ATTACHMENT 5

NON-PROPRIETARY VERSION OF LICENSING SUMMARY REPORT

The following pages provide the non-proprietary version of the licensing summary report provided by Framatome supporting this license amendment request.

ANP-3947NP, "Callaway Unit 1 License Amendment Request Inputs for Use of Framatome Fuel," Revision 0, dated April 2022

[NON-PROPRIETARY]

129 pages follow this cover sheet

Callaway Unit 1 License Amendment Request Inputs for Use of Framatome Fuel

ANP-3947NP
Revision 0

Technical Report

April 2022

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Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

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Nomenclature

Acronym	Definition
ADAMS	Agency-wide Documents Access and Management System
AFD	Axial Flux Difference
AO	Axial Offset
AOO	Anticipated Operational Occurrence
AREA	ARCADIA Rod Ejection Analysis
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BAW	Babcock and Wilcox
BOC	Beginning of Cycle
BOL	Beginning of Life
BU	Burnup
BWFC	B&W Fuel Company
CFR	Code of Federal Regulations
CHF	Critical Heat Flux
COLR	Core Operating Limits Report
CSLL	Core Safety Limit Lines
CUF	Cumulative Usage Factor
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DTC	Doppler Temperature Coefficient
EFPD	Effective Full Power Days
EOC	End of Cycle
EOL	End of Life
FCM	Fuel Centerline Melt
FSAR SP	Final Safety Analysis Report (Standard Plant)
GAD	Gadolinia
GDC	General Design Criteria
GT	Guide Tube
HFP	Hot Full Power
HZP	Hot Zero Power
ID	Inner Diameter
IGM	Intermediate GAIA Mixer
ISG	Intermediate Spacer Grid
L&C	Limitations and Conditions
LAR	License Amendment Request
LEU	Low Enriched Uranium
LFA	Lead Fuel Assembly
LHGR	Linear Heat Generation Rate

Acronym	Definition
LOCA	Loss of Coolant Accident
LOCF	Loss of Coolant Flow
LTA	Lead Test Assembly
MDNBR	Minimum Departure from Nucleate Boiling Ratio
MOC	Middle of Cycle
MSMG	Mid-Span Mixing Grid
MTC	Moderator Temperature Coefficient
MTU	Metric Tons Uranium
NRC	Nuclear Regulatory Commission
OBE	Operating Basis Earthquake
OD	Outer Diameter
OPΔT	Overpower Delta Temperature
OTΔT	Overtemperature Delta Temperature
PCMI	Pellet Clad Mechanical Interaction
PDC	Power Distribution Control
PIE	Post-Irradiation Examination
PLHGR	Peak Linear Heat Generation Rate
POS	Pump Over Speed
PWR	Pressurized Water Reactor
QD	Quick Disconnect
RAI	Request for Additional Information
RCCA	Rod Cluster Control Assembly
RCS	Reactor Coolant System
RIA	Reactivity Initiated Accident
RG	Regulatory Guide
RLBLOCA	Realistic Large Break Loss of Coolant Accident
RPS	Reactor Protection System
RTP	Rated Thermal Power
SAFDL	Specified Acceptable Fuel Design Limit
SBLOCA	Small Break Loss of Coolant Accident
SDM	Shutdown Margin
SER	Safety Evaluation Report
SP	Standard Plant
SRP	Standard Review Plan
SRSS	Square Root of Sum of Squares
SSE	Safe Shutdown Earthquake
T-H	Thermal-Hydraulic
TCD	Thermal Conductivity Degradation
TCS	Transient Cladding Strain

Acronym	Definition
TS	Technical Specifications
USNRC	United States Nuclear Regulatory Commission
VQP	Vendor Qualification Program

1. INTRODUCTION AND DESIGN OVERVIEW

Ameren Missouri (Ameren) has contracted with Framatome to perform a Vendor Qualification Program (VQP) in support of fully licensing the GAIA Fuel Design and Framatome's evaluation methodologies for application at the Callaway Energy Center (Callaway Unit 1). Framatome is scheduled to supply four (4) GAIA Fuel Assemblies for delivery in the Summer of 2023 as part of the VQP demonstration. These assemblies will be licensed to operate in unrestricted core locations.

The Framatome GAIA fuel design consists of a 17x17 fuel rod array with M5 clad fuel rods, Q12 MONOBLOC guide tubes and instrument tube, Alloy 718 HMP top and bottom spacer grids, M5 GAIA spacer grids, M5 intermediate GAIA mixer (IGM) grids, a reconstitutable top nozzle, and the GRIP bottom nozzle. The fuel design is detailed in the United States Nuclear Regulatory Commission (USNRC) approved GAIA Fuel Assembly Mechanical Design Topical Report (Reference 1). Section 2.1 of Reference 1 provides an overview of the design features of the GAIA fuel assembly. Framatome's GAIA fuel design has demonstrated acceptable performance in Westinghouse 3-loop and 4-loop reactors.

Ameren is currently operating four (4) GAIA Lead Fuel Assemblies (LFAs) at the Callaway plant to demonstrate compatibility and performance of this new fuel product for Callaway Unit 1 in support of the VQP.

Section 2 of the report outlines Framatome's mechanical and structural evaluations for the fuel design including the compatibility assessment and review of operating experience. Section 3 discusses the nuclear design bases and the methodologies for transitioning from the co-resident fuel design to the Framatome GAIA fuel design for Callaway Unit 1. Section 4 describes the power distribution control methodology being used at Callaway Unit 1 to ensure that power peaking limits are not exceeded and provides example results. Section 5 provides the thermal-hydraulic design of the reactor that ensures the core can meet steady-state and transient performance requirements without violating the acceptance criteria. Section 6 provides information related to the Callaway Unit 1 transient and accident analyses for the proposed transition. Summary reports for the non-loss-of-coolant accident (non-LOCA), small break LOCA (SBLOCA), and realistic large break LOCA (RLBLOCA) analysis methodologies have been prepared as documented in References 35, 36, and 37 respectively.

Note that demonstration of the evaluation methodologies has been performed with a representative core design that was developed to provide key safety parameters to support a transition from Westinghouse fuel (co-resident fuel) to Framatome's GAIA fuel prior to the development of cycle-specific designs. This provides assurance that the plant licensing bases will be met for the operation of Callaway Unit 1 with the GAIA fuel during transition and full core cycles.

2. MECHANICAL DESIGN

2.1. *Description*

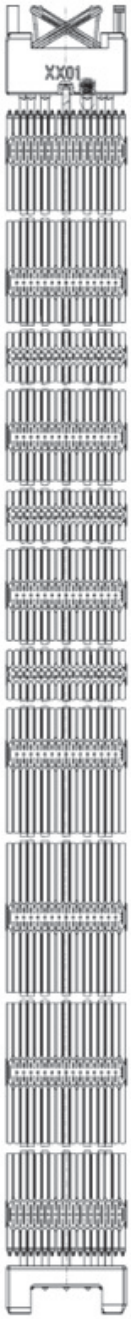
This section evaluates the mechanical design of the Framatome GAIA fuel design intended for batch implementation at Callaway Unit 1 and its compatibility with the co-resident fuel during the transition from mixed-fuel type core configurations to cores with only Framatome GAIA fuel. Framatome has performed mechanical compatibility evaluations to ensure acceptable fit-up with Callaway Unit 1 reactor core internals, fuel handling equipment, and co-resident fuel. A summary of the mechanical compatibility evaluations performed by Framatome is provided in Section 2.3.

The Framatome GAIA fuel assembly design for Callaway Unit 1 is analyzed in accordance with the USNRC-approved mechanical design criteria in Reference 1. All the mechanical design criteria are met up to the licensed peak UO_2 fuel rod burnup of 62 GWd/MTU and peak Gadolinia fuel rod burnup of 55 GWd/MTU.

Section 2.2 provides an overview of operating experience gained by Framatome with Westinghouse 17x17 3-loop and 4-loop plants. Section 2.3 provides a description of the mechanical compatibility assessment of GAIA fuel design with core internals, control components, co-resident fuel assemblies, in-core instrumentation, and handling equipment. Section 2.4 describes the mechanical evaluations performed to show acceptability with the USNRC-approved design criteria.

The GAIA fuel assembly design is generically approved by the NRC in Reference 1. Figure 2-1 provides a summary of the GAIA fuel assembly design at Callaway Unit 1, with component details provided in the subsequent sections.

Figure 2-1: Summary – GAIA Fuel Assembly at Callaway Unit 1



- Standard Reconstitutable Top Nozzle
  - 3-leaf Alloy 718 holddown system with ¼-turn Quick Disconnect (QD)
- Welded Cage
  - [ ]
- 24 – Q12 MONOBLOC Guide Tubes
  - Increased cross-sectional area
  - Low creep properties
- 1 – Q12 Instrument Tube
  - Low creep properties
- 6 – M5 GAIA Spacer Grids
  - 8-line fuel rod contact
  - Trailing edge mixing vanes
- 3 – M5 IGM Grids
  - Additional mixing for thermal performance
  - Trailing edge mixing vanes
- 2 – Alloy 718 HMP End Spacer Grids
  - Upper grid relaxed to mitigate fuel rod bow
  - Positive grip force through End of Life (EOL)
- GRIP Bottom Nozzle
  - High filtering efficiency
  - Stabilized outlet flow
- 264 Fuel Rods
  - Advanced M5 Cladding
  - [ ] UO_2 pellets
  - GAD/Blanket pellets

2.1.1. Fuel Assembly Description

The GAIA fuel assembly design at Callaway Unit 1 incorporates eleven (11) spacer grids, twenty-four (24) guide tubes, one (1) instrument tube, and top and bottom nozzles to provide the structural cage for 264 fuel rods. The eleven (11) spacer grids include six (6) GAIA structural grids axially distributed along the fuel assembly, three (3) IGM grids placed between the GAIA structural grids in the top of the active fuel region, and two (2) HMP end spacer grids, one at the top and one at the bottom. The fuel rods are slightly raised off the bottom nozzle and are laterally supported by the GAIA spacer grids and HMP end spacer grids.

