the ARCADIA code system in TR ANP-10297P-A, Revision 0 and TR ANP-10297P-A, Revision 0, Supplement 1PA, Revision 1 for the cycles including the transition cycles and future operation of Callaway with the Framatome GAIA fuel design. The ARCADIA code system is an NRC staff approved Framatome neutronics methodology and associated codes.

3.8.2 Nuclear Core Design Evaluations

The Callaway transition consists of two transition cycles followed by the reference cycle or representative cycle with full core of GAIA fuel. The loading patterns for a core power of 3565 MWt were developed based on design requirements (e.g., energy, peaking, and pin burnup limits) specified for Callaway. The first transition cycle contains fresh Framatome GAIA fuel with once-burnt and twice-burnt co-resident fuel. The second transition cycle contains fresh and once burnt Framatome GAIA fuel with twice-burnt co-resident fuel. The third cycle or reference cycle contains only Framatome GAIA fuel. These core designs show that sufficient margin exists between typical safety parameter values and the corresponding limits to allow flexibility in the development of reload cores.

The peaking factors and compliance with TS requirements have been achieved by the combination of fresh fuel enrichment loading and integrated burnable absorber enrichments and loadings. Changes in boron concentration and axial offset are typical of normal cycle-to-cycle variations in the core design.

3.8.3 Results and Conclusion

Key safety parameters are listed in table 3-2 of this SE and their margins are maintained during the transition from co-resident fuel to Framatome GAIA fuel design. Power peaking and reactivity parameters remain within the limits specified in the TSs for the transition cycles and reference cycle and are consistent with and bounding of cycle-to-cycle variations in core loading patterns. These parameters vary from cycle-to-cycle satisfying energy requirements controlled though the feed batch size and Uranium enrichment together with the use of Gadolinia bearing fuel rods as absorbers.

The NRC staff recognizes that these findings were based on the nominal core designs provided in the LAR, which demonstrate that the licensee can meet its design requirements using the GAIA fuel without any unusual changes to core behavior that may warrant further review investigation.

The NRC staff reviewed the details of the Callaway core design during the transition from co-resident fuel to Framatome GAIA fuel design as presented in TR ANP-3947P. The NRC staff determined that during the transition process, the peaking factors and key safety parameters are maintained within their specified limits. The NRC staff confirmed that cycle checks are performed against the reference core design safety parameters. The NRC staff determined that the core design during transition has been performed according to NRC approved methodology.

Table 3-2: Range of Key Safety Parameters

Technical Specification	Safety Parameter	Technical Analysis Value
TS 1.1	Nominal Reactor Core Power (MWt)	3565
Not a TS	Nominal Coolant System Pressure (psia)	2250
TS 3.1.1 Core Operating Limit Report (COLR) Section 2.1	Shutdown Margin (SDM) (per cent mille (pcm)) Mode 1-4 Mode 5	≥ 1300 ≥ 1000
TS 3.1.3 COLR Section 2.2.1	Most Positive Moderator Temperature Coefficient (MTC) (pcm/°F)	≤ +5 (Power ≤ 70%) ≤ 0 (Power = 100%) Linear ramp from +5 at 70% to 0 at 100%
TS 3.1.3 COLR Section 2.2.1	Most Negative MTC (pcm/°F)	> -47.9
Not a TS	Doppler Temperature Coefficient (DTC) (pcm/°F)	-1.871 to -1.485
Not a TS	Beta-Effective	0.0052 to 0.0063
Not a TS	Power Coefficient	The power coefficient is negative at all operating power levels relative to hot zero power
TS 3.2.1 COLR Section 2.5	Heat Flux Hot Channel Factor (FQ(z))	2.5
TS 3.2.2 COLR Section 2.6	Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)	1.65
TS 3.2.3 COLR Section 2.7	Axial Flux Difference at (100 percent Power) (%. Δ)	-15 to +10

3.9 Thermal and Mechanical Design

This section describes the T-H analyses that support the transition to the Framatome GAIA fuel design at Callaway. The input parameters are from design documents, fuel assembly and component characteristics established by mechanical/hydraulic testing, and plant parameters provided by the licensee. The thermal and mechanical design of the core are established based on the following acceptance criteria in SRP section 4.4

- There is at least a 95 percent probability at a 95 percent confidence level that the hot fuel rod in the core does not experience DNB during Condition I or II events.
- There is at least a 95 percent probability at a 95 percent confidence level that the hot fuel rod in the core does not melt during Condition I or II events.