Twenty-four (24) ¼-turn QD mechanisms attach the top nozzle to the cage. The QD attachments welded to the top of each guide tube allow the top nozzle to be removed remotely under water without the generation of loose parts and without the need for replacement parts.

Twenty-four (24) cap screws attach the GRIP bottom nozzle to the cage. The self-capturing attachments allow the bottom nozzle to be removed remotely under water without the generation of loose parts and without the need for replacement parts.

2.1.2. Fuel Rod Description

The GAIA fuel rod design at Callaway Unit 1 has standard components, including M5 cladding, Zircaloy-4 end caps, UO_2 and Gadolinia pellets, blanket pellets, and a nickel alloy plenum spring.

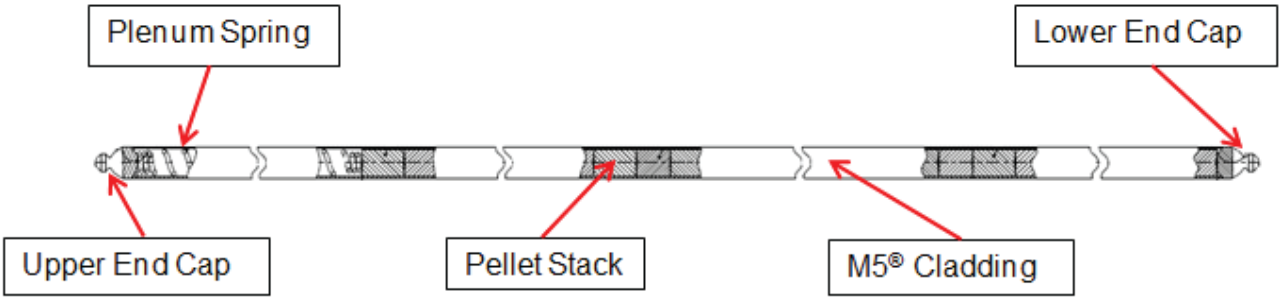
Rods are pressurized with helium to provide good heat transfer, reduce clad creepdown, and restrict pellet to cladding interaction. The design utilizes a 144 inch fuel stack length, pellet diameter of 0.3225 inch, and a nominal 0.0065 inch diametric pellet to cladding clearance. The cylindrically shaped pellets are sintered to a nominal density up to **[ ]**. Dished ends and geometric edge features ease the pellet loading into the cladding to prevent chipping and reduce the tendency of the pellet to assume an hourglass shape during operation.

The Zircaloy-4 upper and lower end caps have the same geometry. They have a bullet nose feature which provides a smooth flow transition and facilitates rod insertion during fabrication. End caps have a grippable top-hat shape that allows for removal from the fuel assembly in either direction. Upset-shape welds connect the end caps to the cladding.

The nickel alloy plenum spring is placed in the upper region, preventing the formation of fuel stack gaps during shipping and handling and allowing fuel stack expansion during operation.

Figure 2-2 highlights the fuel rod design features.

Figure 2-2: Fuel Rod Assembly



2.1.3. GAIA Spacer Grids

The GAIA fuel assembly design at Callaway Unit 1 incorporates six (6) GAIA structural grids, which are designed to combine Critical Heat Flux (CHF) performance, mechanical performance, and fretting resistance. Constructed from M5 strip material, the individual strips are slotted and assembled in an egg-crate configuration and welded at each grid strip intersection. The trailing edges of the inner strips are equipped with mixing vanes. The outer strip precludes handling damage by incorporating a thicker strip, butt-welded corner joints inboard of the square envelope, and large lead-in tabs.

Spring hulls in spacer cells provide the interface with the fuel rods. They are inserted into the bottom of the grid and welded at the bottom strip intersections. [

]. The spring hulls seat in each cell corner and vertically align with the fuel rod emulating the 8-line contact area of Framatome's HTP design. It has been confirmed by comparative tests and PIE that the fretting behavior of the GAIA spacer grid rod support is consistent with the HTP spacer grid. The magnitude of the grid restraining force on the fuel rod is set high enough to ensure sufficient fuel rod support without overstressing the cladding at the points of contact or inducing excessive axial load on the fuel rod.

The GAIA spacer grid (also referred to as intermediate spacer grid (ISG)) is connected to the twenty-four (24) guide tubes and one (1) instrument tube by [

] on each tube. In addition to maintaining the spacer grid axial positions, the weld connections help increase the fuel assembly lateral stiffness [

] welds result in a high localized rotational stiffness and high overall cage lateral stiffness. The high overall cage lateral stiffness allows the fuel assembly to resist twist and bow, particularly at high burnups when spacer grid relaxation significantly reduces the coupling between the fuel rod and structural cage.

The GAIA spacer grid maintains adequate elastic strength to support external loads (i.e., seismic and LOCA), consistent with Framatome's other spacer grid designs. In addition, the GAIA spacer grid geometry remains stable after exceeding its elastic range due to the spring hull, providing localized reinforcement at each strip intersection. Under compressive dynamic loading, this stabilizing effect results in improved mechanical behavior which resists large plastic deformation and loss of load-carrying capacity associated with a localized "buckling" failure mode. Instead, deformation of the spacer grid is uniformly distributed in each row as the [ ]. With this "compressive" failure mode, geometric changes in the guide tube and fuel rod arrays remain relatively small. Under severe external dynamic loads, the GAIA spacer grid supports safe shutdown by maintaining the coolable geometry limits and providing a path for control rod insertion. The ability of the spacer grid to [ ], via uniformly distributed deformations, protects the core during severe postulated accident conditions.

Figure 2-3 through Figure 2-6 highlight the GAIA spacer grid design features.

Figure 2-3: GAIA Spacer Grid

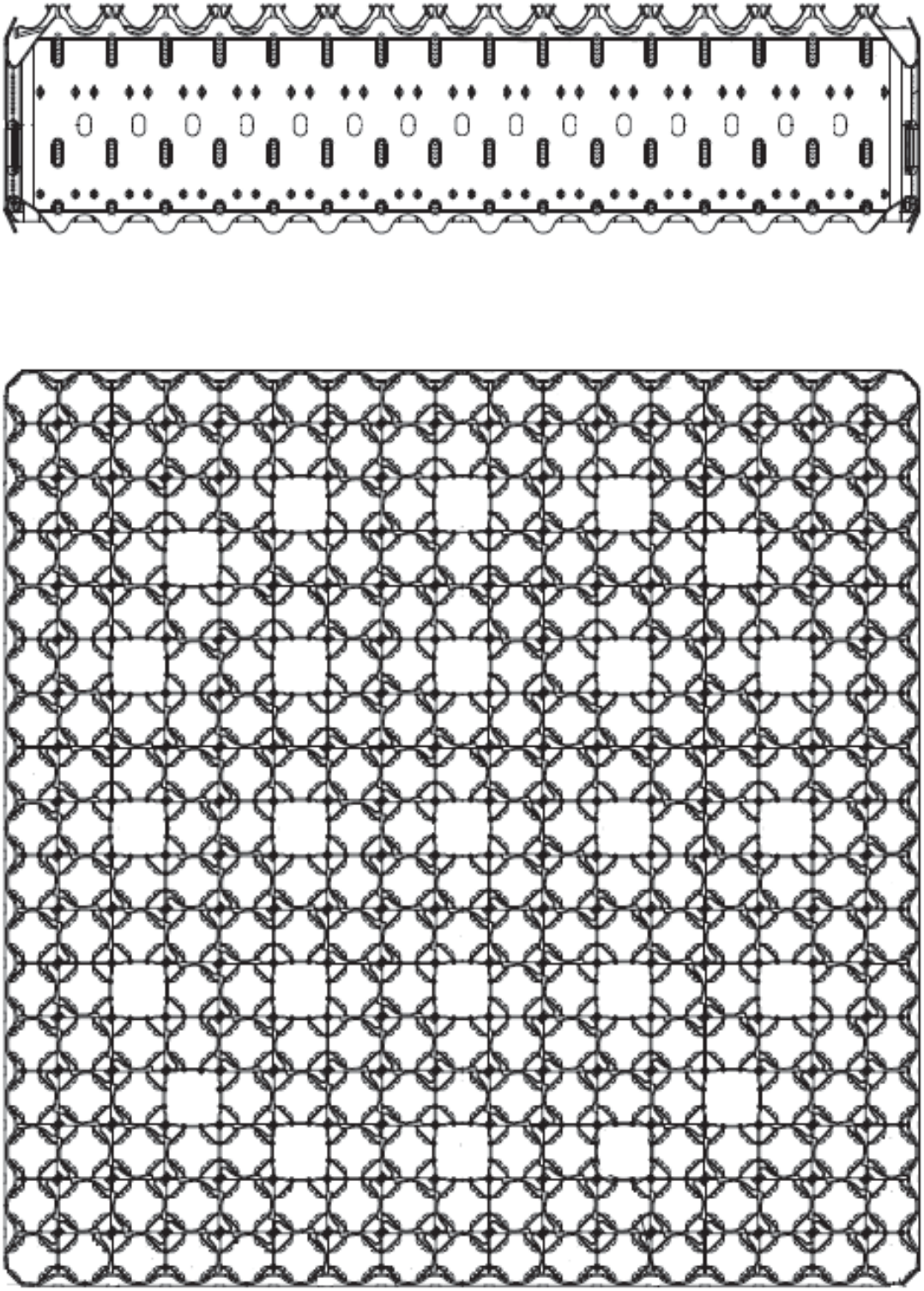


Figure 2-4: GAIA Spacer Grid – Inner/Outer Strip Features

Figure 2-5: GAIA Spacer Grid – Spring Hull Features

Figure 2-6: GAIA Spacer Grid Connection

2.1.4. Intermediate GAIA Mixing (IGM) Grids

The GAIA fuel assembly design at Callaway Unit 1 incorporates three (3) IGM grids, which are designed based on Framatome's Advanced MK-BW Mid-Span Mixing Grid (MSMG) design. The IGM grids provide additional flow mixing in the high-heat flux region for improved Departure from Nucleate Boiling (DNB) margin. Constructed from M5 material, the individual strips are slotted and assembled in an egg-crate configuration and welded at each strip intersection. To minimize the effect on bundle pressure drop, and to limit the additional material added within the active fuel region, the spacer grids are made from strips that are axially shorter than the GAIA spacer grids. The IGM has a smaller envelope than the adjacent GAIA spacer grids, which minimizes mechanical interaction with adjacent fuel assemblies.

Similar to the GAIA spacer grid, the trailing edges of the inner strips are equipped with mixing vanes to enhance mixing. The mixing vane pattern is consistent with the GAIA spacer grid.

Stops formed in each of the four (4) cell walls prevent the fuel rods from contacting the mixing vanes but impose no grip force (or slip load) onto the rods; thus, these are designated "non-contacting" spacer grids. The outer strip design precludes handling damage by incorporating a larger strip thickness (than the inner strips), wrap around corners which are inboard of the square envelope, and large leading and trailing edge lead-in tabs.

The IGM is connected to twenty-four (24) guide tubes and one (1) instrument tube by resistance spot welding to four (4) weld tabs on the top side. In addition to maintaining the spacer grid axial positions, the welded connections increase the fuel assembly lateral stiffness.

Figure 2-7 and Figure 2-8 highlight the IGM grid design features.

Figure 2-7: IGM Grid

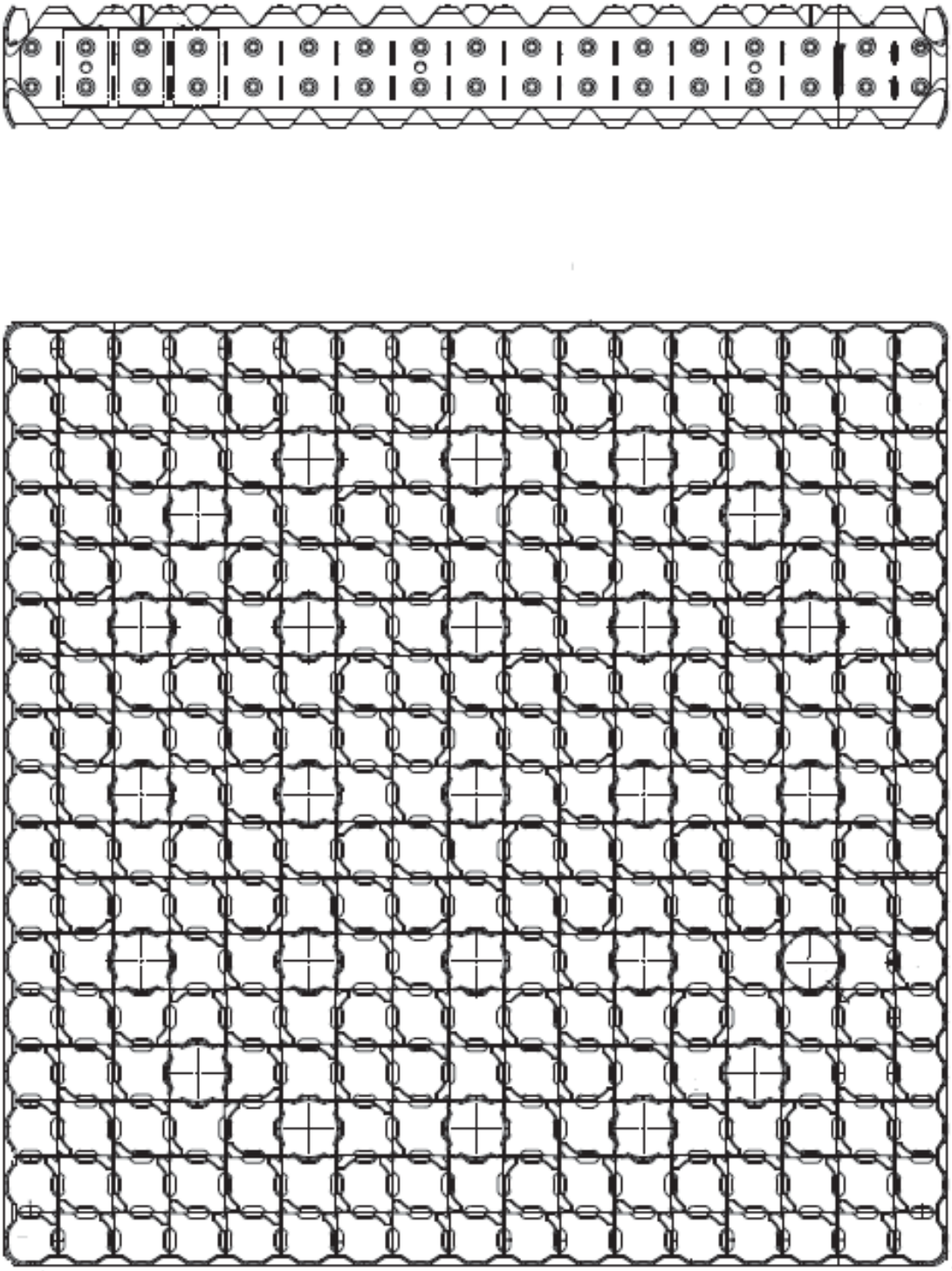
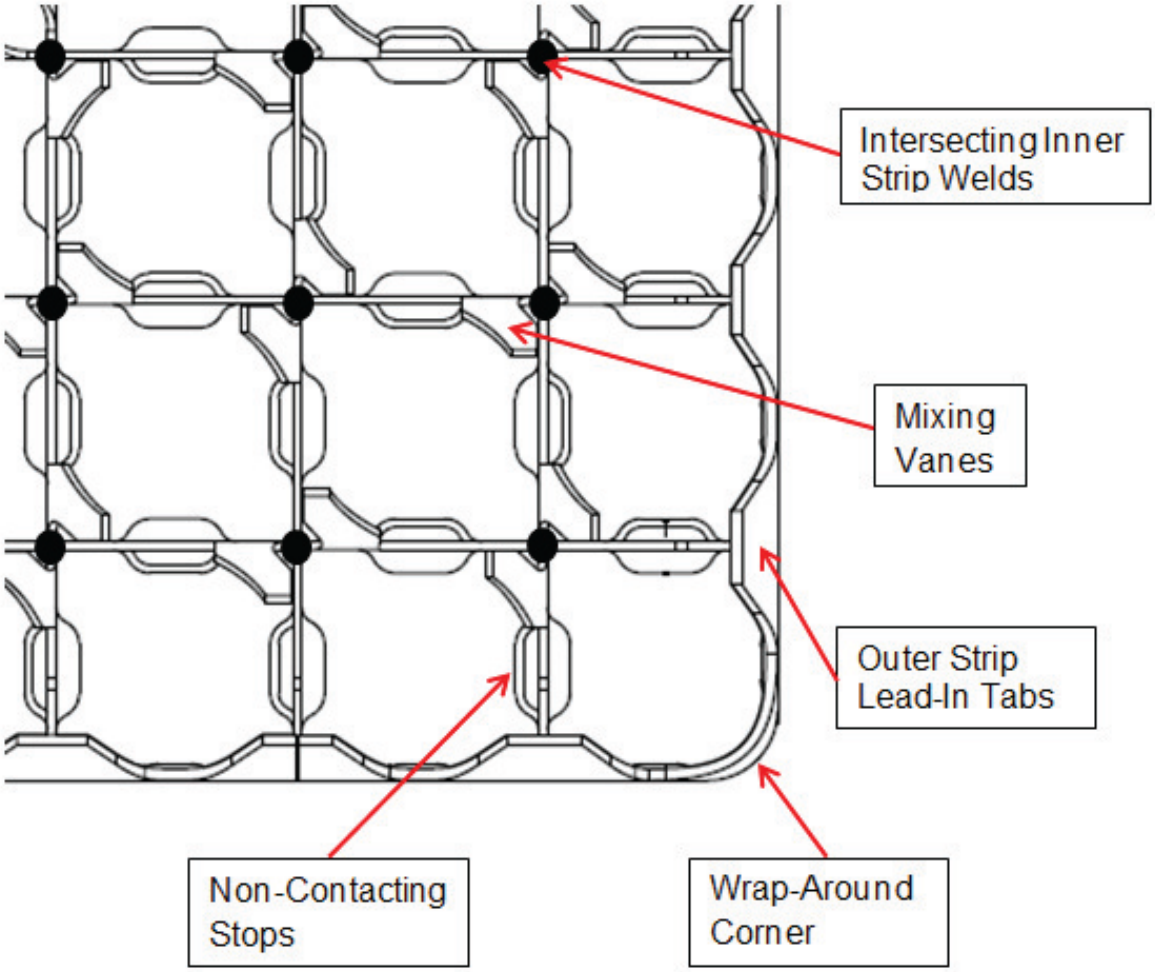


Figure 2-8: IGM Grid Features



2.1.5. HMP End Spacer Grids

The GAIA fuel assembly design at Callaway Unit 1 incorporates two (2) HMP end spacer grids. Constructed from precipitation-hardened nickel Alloy 718, individual HMP doublet strips are slotted and assembled in an egg-crate configuration and welded at each strip intersection. Each doublet is made from two individual singlet strips tack welded together, forming a straight flow channel through the center of two (2) opposing springs. The straight flow channel minimizes hydraulic resistance and is designed for locations outside the active fuel region where flow mixing is less important. The springs laterally preload and center the fuel rod within each cell and create the grid-to-rod fretting resistant 8-line interface. The outer strip design precludes handling damage by incorporating lap welded corners which are inboard of the square envelope, and large leading and trailing edge lead-in tabs.

The nickel Alloy 718 material provides high strength and reduced cell irradiation relaxation. The reduced cell relaxation, in combination with the 8-line interface, ensures that the HMP end spacer grid provides fuel rod lateral support and significant resistance against fretting wear throughout the design life.

One (1) HMP end spacer grid is used at the bottom of the fuel assembly, a region of the core where crossflows can be higher. One (1) “relaxed” HMP end spacer grid is used at the top of the fuel assembly. The “relaxed” HMP end spacer grid has a reduced slip load compared to the bottom HMP end spacer grid, which is intended to reduce the fuel rod compressive forces due to axial fuel rod growth and mitigate fuel rod bow.

The HMP end spacer grids are axially restrained by spacer sleeves, which are resistance spot welded directly to the guide tubes and instrument tube above and below the spacer grid.