Analytical assurance that DNB will not occur is provided by showing the calculated DNBR to be higher than the 95/95 design limit DNBR for Condition I and II events. The assurance that FCM will not occur is provided by comparing PLHGR to the LHGR corresponding to FCM. Assurance that fuel melting will not occur is provided by showing that the PLHGR is below the FCM limit for Condition I and II events. This evaluation was performed and the staff confirmed that the licensee has acceptable margin for each cycle as part of the reload licensing process. Based on its review, the staff finds that the licensee’s reload licensing process has acceptable methods for performing these analyses and evaluations for GAIA fuel.

3.9.1 T-H Design Methodology

Section 3.9 of report ANP-3969P provides a list of analysis methodologies used to evaluate the GAIA fuel assembly. The NRC-approved code, COBRA-FLX, is used to perform the T-H compatibility analysis and guide tube boiling analysis. Section 3.5.2 of this SE describes the computer codes and related methodologies used in the T-H analysis for Callaway. The impact of rod bowing on the MDNBR and PLHGR is evaluated using the rod bow methodology described in TR BAW-10227-A. Based on its review, the staff finds this T-H methodology is acceptable for use of GAIA fuel the Callaway station.

3.9.2 T-H Compatibility

Based on its review, the staff finds that the T-H compatibility analysis demonstrates that the GAIA fuel assembly and the co-resident fuel assembly are thermal-hydraulically compatible. The NRC staff examined the following aspects of T-H compatibility at Callaway:

Core Pressure Drop

The pressure drop calculations for the Callaway VQP and lead fuel assembly programs were determined using COBRA-FLX models with full-core GAIA, co-resident fuel, and operating cycle 25 lead fuel assembly configurations in TR ANP-10311P-A, Revision 1. The co-resident fuel assemblies have a higher overall resistance to flow than the Framatome GAIA fuel assemblies; therefore, as the core transitions from a full core of co-resident fuel to a full core of GAIA, the core pressure drop decreases. The total pressure drop associated with the full core of Framatome GAIA is lower than the total pressure drop of the co-resident core by a value of **[[ ]]** The pressure drop profile for a full Callaway core for both fuel designs is illustrated in figure 3-1 of this SE. The NRC staff recognizes that the impact of this pressure drop change on various core characteristics important to the safety analyses such as bypass flow, total core flow, etc. are discussed below.

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Total Bypass Flow

The core bypass flow is divided between the flow that bypasses the core by flowing through the assembly guide tubes and the flow that bypasses the core entirely. The bypass flow includes the following flow paths: guide tubes, vessel upper head, inlet-to-exit nozzle, and core barrel/baffle. The amount of bypass flow through the fuel assembly guide tubes is a function of the fuel assembly guide tube geometry, core pressure drops, and flow through various flow paths through the reactor vessel. The core pressure drop for a full core of Framatome GAIA fuel assemblies is lower than the core pressure drops for the co-resident fuel. As a result, the driving force for bypass flow decreases and the total bypass flow fraction decreases transitioning from the co-resident fuel to the Framatome GAIA fuel assemblies. The minor differences between the assembly geometries of GAIA and co-resident fuel have an insignificant impact on the total bypass flow.

Crossflow Velocity

The crossflow velocity is driven by the assembly-to-assembly crossflows associated with both fuel designs. The crossflow velocity is calculated with COBRA-FLX. A bounding core

configuration was considered for this analysis to cover mixed core configurations associated with the transition. **[[**
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RCS Flow Rate

The evaluation of primary system coolant loop flow indicates that the transition from a full core co-resident fuel to a full core Framatome GAIA fuel results in an increase in the RCS loop flow due to the lower pressure drop in the GAIA fuel assembly. **[[**
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Transition Core DNB Performance

The COBRA-FLX code was used to analyze the effect of the fuel transition on the DNB performance of the Framatome GAIA fuel assemblies. The purpose of this evaluation was to investigate the fuel loss coefficients on the DNB performance in a mixed core Westinghouse (co-resident) fuel and Framatome GAIA fuel. **[[**

]] Based on its review, the staff finds that the effects of GAIA fuel on co-resident fuel during the fuel transition to a full core of GAIA fuel have been acceptably evaluated and accounted for in the licensee's reload licensing process.