Figure 2-9 through Figure 2-11 highlight the HMP end spacer grid design features.

Figure 2-9: HMP End Spacer Grid

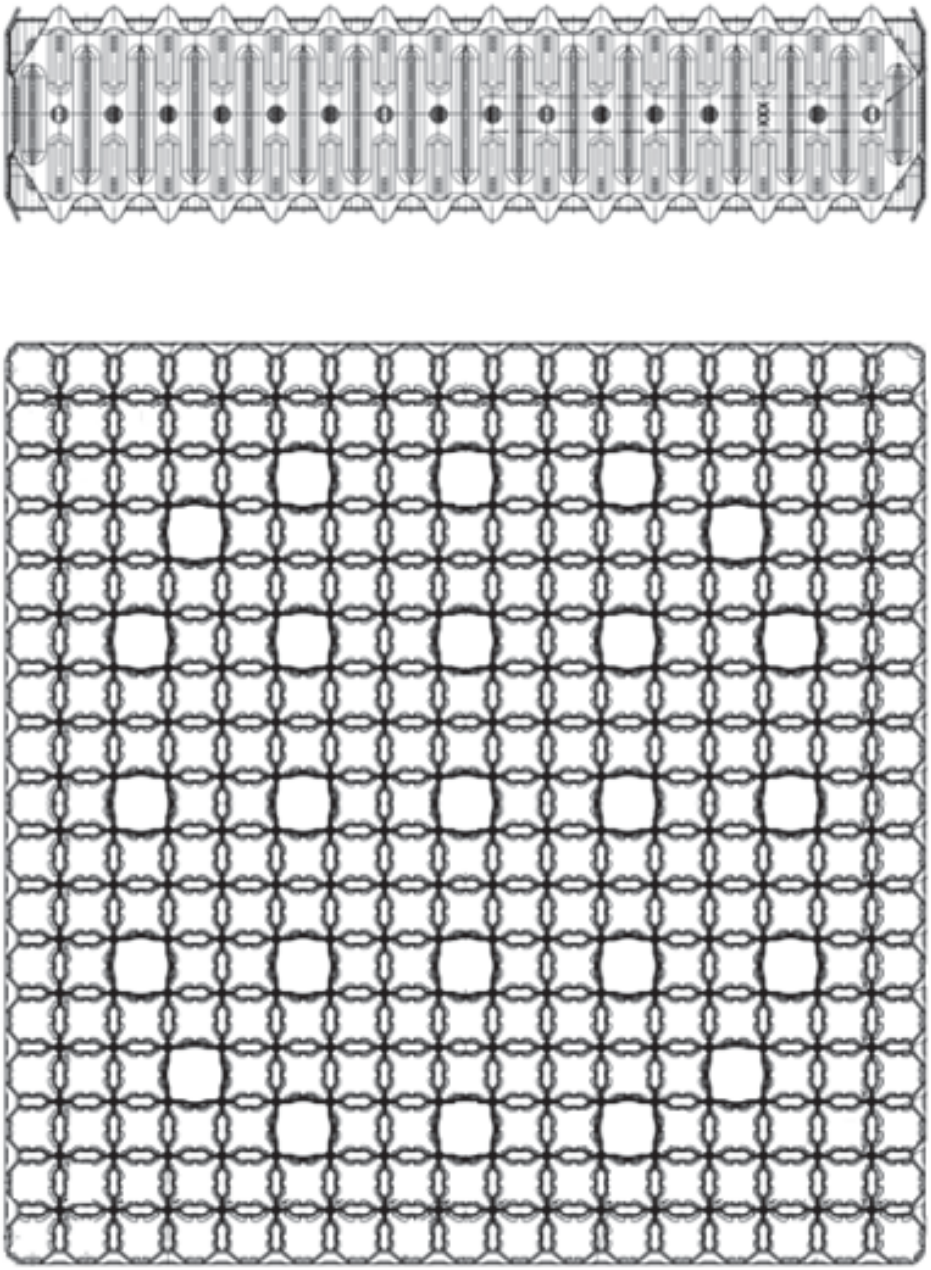


Figure 2-10: HMP End Spacer Grid Features

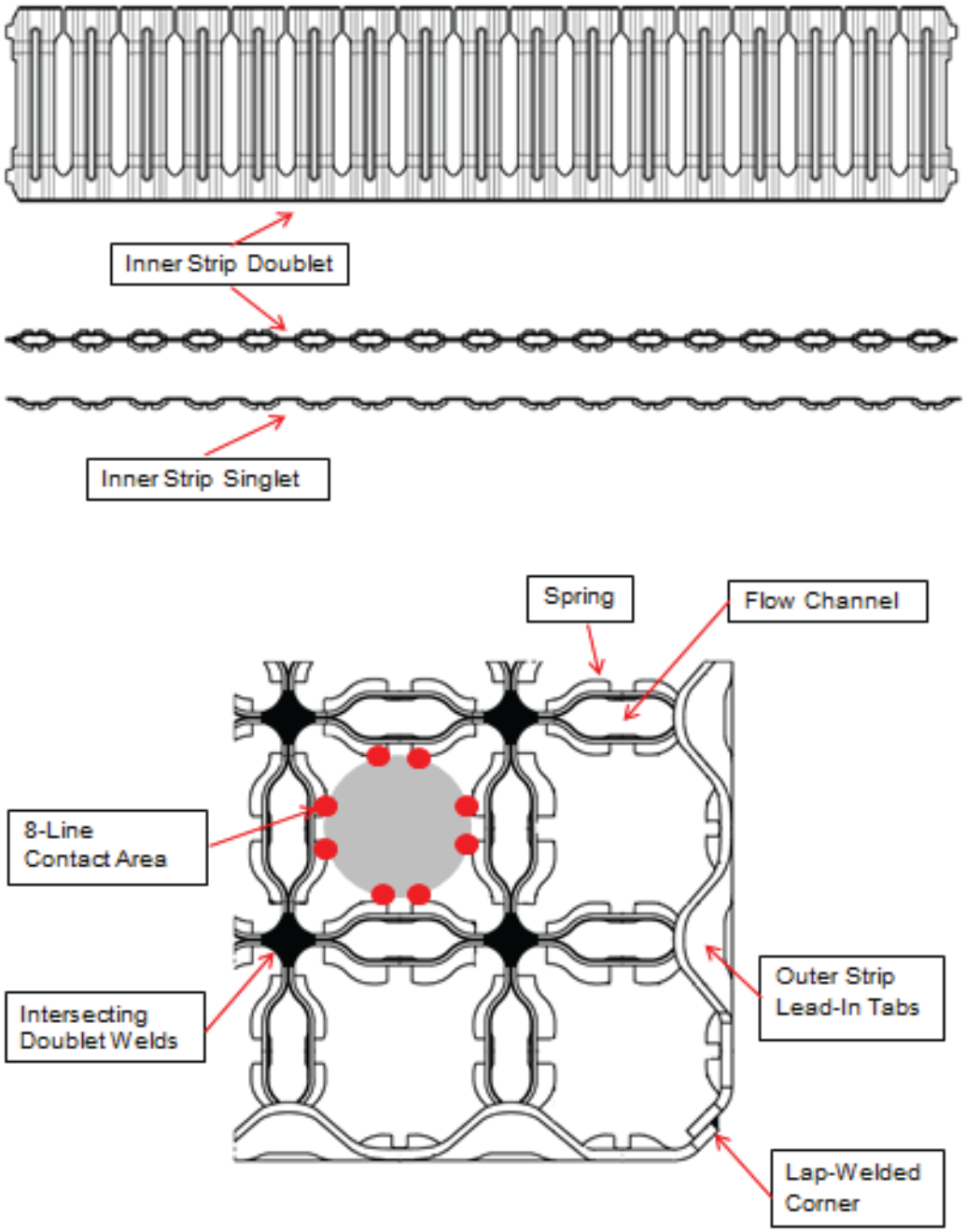
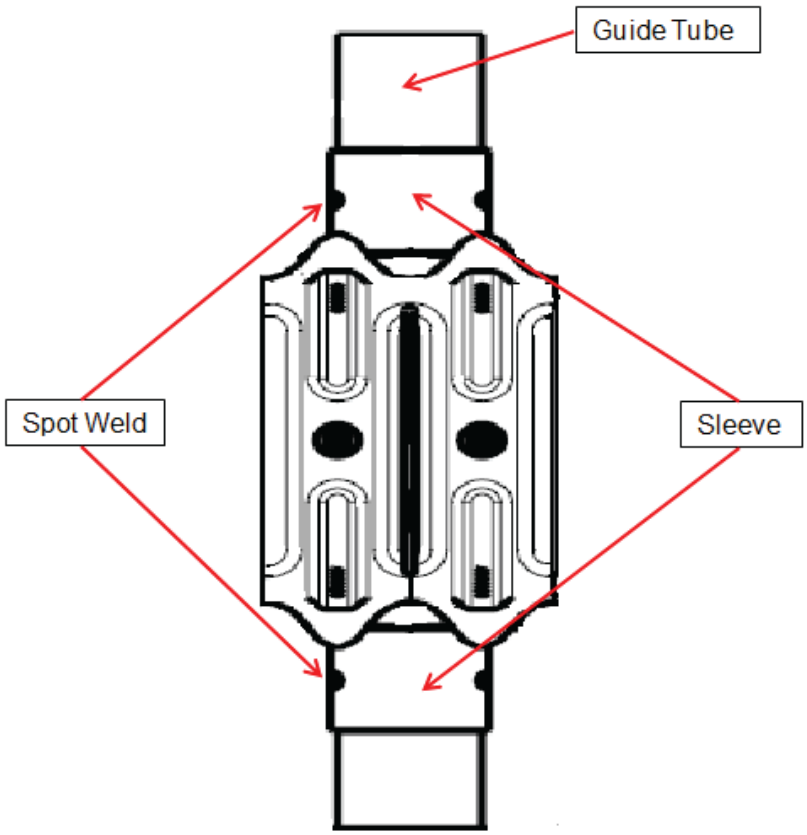


Figure 2-11: HMP End Spacer Grid Connection



2.1.6. Top Nozzle

The GAIA fuel assembly design at Callaway Unit 1 incorporates Framatome's standard top nozzle design. The top nozzle consists of two (2) high strength stainless steel bi-block frames, welded together to form a box-like structure. The upper structure interfaces with the reactor internals and the core components. The lower structure and grillage flow-hole pattern is designed to balance low pressure drop and strength requirements.

Four (4) sets of leaf springs made of nickel Alloy 718 are fastened to the nozzle with nickel Alloy 718 clamp screws. During operation, the springs prevent fuel assembly lift from hydraulic forces which ensures positive interaction with the upper and lower core internals. The upper leaf has an extended tang that engages a cutout in the top plate of the nozzle. This arrangement ensures spring leaf retention in the unlikely event of a spring leaf or clamp screw failure.

The attachment of the top nozzle to the guide tubes consists of a ¼-turn QD assembly locking mechanism, which allows easy top nozzle removal and replacement. Removal and replacement requires only hand tools, generates no loose parts, and does not require any replacement hardware. This contributes to a significant reduction in the amount of time required to perform fuel repairs and inspections. The locking mechanism is designed to rotate 90° in either direction to lock or unlock, and provides a positive indication when rotation is complete.

Figure 2-12 and Figure 2-13 highlight the top nozzle design features.

Figure 2-12: Top Nozzle

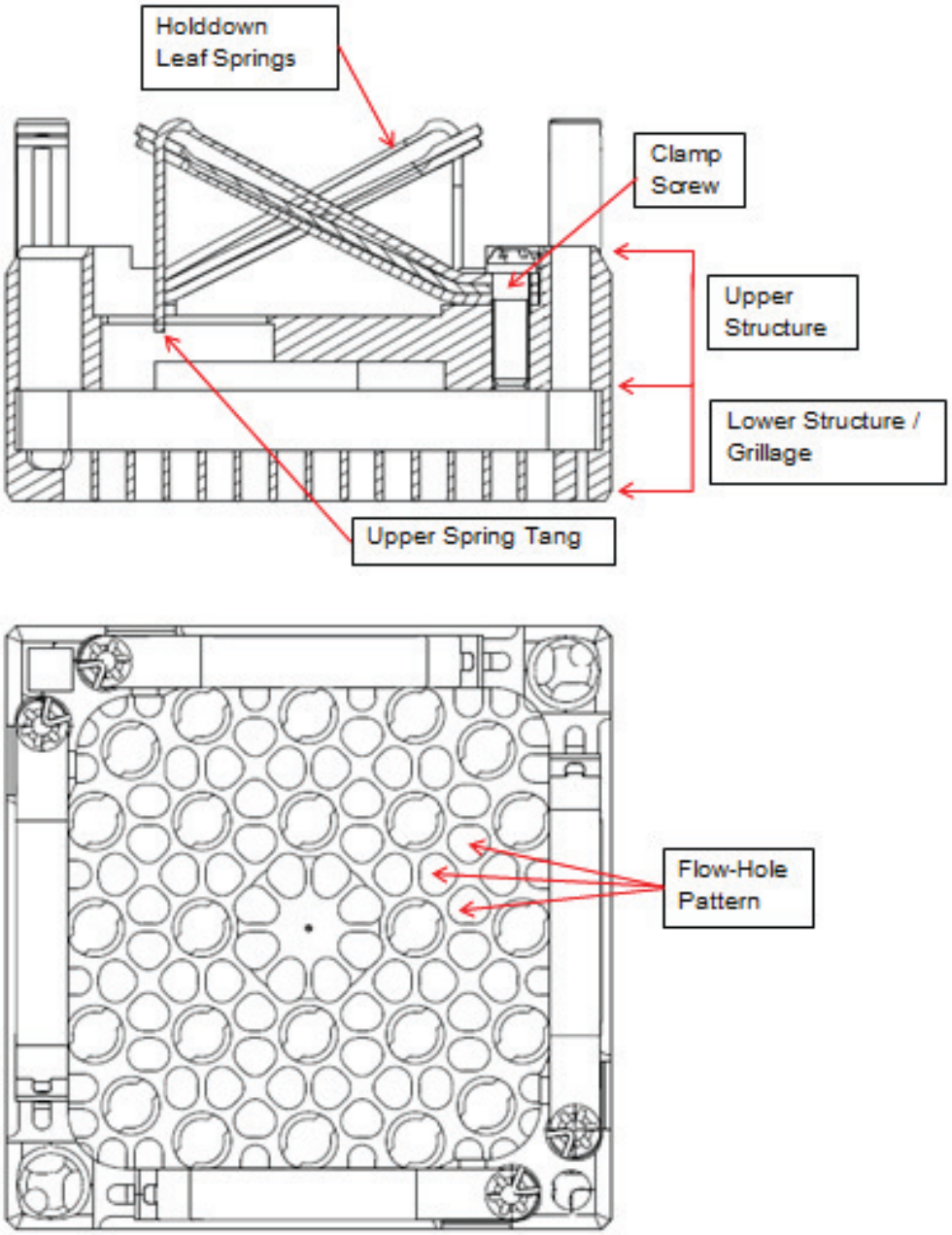
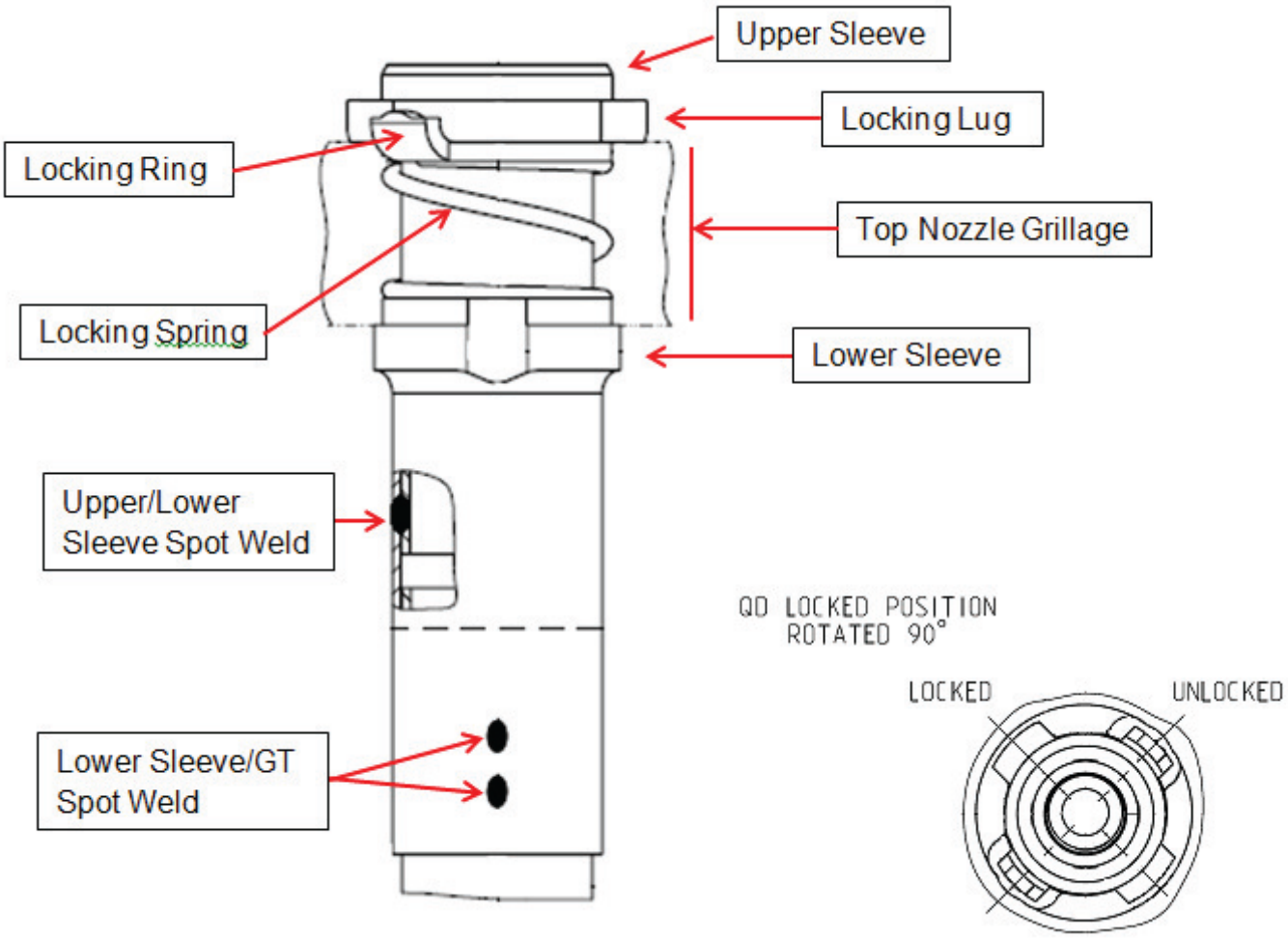


Figure 2-13: Top Nozzle 1/4-Turn QD



2.1.7. GRIP Bottom Nozzle

The GAIA fuel assembly design at Callaway Unit 1 incorporates the GRIP bottom nozzle design, which combines high filter efficiency, high mechanical robustness, and low pressure drop characteristics, along with features for flow stabilization and protection against excessive rod vibration.

The GRIP bottom nozzle is made of three basic components. A one-piece stainless steel machined frame with deep ribs provides the main structure. Four (4) stainless steel feet are welded to the frame and provide the interface features for the lower core internals. A high strength stainless steel filter plate is fastened on the bottom face of the frame to provide high filtering efficiency.

Counter-bores in the top surface of the frame, and a leading edge bullet-nose feature in the middle of the frame, are designed to align with the fuel rod lower end caps. The fuel rod seats on the bottom of the counter-bore towards end of life, when the fuel assembly is most susceptible to grid-to-rod fretting. The seated fuel rod down in the counter-bore, in combination with the leading edge bullet nose, results in a streamlined flow through the nozzle, which minimizes the lower region turbulence. When the lower end cap is encapsulated within the counter-bore, it protects against excessive fuel rod vibration that could be caused by flow anomalies in the lower region.