Control Rod Drop Times

An assessment was performed to confirm that the TS requirement for the control rod drop time is not changed during the fuel transition. The control rod drop time is primarily dependent on the number, size, and location of the guide tube weep holes, as well as the inner diameter and height of the guide tube dashpot region. Since the Framatome guide tubes and co-resident guide tubes are similar, the control rod drop time will not be significantly impacted by the fuel transition and will remain below the required drop time of 2.7 seconds. The staff finds this assessment acceptable as drop times are confirmed to be less 2.7 seconds following each refueling per surveillance requirement 3.1.4.3 contained in Callaway's technical specifications.

Thermo-Hydrodynamic Instability

Callaway has been evaluated for susceptibility to various forms of instabilities and found to be resistant to all of them. The thermo-mechanical evaluation performed by Framatome as discussed in TR ANP-3947P demonstrates that Callaway with GAIA fuel assemblies have ample margin to the conditions that might lead to thermo-hydrodynamic instabilities. The finds that the Callaway reactor fueled with GAIA assemblies will therefore satisfy Framatome's T-H acceptance criteria for avoiding thermo-hydrodynamic instabilities.

Rod Bow

Rod Bow is a phenomenon that can occur to fuel assemblies that are irradiated in the reactor. When rod bowing occurs, the local power peaking and local flow conditions can be impacted, which can reduce the FCM limit and DNBR margin. The impact of rod bow on the GAIA fuel assembly as a function of burnup is evaluated using the methodology in TR BAW-10227-A, Revision 2, Q3P Revision 0, “Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel,” July 2021 (Reference 39). The licensee’s rod bow assessment determined DNBR and LHGR penalties for rod bow. The calculated rod bow penalties shown in table 3-3 are applied to DNBR and PLHGR for fuel assemblies exceeding the exposure thresholds in the appropriate analyses. No rod bow penalty is required for fresh fuel analyzed for the VQP. The DNBR rod bow penalty is applied to the calculated value of MDNBR using the following equation:

$$MDNBR = MDNBR_{calculated}\,(1 - \delta_{rod\ bow}^{penalty})$$

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Guide Tube Heating

Boiling of coolant within the guide tubes has the potential to increase corrosion rates and be detrimental for neutron moderation. Guide tube heating analysis was performed using the COBRA-FLX code to demonstrate that boiling will not occur within the guide tubes of the Framatome GAIA fuel assemblies during operating conditions. The analysis demonstrates that for all calculated control rod heating rates, boiling is precluded within the guide tube. The staff finds that methods and process for this analysis acceptable. The guide tube heating/boiling analysis is performed for each cycle during the reload licensing process.

3.9.3 Hydraulic Characterization Comparison Between GAIA and Co-resident Fuels

Westinghouse and Framatome have assessed the impact of hydraulic differences between the two fuel designs on the fuel mechanical and T-H performance of their fuel. The NRC staff had an opportunity to review details of the evaluations referenced in the LAR and summarized below as part of a regulatory audit (Reference 40), and confirmed that the conclusions in the LAR are accurate.

Westinghouse

Westinghouse has evaluated the impact of the GAIA assemblies on the fuel mechanical design for the resident Westinghouse 17x17 Vantage+ fuel design. The evaluation addressed fuel assembly lift forces and top nozzle hold-down forces, seismic/LOCA analyses, fuel handling, and potential flow induced vibration and grid-to rod fretting wear concern. Based on the evaluations and analysis, Westinghouse concluded that the resident Westinghouse 17x17 Vantage+ fuel design will not be adversely affected by the presence of eight Framatome GAIA fuel assemblies.

The T-H analysis has resulted in DNB penalties for operating cycle 27, and any other cycle containing GAIA fuel will be applied a penalty in accordance with TR WCAP-11837P-A.

Framatome

The differences in hydraulic characteristics between the resident and GAIA fuel assembly designs have been evaluated for impact on mechanical and T-H design criteria applicable to GAIA fuel. SAFDLs and the pressure drop profile between the two assembly types have been calculated, and crossflow velocities affecting the Framatome GAIA fuel assemblies were analyzed using COBRA-FLX to assure satisfactory performance during the transition. Several transition cores were assessed, and the bounding configuration (highest cross flow velocity) was identified to cover all mixed core configurations associated with the transition.
Based on its review, the staff finds methods used for this evaluation acceptable and the results from these analyses demonstrate that the fuel design is acceptable to ensure mechanical SAFDL compliance.