The GRIP bottom nozzle lower connection incorporates a self-securing QD feature, allowing removal and replacement of the nozzle with no loose parts and no replacement hardware. The self-securing screws remain in the bottom nozzle during handling.

Figure 2-14 through Figure 2-16 highlight the GRIP bottom nozzle design features.

Figure 2-14: GRIP Bottom Nozzle

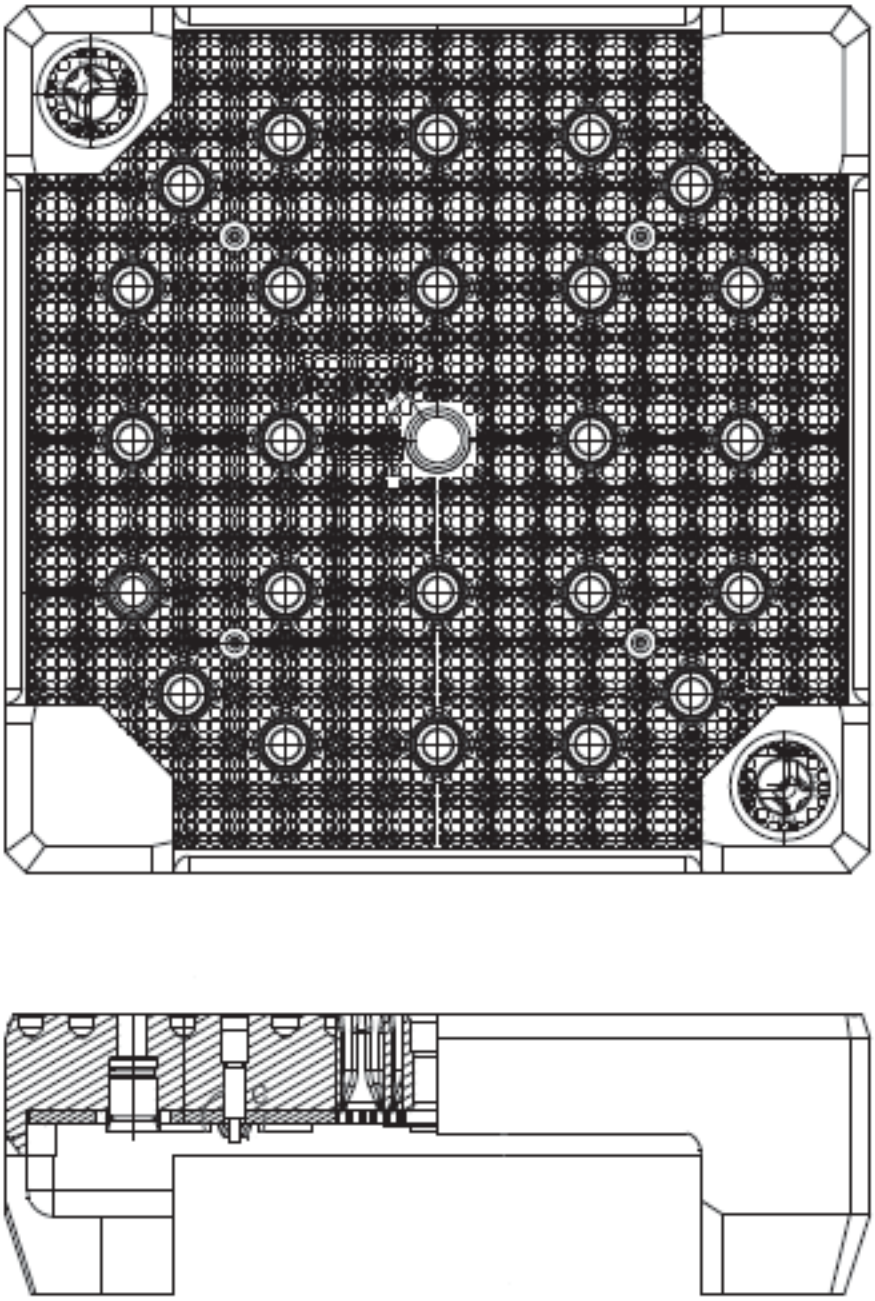


Figure 2-15: GRIP Bottom Nozzle Filter

Figure 2-16: GRIP Bottom Nozzle Connection

2.1.8. MONOBLOC Guide Tube and Instrument Tube

The GAIA fuel assembly design at Callaway Unit 1 incorporates twenty-four (24) MONOBLOC guide tubes and an instrument tube. The MONOBLOC guide tube Outer Diameter (OD) of [ ] is constant over the entire length, which results in additional material thickness and structural reinforcement in the dashpot region. This provides additional resistance to twist and bow at high burnups when spacer grid relaxation significantly reduces the coupling between the fuel rod and structural cage.

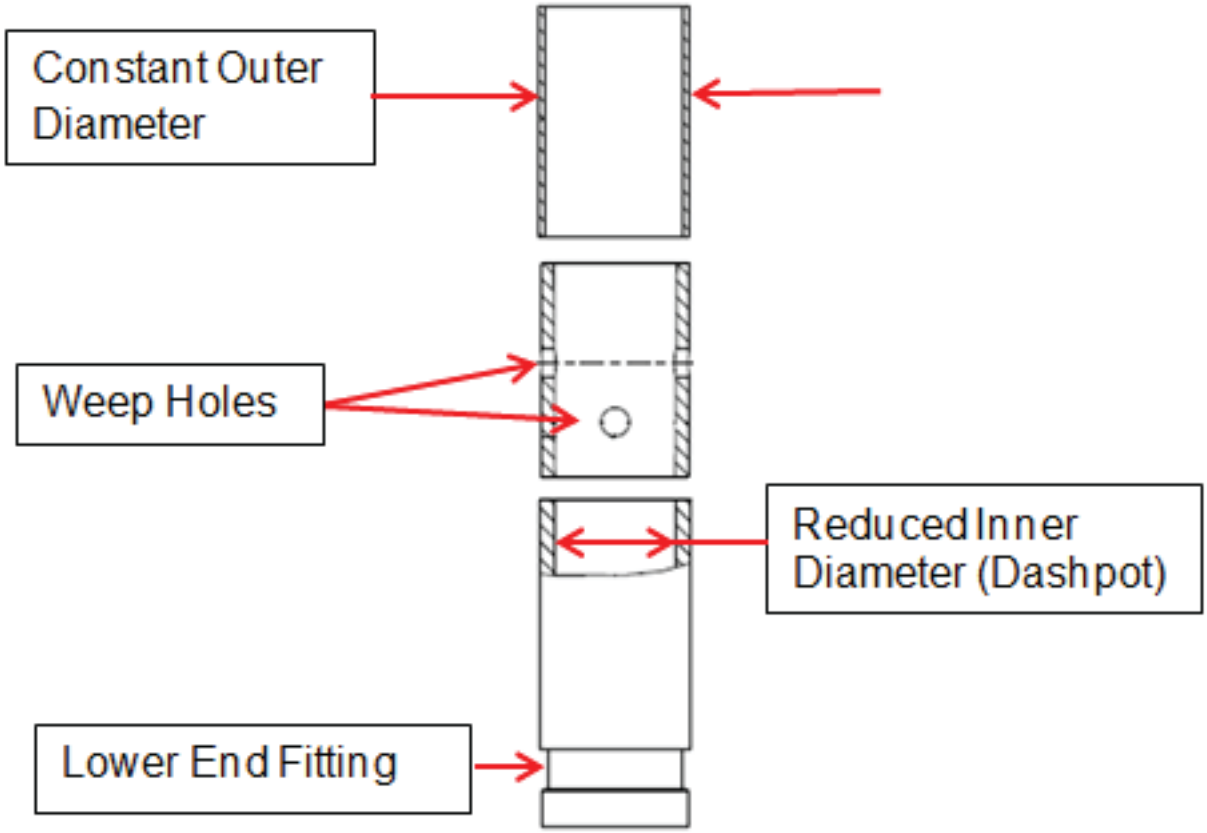
Four (4) weep holes in the dashpot region allow coolant outflow during Rod Cluster Control Assembly (RCCA) insertion and coolant inflow to control components during normal operation.

The MONOBLOC Guide Tube (GT) is constructed of Framatome's Q12 quaternary alloy which is approved for use as a structural material in Reference 4. Q12 is an evolutionary development based on Framatome's M5 metallurgy, with small amounts of tin and iron added. Tin improves resistance to creep, and its content is limited to prevent degradation of the corrosion kinetics. Iron, in combination with niobium, is a key element for ensuring good corrosion performance. For structural components, these modifications provide higher irradiation creep strength without compromising the corrosion resistance. Q12 is processed the same way as M5, resulting in a fully recrystallized microstructure with fine grains and uniformly distributed precipitates.

The Q12 instrument tube has a uniform Inner Diameter (ID) and OD. It is centrally located within the 17x17 array, extends the length of the fuel, and is fixed to the cage by resistance welds.

Figure 2-17 highlights the MONOBLOC guide tube design features.

Figure 2-17: MONOBLOC Guide Tube



2.1.9. Materials

Table 2-1 summarizes the materials utilized on the GAIA design at Callaway Unit 1, identifying the alloys and corresponding components.

Table 2-1: Materials Utilized on the GAIA Design

Material	Component
M5	Fuel Rod Cladding
	IGM Grids
	GAIA Structural Spacer Grids
Zircaloy-4	Quick Disconnect Sleeves
	Fuel Rod End Caps
	Guide Tube End Fittings
	Spacer Capture Rings
Q12	Guide Tubes
	Instrument Tube
Stainless Steel	Top and Bottom Nozzle Structures
	Lower Connection Screws
	Pin Holddown Springs
	Bottom Nozzle Filter Plate, Fastening Pins, Washers
Nickel Alloy	HMP End Spacer Grids
	Lower Connection Locking Rings
	Holddown Leaf Springs, Spring Screws
	Quick Disconnect Locking Springs, Rings, Lugs
	Fuel Rod Plenum Springs
UO_2, and Gd_2O_3-UO_2	Fuel pellets

2.2. *Operational Experience of Framatome GAIA Fuel Assemblies in Westinghouse 17x17 Reactors*

A summary of GAIA fuel assembly operating experience is provided in this section. Four (4) GAIA lead assemblies were inserted in the core of an international reactor in 2012. Eight (8) GAIA lead assemblies were inserted in the core of a USA reactor in 2015. Both the international and USA plants are Westinghouse 3-loop designs. Four (4) lead test assemblies were inserted in the core of another 4-loop USA reactor in 2019. Four (4) lead assemblies were inserted in Callaway Unit 1 in 2020. To date, GAIA fuel assemblies have not had any fuel rod leakers.