The GAIA fuel assembly is associated with less overall flow resistance than the resident fuel. This 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. Margin to the DNB acceptance criteria for event specific analyses is confirmed and documented in report ANP-3969P.

The NRC staff reviewed the T-H design of the Callaway core, including T-H compatibility between GAIA fuel and co-resident fuel, hydraulic characterization of both Westinghouse and Framatome GAIA fuel. Based on its review, the NRC staff determined that the acceptance criteria for T-H design as per SRP section 4.4 have been met.

3.10 Rod Ejection Accidents (REA Analysis)

The NRC staff reviewed an REA analysis provided in report ANP-4012P, Revision 1, “Callaway Rod Ejection Accident Analysis,” dated November 2022 (enclosures 3 (non-proprietary) and 4 (proprietary) of the LAR supplement dated December 1, 2022), which was subsequently superseded by an updated REA analysis (report ANP-4012P, Revision 2, August 2023) provided by the licensee in enclosures 2 (non-proprietary) and 3 (proprietary) to the LAR supplement dated August 3, 2023, to correct an issue identified in the original analysis. The discussion below was confirmed to be applicable to the more recent analysis in report ANP-4012P, Revision 2.

Accident Description and Analysis Method

This event is initiated by a postulated rupture of a control rod drive mechanism housing that allows the full system pressure to act on the drive shaft, which ejects its control rod from the core. The consequence is rapid positive reactivity insertion, a core power excursion, and an increase in radial power peaking, which potentially leads to localized fuel rod damage. The power excursion will be mitigated by the fuel temperature (Doppler) feedback, and, in some cases, the event is terminated by the RPS with a reactor trip in response to changes in neutron flux or system pressure.

The analysis was performed based on the VQP representative cycle design and is applicable to transition cycles containing co-resident fuel with the GAIA fuel, and cycle designs containing a full core of GAIA fuel. The analysis is performed using Framatome's AREA methodology in TR ANP-10297P-A, Revision 0. This methodology is compliant with the criteria defined in RG 1.236. The criteria within the AREA methodology consists of the following:

- TR ANP-10338P-A, Revision 0, section 6.6.2 applies the DNBR criterion to non-prompt critical REAs and to prompt critical REAs at times greater than 3 seconds (s) after the pulse. For prompt critical REAs the high clad temperature failure criterion is used for the first 3 s.

- The enthalpy rise limit is based on excess hydrogen as defined in RG 1.236. The enthalpy limit used for high temperature cladding failure threshold in RG 1.236 is a function of internal pin pressure with a maximum of 170 calories per gram (cal/g) for internal pressures less than system pressure and a minimum limit of 100 cal/g for internal pressures higher than system pressure.

RG 1.236 has the following restrictions for coolability:

- Peak radial average fuel enthalpy must remain below 230 cal/g.

- A limited amount of fuel melting is acceptable provided it is restricted to the fuel centerline region and is less than 10 percent of pellet volume. The peak fuel temperature in the outer 90 percent of the pellet's volume must remain below incipient fuel melting conditions.

Methodologies Implied in TR ANP-10297P-A, Revision 0

- GALILEO (Reference 41) is used as the fuel performance code in this analysis, which is the GALILEO version approved by NRC.

- A fuel temperature uncertainty of **[[   ]]** is used in the AREA methodology. This value is based on the database used with the version of GALILEO.

- In the Callaway REA analysis, prompt critical is defined when transients have a **[[**

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- The enthalpy rise limits are based on prompt critical testing. For non-prompt critical ejected rod worths, there is no fast power pulse.

- **[[ ]**

- System pressure calculations were not performed as part of the Callaway REA event analysis.

Cycle Inputs

The Callaway REA analysis was performed for a full core Framatome GAIA fuel with M5® cladding considering the impact of transition cycles.

The analysis is performed for **[[ ]]** times in life (TIL) and at **[[ ]]** power levels for each TIL. The TILs considered are **[[ ]]**. The selected power levels are **[[ ]]**. Table 3-4 below, lists the power levels and the corresponding rod position used in the analysis.

Table 3-4 Callaway Rod Insertion Limits with Respect to Power

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Table 3-5 below, lists the required penalizations applied in the AREA analysis with the depressurization curve supporting the MDNBR analysis provided in figure 2-1 of report ANP-4012P, Revision 2 (enclosures 2 and 3 of Reference 5).

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REA Limits Generated by GALILEO

The Pellet Clad Mechanical Interaction limits for excess hydrogen are calculated using GALILEO. The hydrogen update model in GALILEO calculates the total hydrogen content in the clad as a function of fuel rod burnup. The AREA methodology is used to estimate the limiting corrosion as a function of burnup to generate the enthalpy rise failure limit. Excess hydrogen is calculated by subtracting the solubility limit from the GALILEO prediction of total hydrogen content at selected burnups from the bounding fuel pin history depletion.

Fuel Integrity Summaries

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]] The margins reported are based on the calculated value minus the limit, so that a negative number is favorable. A positive value indicates a violation of the limit. Additional detail is provided for the cases with the least margin to the limit for fuel melt, fuel rim melt, MDNBR, enthalpy, and enthalpy rise. **[[**

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The limiting results for the transient cases at each power level are listed in tables 4-2 through 4-6 of report ANP-4012P, Revision 2 for **[[**
]], respectively, for GAIA fuel. The results reported in tables 4-2 through 4-6 are summarized in table 3-6 below, which provides limiting criteria for power level, cycle burnup, limiting value, and estimated level of conservatism (limiting value – nominal value).

Table 3-6 Measure of Conservatism for Limiting Results

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The NRC staff reviewed the Callaway REA analysis as described in ANP-4012P, Revision 2 (enclosures 2 and 3 of Reference 5). The NRC staff determined that the analysis was performed according to NRC-approved AREA methodology and is consistent with RG 1.236. The fuel related acceptance criteria for this event are evaluated to support the fuel transition. The use of Framatome's AREA methodology coupled evaluation model permits the use of conservative but **[[**
]] The AREA methodology implementation for mixed core applications is addressed in the development of parameter biasing to account for cycle-to-cycle changes with co-resident fuel. The NRC staff determined that the REA analysis provides ample margin to limits for fuel temperature, fuel rim temperature, MDNBR, enthalpy rise that mean there are no fuel failures associated with this event.

3.11 Summary and Conclusions on GAIA Fuel Design, and Evaluations of Mechanical, Structural, T-H, and REAs

The NRC staff reviewed the Framatome's GAIA fuel design, mechanical and structural evaluations for the fuel design including the compatibility assessment with co-resident fuel, nuclear design bases and the methodologies for the transition and reference cycles. The NRC staff reviewed the T-H design of the reactor core that ensures the core can meet steady-state and transient performance requirements without violating the acceptance criteria. The NRC staff also reviewed the report ANP-4012P, Revision 2, which describes the Callaway REA with ARCADIA methodology satisfying the acceptance criteria for DNBR, enthalpy rise, and enthalpy limit. The NRC staff determined that these documents provide assurance that the plant licensing bases will be met for the anticipated operation of the Framatome GAIA fuel during the transition supports the use of GAIA fuel at Callaway.

3.12 CONCLUSIONS

The NRC staff reviewed the LAR in conjunction with additional and supplemental information listed in various sections of this SE related to the proposed amendments to allow loading of a limited number of Framatome GAIA fuel assemblies with M5® cladding material starting in operating cycle 27 at Callaway.

Based on its review, as summarized in various sections of this SE, the NRC staff concludes that the licensee provided adequate technical basis to support the proposed TS changes. Specifically, the NRC staff finds the licensee has demonstrated that (1) it complies with the staff limitations and conditions imposed for application of the TRs where applicable, (2) the Framatome GAIA fuel assembly specific safety analyses results meet the applicable licensing criteria, and (3) the proposed TS changes are acceptable and satisfy the 10 CFR 50.36 requirements. Further, as noted in 10 CFR 50.36(c)(2), "[w]hen a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met." Therefore, the NRC staff finds there is reasonable assurance of public health and safety.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Missouri State official was notified of the proposed issuance of the amendment on August 24, 2023. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment allows for the use of up to eight Framatome GAIA fuel assemblies to demonstrate operating characteristics for supporting the option of transitioning from the use of fuel manufactured by Westinghouse Electric Company. The amendment would revise the TSs to allow use of Framatome GAIA fuel with M5® as a fuel cladding material. The amendment is supported by a separate exemption request from the provisions of 10 CFR 50.46 and Appendix K of 10 CFR Part 50. The amendment changes requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. Based on the licensee's response during an NRC regulatory audit (Reference 40), the NRC staff has determined that the amendments involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on March 7, 2023 (88 FR 14184). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Farnsworth, D. E., Union Electric Company, dba Ameren Missouri letter to NRC, "Docket Number 50-483, Callaway Plant Unit 1, Union Electric Co., Renewed Facility Operating License NPF-30, Application for Technical Specification Change and Exemption Request Regarding Use of Framatome GAIA Fuel (LDCN 22-0002)," dated October 12, 2022 (Agencywide Documents Access and Management System (ADAMS) Accession Package ML22285A115).