The four (4) GAIA lead assemblies, inserted in 2012, in the international reactor successfully completed four (4) 12-month cycles of leaker-free performance, and two (2) assemblies successfully completed a fifth 12-month cycle with leaker-free performance.

The eight (8) lead assemblies in the 3-loop USA reactor successfully completed two (2) 18-month cycles, and four (4) assemblies completed a third 18-month cycle, with leaker-free performance. The four (4) lead assemblies in the 4-loop USA reactor (inserted in 2019) are currently in their second cycle, and the four (4) lead assemblies in Callaway Unit 1 are currently in their first cycle of operation.

In the USA, GAIA fuel assemblies in batch quantity are currently in their first cycle of operation in a Westinghouse 3-loop plant. In Europe, GAIA fuel assemblies in batch quantity are in their second cycle of operation at two (2) Westinghouse 3-loop plants.

[

]

[

]

2.3. *Mechanical Compatibility*

Framatome and Ameren have performed an extensive review of the interfaces between the GAIA fuel assembly design and the core internals, control components (including RCCAs), co-resident fuel assemblies, in-core instrumentation, and handling equipment. Table 2-2 shows a comparison of the major dimensions of the Callaway Unit 1 Framatome fuel design and the Callaway Unit 1 co-resident fuel design.

Table 2-2: Comparison of Nominal Mechanical Design Features

Feature	Framatome Fuel Design	W Fuel Design (Co-Resident)
Fuel Assembly Overall Length, inch	159.86	159.975
Bundle Pitch, inch	8.466	8.466
Number of Bundles in Core	193	193
Fuel Rod Overall Length, inch	151.89	152.870 (Performance+) or 152.490 (Vantage+)
Fuel Rod Pitch, inch	0.496	0.496
Number of Fuel Rods / Assembly	264	264
Number of Guide Tubes / Assembly	24	24
Number of Instrumentation Tubes / Assembly	1	1
Fuel Rod Cladding OD, inch	0.374	0.360
Fuel Rod Cladding Thickness, inch	0.0225	0.0225
Fuel Pellet Diameter, inch	0.3225	0.3088
Active Fuel Stack Length (BOL, cold), inch	144	144
Axial Blanket Length (top & bottom), inch	6.00 (UO_2 rods) 10.00 (GAD rods)	6.00 (UO_2 rods)
Burnable Poison Material	Gd_2O_3 - UO_2	UO_2 with ZRB2 coating
Fuel Rod Cladding Material	M5	ZIRLO®[1]
Guide Tube Material	Q12	Zircaloy-4 or ZIRLO®[1]
Instrument Tube Material	Q12	Zircaloy-4
Number of Structural Grids	6	6
Number of Intermediate Mixing Grids	3	3
Bottom Grid	Alloy 718 HMP	Inconel-718
Top Grid	Alloy 718 HMP	Inconel-718

[1] ZIRLO is a registered trademark of the Westinghouse Electric Company.

2.3.1. Fuel Assembly

The GAIA fuel assembly overall length is confirmed to be compatible with the dimensions of the core internals (spacing between core support plates and fuel alignment pins) throughout life at hot and cold conditions. Additionally, the elevation of the control component seating surface on GAIA is the same as that of the co-resident fuel. The fuel handling equipment will access GAIA fuel in a nearly equivalent axial window as the co-resident fuel. The allowable length of travel for RCCAs is greater in GAIA than co-resident fuel. An axial growth analysis confirms adequate clearances exist between the fuel assembly and core internals, as well as the fuel rod to fuel assembly top nozzle, up to the licensed fuel rod and fuel assembly burnup limits.

The fuel assembly array type, number of fuel rods and guide tubes, fuel assembly pitch, and fuel rod pitch dimensions are the same as for the co-resident fuel.

These evaluations demonstrate the GAIA fuel design is compatible with the reactor internals, control components (including RCCAs) and co-resident fuel in the core.

The following sections contain additional evaluations of individual fuel assembly components, including the Top Nozzle, the Bottom Nozzle, and the Guide Tubes.

2.3.2. Top Nozzle

The mechanical compatibility of the top nozzle is explicitly evaluated because it interfaces with the reactor core guide pins, interfaces with all the fuel handling grapples when moving the fuel assembly, and interfaces with the control components, including neutron sources, thimble plugs and RCCAs.

The following top nozzle compatibility interfaces are confirmed: core pin engagement throughout life, holddown spring contact throughout life, top nozzle envelope consistent with co-resident fuel design, fuel handling equipment fit, and control components, neutron source assemblies, and thimble plug assemblies fit.

The top nozzle evaluations show the top nozzle is mechanically compatible.

2.3.3. Bottom Nozzle

The bottom nozzle requires compatibility evaluations because it mates with the features of the lower core support plate. The bottom nozzle compatibility interfaces are confirmed for lower core support plate guide pin engagement throughout life, consistent envelope with the co-resident fuel design, and auxiliary equipment (e.g., lead-in shoehorns).

Although bottom nozzle lead-in features are not available to Framatome for co-resident fuel, GAIA instrument tube ID is larger than the co-resident fuel, indicating sufficient clearance for instrumentation. In addition, based on proven operating experience with other GAIA LTA programs and similar Framatome bottom nozzle geometries, the lead-in interface of the GRIP bottom nozzle is determined to be adequate.

All evaluations show the bottom nozzle is compatible.

2.3.4. Guide Tubes

Besides being the structural components of the fuel assembly, the guide tubes interface with the control rods. Table 2-3 shows a comparison of guide tube dimensions between the GAIA and the co-resident fuel designs. GAIA guide tubes maintain the same Beginning of Life (BOL), cold inner diameter in the dashpot as the co-resident fuel. The GAIA guide tube inner diameter in the non-dashpot region is larger than the co-resident fuel guide tube ID, thus having a larger radial gap with the control rods. Weep hole elevation compatibility is confirmed to be consistent enough with co-resident fuel to prevent unexpected drop time differences or affect GT boiling. The dashpot location is comparable between GAIA and co-resident fuel design, which is another indication that RCCA drop time for GAIA will be comparable to co-resident fuel. RCCA drop times are further addressed in the Thermal Hydraulics analyses.

All evaluations demonstrate guide tube compatibility is ensured.

Table 2-3: Comparison of Nominal GAIA and Co-Resident Fuel Guide Tube Features

Feature	Framatome Fuel Design	W Fuel Design (Co-Resident)
Guide tube ID (non-dashpot region), inch	0.451	0.442
Guide tube ID (dashpot region), inch	0.397	0.397
Guide tube OD (non-dashpot region), inch	[ ]	0.474
Guide tube OD (dashpot region), inch	[ ]	0.430
Weep hole diameter, inch	0.094	≤ 0.094

2.4. *Mechanical Design Evaluations*

2.4.1. Description

The mechanical design evaluations are performed using the USNRC-approved design methods and are evaluated to the USNRC-approved design criteria (Reference 1). These criteria are consistent with the specified acceptable fuel design limits (SAFDLs) identified in Chapter 4.2 of the Standard Review Plan (Reference 2).

The fuel analyses are broadly separated into fuel rod analyses and structural analyses. Consistent with NUREG-0800 Chapter 4.2 (Reference 2), the following fuel assembly mechanical and fuel rod thermal-mechanical analyses were considered for the mechanical design evaluation.

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

- Lift-Off

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

The GAIA Fuel Assembly Mechanical Design Topical Report (Reference 1) is the governing document for the mechanical and thermal-mechanical analyses, including the associated criteria (SAFDLs) and methods. The fuel assembly mechanical compatibility analysis is governed by internal Framatome requirements. Table 2-4 provides a summary of the major methods used to analyze the GAIA fuel assemblies in Callaway Unit 1.

Table 2-4: Summary of Mechanical Methods

Analysis	Method
Normal Operation Component Stress and Load Limits	ANP-10342P-A (Ref. 1)
Faulted Component Stress and Load Limits	ANP-10342P-A (Ref. 1) ANP-10337P-A (Ref. 3)
Fretting Wear	ANP-10342P-A (Ref. 1)
Fuel Assembly / Fuel Rod Growth	ANP-10342P-A (Ref. 1) ANP-10334P-A (Ref. 4) BAW-10240P-A (Ref. 5)
Fuel Assembly Bow	ANP-10342P-A (Ref. 1)
Fuel Assembly Lift-Off	ANP-10342P-A (Ref. 1)
Fuel Rod Fatigue	ANP-10342P-A (Ref. 1) BAW-10227P-A (Ref. 6) BAW-10231P-A (Ref. 7)
Fuel Rod Oxidation	ANP-10342P-A (Ref. 1) BAW-10231P-A (Ref. 7)
Fuel Rod Internal Pressure	ANP-10342P-A (Ref. 1) BAW-10231P-A (Ref. 7) BAW-10183P-A (Ref. 8)
Fuel Rod Internal Hydriding	ANP-10342P-A (Ref. 1)
Fuel Rod Creep Collapse	ANP-10342P-A (Ref. 1) BAW-10227P-A (Ref. 6) BAW-10231P-A (Ref. 7) BAW-10084P-A (Ref. 9)
Fuel Centerline Melt	ANP-10342P-A (Ref. 1) BAW-10231P-A (Ref. 7) EMF-92-081(P)(A) (Ref. 10)
Fuel Rod Transient Cladding Strain	ANP-10342P-A (Ref. 1) BAW-10231P-A (Ref. 7)
Fuel Rod Stress and Buckling	ANP-10342P-A (Ref. 1) BAW-10227P-A (Ref. 6)

2.4.2. Input Parameters and Assumptions

The input parameters used to perform the mechanical analyses include fuel design information derived from design documents, fuel assembly and component characteristics established by mechanical / hydraulic testing, plant parameters provided by Ameren, fatigue duty cycles created using the fatigue transients provided in the Callaway Unit 1 Final Safety Analysis Report Standard Plant (FSAR SP), and fuel rod power histories generated for the representative core designs by Framatome. For the cycles analyzed for this amendment request, the power histories were created using expected typical cycle core designs projected to the design life of the fuel. These cycle designs were created using USNRC-approved codes and methods.