2. Farnsworth, D. E., Union Electric Company, dba Ameren Missouri letter to NRC, "Docket Number 50-483, Callaway Plant Unit 1, Union Electric Co., Renewed Facility Operating License NPF-30, Supplement to License Amendment Request and Exemption Request Regarding Use of Framatome GAIA Fuel (LDCN 22-0002) (EPID L-2022-LLA-0150 and EPID L-2022-LLE-0030)," dated December 1, 2022 (Package ML22335A497).

3. Farnsworth, D. E., Union Electric Company, dba Ameren Missouri letter to NRC, "Docket Number 50-483, Callaway Plant Unit 1, Union Electric Co., Renewed Facility Operating License NPF-30, Post-Audit Supplement to License Amendment Request and Exemption to Allow Use of Framatome GAIA Fuel (LDCN 22-0002) [EPID L-2022-LLA-0150 and EPID L-2022-LLE-0030]," May 9, 2023 (Package ML23129A793).

4. Jungmann, B. L., Union Electric Company, dba Ameren Missouri letter to NRC, "Docket Number 50-483 Callaway Plant Unit 1 Union Electric Co. Renewed Facility Operating License NPF-30, Post-Audit Follow-up Information in Support of Callaway's License Amendment Request and Proposed Exemption to Allow Use of Framatome GAIA Fuel (LDCN 22-0002) [EPID L-2022-LLA-0150 AND EPID L-2022-LLE-00301]," dated June 21, 2023 (ML23172A145).

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6. Chawla, M., NRC, letter to F. Diya, Union Electric Company, dba Ameren Missouri, “Callaway Plant, Unit No. 1 – Exemption from the Requirements of 10 CFR Part 50, Section 50.46 and Appendix K Regarding Use of M5 Cladding Material (EPID L-2022-LLE-0030), dated October 5, 2023 (Package ML23234A159).

7. Meyer, S. J., Ameren Missouri, letter to NRC, “Docket Number 50-483, Callaway Plant Unit 1, Union Electric Co., Renewed Facility Operating License NPF-30, Final Safety Analysis Report Revision OL-25 and Technical Specification Bases Revision 16,” dated June 22, 2021 (Package ML21195A333).

8. NRC, “Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” NUREG-0800

   a. Section 3.7.1, Revision 4, “Seismic Design Parameters, dated December 2015 (ML14198A460).

   b. Section 4.2, Revision 3, “Fuel System Design,” dated March 2007 (ML070740002).

   c. Section 4.4, Revision 2, “Thermal and Hydraulic Design,” dated March 2007 (ML07055060).

   d. Chapter 15, Revision 3, “Introduction - Transient and Accident Analyses,” dated March 2007 (ML070710376).

9. NRC, “Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents,” RG 1.236, dated June 2020 (ML20055F490).

10. NRC, “Combining Modal Responses and Spatial Components in Seismic Response Analysis,” Regulatory Guide 1.92, Revision 3, dated October 2012 (ML12220A043).

11. NRC, “Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors,” GL 2004-02, dated September 13, 2004 (ML042360586).

12. NRC, “Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength,” IN 2012-09, dated June 28, 2012 (ML113470490).

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16. AREVA Inc., "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," TR EMF-2103(P)(A), Revision 3, dated June 2016 (Package ML16286A579).

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18. Lingam, S., NRC, letter to M. L., Lacal, Arizona Public Service Company, "Palo Verde Nuclear Generating Station, Units 1, 2, and 3 - Issuance of Amendment Nos. 212, 212, and 212 to Revise Technical Specifications to Support the Implementation of Framatome High Thermal Performance Fuel (EPID L-2018-LLA-0194)," dated March 4, 2020 (ML20031C947 (public) and ML20031C968 (not publicly available proprietary information)).

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8.0 ACRONYMS/ABBREVIATIONS

Acronym	**Definition**
AC	Alternating Current
AFW	Auxiliary Feedwater
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
AOR	Analysis of Record
AREA	ARCADIA Rod Ejection Accident
ASME	American Society of Mechanical Engineers
BOC	Beginning-of-Cycle
BOL	Beginning of Life
BDMS	Boron Dilution Mitigation System
BWR	Boiling Water Reactor
CAP	Containment Accident Pressure
CFR	*Code of Federal Regulations*
CHF	Critical Heat Flux
COLR	Core Operating Limit Report
CVCS	Chemical and Volume Control System
CWO	Core Wide Oxidation
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio

Acronym	Definition
DTC	Doppler Temperature Coefficient
ECCS	Emergency Core Cooling System
EFPD	Effective Full Power Days
EM	Evaluation Model
EOC	End-of-Cycle
EOL	End of Life
ESFAS	Engineered Safety Feature Actuation System
^oF	Degrees Fahrenheit
$F_{\Delta H}$	Nuclear Enthalpy Rise Factor/Radial Peaking Factor
F_Q	Total Peaking Factor/Global Peaking Factor
FCM	Fuel Centerline Melt
FSAR SP	Final Safety Analysis Report (Standard Plant)
FSRR	Fuel Swelling, Rupture, and Relocation
GDC	General Design Criteria
GL	Generic Letter
gpm	gallons per minute
GSI	Generic Safety Issue
GWd	Gigawatt days
HFP	Hot Full Power
HHSI	High Head Safety Injection
HMP	High Mechanical Performance
HZP	Hot Zero Power
IFM	Intermediate Flow Mixer
IGM	Intermediate GAIA Mixing
IHSI	Intermediate Head Safety Injection
IN	Information Notice
ISG	Intermediate Spacer Grid
k(z)	Axial-Dependent Peaking Factor
L&C	Limitation and Condition
LAR	License Amendment Request
LBLOCA	Large Break Loss-of-Coolant Accident
LCO	Limiting Condition of Operation
LHGR	Linear Heat Generation Rate
LHSI	Low Head Safety Injection
LOCA	Loss-of-Coolant Accident
LOOP	Loss-of-Offsite Power
M&E	Mass and Energy
MDNBR	Minimum Departure from Nuclear Boiling Ratio
MLO	Maximum Local Oxidation
MOC	Middle of Cycle
MSLB	Main Steam Line Break
MTC	Moderator Temperature Coefficient
MTU	Metric Ton Uranium
MWt	Megawatt thermal
NPSH	Net Positive Suction Head
NQA	Nuclear Quality Assurance
NRC	Nuclear Regulatory Commission

Acronym	Definition
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
OTΔT	Overtemperature Delta Temperature
pcm	percent mille (one-thousandth of a percent)
PCT	Peak Cladding Temperature
PLC	Pressure Loss Coefficient
PLHGR	Peak Linear Heat Generation Rate
ppm	parts per million
psi	pounds per square inch
PSV	Pressurizer Safety Valve
PV	Pressure-Velocity
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
REA	Rod Ejection Analysis
RG	Regulatory Guide
RHR	Residual Heat Removal
RPS	Reactor Protection System
RTP	Rated Thermal Power
SAFDL	Specified Acceptable Fuel Design Limit
SBLOCA	Small Break Loss-of-Coolant Accident
SE	Safety Evaluation
SG	Steam Generator
SI	Safety Injection
SL	Safety Limit
SPC	Siemens Power Corporation
SRM	Swelling and Rupture Model
SRP	Standard Review Plan
SRSS	Square Root of Sum of Squares
SSE	Safe Shutdown Earthquake
T-H	Thermal-Hydraulic
TIL	Time in Life
TR	Topical Report
TS	Technical Specification
UO2	Uranium di-Oxide
UTL	Upper Tolerance Limit
VQP	Vendor Qualification Program

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Date: October 5, 2023

SUBJECT: CALLAWAY PLANT, UNIT NO. 1 – ISSUANCE OF AMENDMENT NO. 235 TO REVISE TECHNICAL SPECIFICATIONS TO USE FRAMATOME GAIA FUEL (EPID L-2022-LLA-0150) DATED OCTOBER 5, 2023

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