2.4.3. Fuel Assembly Structural Analyses

The fuel assembly structural analysis is separated into normal operating analysis, shipping and handling analysis, and faulted condition analysis. The normal operating analysis evaluates the fuel assembly stress state during start-up, steady state operation, shutdown, and Anticipated Operational Occurrences (AOO's) and compares it with the criteria established in USNRC-approved ANP-10342P-A (Reference 1). The shipping and handling analysis evaluates the fuel assembly against handling limits established in Reference 1 and shipping load limits established in Framatome shipping specifications. The Safety Evaluation Report (SER) for Reference 1 approves the usage of this Topical Report for GAIA licensing applications up to 62 GWd/MTU maximum UO_2 fuel rod burnups with four (4) limitations and conditions that are addressed in this report.

The faulted condition analysis evaluates the structural response of the fuel assembly to externally applied forces such as earthquakes and postulated pipe breaks in the reactor coolant system. The faulted condition analysis is performed in accordance with the USNRC-approved methodology and criteria established in ANP-10337P-A (Reference 3). This methodology has been approved for use with the GAIA fuel design as discussed in ANP-10342P-A (Reference 1). The limitations and conditions outlined in the SER for ANP-10337P-A are addressed in this report.

2.4.3.1. GAIA Mechanical Topical Report Restrictions

The limitations and conditions placed on the use of Reference 1 originate from Section 4.0 of the SER and are presented in Table 2-5 along with method of adherence.

Table 2-5: GAIA Mechanical Design Topical Report Limitations and Conditions

Limitation or Condition	Method of Adherence
This GAIA fuel assembly design is approved for use with low enrichment uranium (LEU) fuel, which has been enriched to less than or equal to 5 percent.	This condition is met on a cycle-to-cycle basis through core design where the maximum enrichment is limited to 5 percent.
This GAIA fuel assembly design is licensed for a maximum fuel rod burnup of 62,000 Megawatt-days/metric ton of Uranium.	The cycle specific core design and fuel rod analyses ensure the maximum fuel rod burnup of 62,000 MWd/MTU is not exceeded.
The final LTA program PIE report shall be submitted to NRC staff prior to any reload batch GAIA assemblies reach the third cycle of operation.	The GAIA LTA PIE Report (Reference 11), was submitted to the NRC in July 2021.
As part of the plant-specific LAR implementing GAIA, the licensee must demonstrate acceptable performance of GAIA under RIA conditions, including fuel damage, coolable geometry, and radiological consequences, using approved methods. Current guidance and analytical limits are found in SRP 4.2 Appendix B. Newer guidance is expected soon (e.g., DG-1327). The licensee should consider the most up-to-date guidance and analytical limits at the time of submittal. Alternative means to demonstrate compliance will be considered on a case-by-case basis.	An analysis is performed that considers the requirements of RG 1.236. This analysis demonstrates that GAIA fuel assemblies meet the criteria specified in RG 1.236 (e.g., enthalpy rise within the fuel rod, centerline fuel temperature, rim fuel temperature, and departure from nucleate boiling) for allowed fuel damage associated with an ejected control rod event.

2.4.3.2. Additional Discussion of the Faulted Condition Analysis

The faulted loads calculation and component qualification for service is performed using the methods and techniques in the USNRC-approved topical report ANP-10337P-A (Reference 3). The analysis is performed based on structural models obtained by benchmarking to tests performed on prototypical fuel assemblies and components. The tests performed on the full fuel assemblies provide the main dynamic characteristics of the GAIA fuel assembly for lateral and vertical models. The tests on spacer grid and other components provide both dynamic and strength characteristics that are used in both model definition and margin calculation. The GAIA grid strength is defined in a very conservative manner, based on the range of linear behavior of this grid (Limitations and Conditions (L&C) 1 in Section 2.4.3.3), and provides substantial margin for control rod insertion as measured by the grid permanent deformation (L&C 2.b in Section 2.4.3.3). 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.

One important issue addressed in the ANP-10337P-A (Reference 3) methodology and reflected in the Callaway Unit 1 GAIA fuel assessment is the treatment of the USNRC Information Notice IN-2012-09, "Irradiation Effects on Spacer Grid Crush Strength," (Reference 13). IN-2012-09 requires 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 this end, two sets of fuel assemblies and spacer grids conditioned for BOL and simulated EOL respectively (in accordance with the methodology of ANP-10337P-A) have been tested, and two sets of models, BOL and EOL have been developed and analyzed, with separate margins reported for each case. The EOL condition [ ] in the core row models. The spacer grid impact load margins are decided based on the maximum load for all core configurations. The SSE impacts are Square Root of Sum of Squares (SRSS) combined with the LOCA impacts.

The guide tube stress margins are analyzed based on [ ] model. The loads are calculated by [ ]. Therefore, a 3-D load combination per the requirements of NRC Regulatory Guide 1.92 (Reference 14) is implemented. The lateral deflections are extracted from the core row models by using the position in the core row in the two lateral directions of each core location.

Two aspects of this analysis require further detailed discussion:

1. The dynamic characteristics [ ] row models.
2. The [ ] events.

2.4.3.2.1. Dynamic Characteristics [ ]

Since precise values for the fuel assembly natural frequencies and spacer grid dynamic characteristics of [

].

[ ]

For operating temperature conditions, the [ ] models, initially benchmarked to the ambient temperature frequency ranges, have been converted analytically [

].

For the spacer grids, [

].

[

]

1. [

]. The design margins on the main GAIA grids are the dominant criterion.

2. [ ].

3. [ ]. The effect of these parameters on the grid impact margins is negligible.

4. [ ].

[ ], and the subsequent analysis is conservative.

2.4.3.2.2. Seismic and LOCA Time History Generation

The faulted analysis involves the solving of the non-linear equations of motion of the core row models under imposed displacement, velocity, and acceleration boundary conditions at the interfaces between the fuel assemblies and the reactor internals (lower plate, upper core plate and baffle plate elevations corresponding to the fuel assembly grid elevations). The boundary condition displacement and velocity time histories are input explicitly, while the acceleration is computed internally by the Framatome proprietary structural dynamic code CASAC (Reference 3).

For the lateral seismic cases, Framatome generates a corresponding set of [

] thus ensuring a conservative level of seismic and LOCA loading.

Because the motions of [

] ensure the conservative character of these time histories.

For the lateral LOCA case, time histories are generated [

] with substantial margins.

The vertical seismic cases are very similar in approach [

].

The vertical LOCA events have [

] in the faulted analysis of the GAIA fuel.

The [

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

2.4.3.3. GAIA Externally Applied Dynamic Excitations Topical Report Restrictions

The limitations and conditions placed on the use of Reference 3 are from Section 5.0 of the SER. They are shown as follows, along with method of adherence.

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

[ ]

Response 1: *Framatome has conducted the necessary dynamic grid crush testing to demonstrate the behavior defined in items a., b., and c. above.*

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: *Framatome has defined allowable spacer grid deformation limits that are in accordance with items a. and b. above.*

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 ANP-10337P (as updated by RAIs) 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 ANP-10337P would not be considered to constitute a departure from a method of evaluation in the safety analysis. Such changes may be used in licensing calculations without NRC staff review and approval. However, all code changes must be documented in an auditable manner to meet the quality assurance requirements of 10 CFR Part 50, Appendix B.

   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 for the intended application of the code.

   Response 3: *For the Callaway Unit 1 faulted analysis, Framatome has used CASAC exclusively. 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 Callaway Unit 1 reactor is part of the "current fleet" of PWR reactors in place at the time of approval of ANP-10337P.*

5. 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 application of the generic damping values from ANP-10337P falls within the range of intended application. Furthermore, the application of the damping values indicated in ANP-10337P-A to the GAIA fuel design is approved in ANP-10342P-A (Reference 1).*

6. The 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 performance of the Callaway Unit 1 GAIA fuel rods is evaluated in the same manner as demonstrated in the sample problem for ANP-10337P-A (Reference 3).*

7. As indicated in 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 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 ANP-10337P-A (Reference 3).*

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 analysis performed under ANP-10337P-A (Reference 3) is performed in accordance with RG 1.92 and combines load based on three-orthogonal components.*

9. **[**

].

Response 9: *The grid impact loads predicted by Framatome for the Callaway Unit 1 GAIA fuel design* **[** **]**.

2.4.4. Fuel Rod Analyses

The fuel rod thermal-mechanical performance evaluations are performed using USNRC-approved codes and methodologies as defined in the GAIA Fuel Assembly Mechanical Design Topical Report (Reference 1). The fuel rod analyses evaluate all SAFDLs listed in Section 2.4.1 and are performed to demonstrate that the fuel rod design criteria defined within Reference 1 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. Since these evaluations are dependent on the rod power and core operating parameters, the approved Framatome methodology requires these analyses to be verified and/or re-performed on a cycle-specific basis to ensure the actual cycle design will not result in SAFDL non-compliance. The LAR representative core designs are analyzed to demonstrate the fuel design is acceptable and provide results showing SAFDL compliance. All fuel rod analyses are based on inputs which either represent or bound operation at Callaway Unit 1.

2.4.4.1. Cladding Fatigue

The design criterion for cladding fatigue is defined in Section 8.1.2.1 of Reference 1:

For M5 fuel rod cladding, the CUF shall be less than 0.9.

The fuel rod fatigue evaluation assesses the fatigue cumulative usage factor (CUF) for the M5 cladding and confirms it is below the limit of 0.9. Calculations are performed using the USNRC-approved COPERNIC fuel rod analysis code (Reference 7) and the general methodology of Reference 6.

2.4.4.2. Cladding Oxidation

The design criterion for cladding oxidation is defined in Section 8.1.4.1 of Reference 1:

Cladding peak oxide thickness shall not exceed a best-estimate predicted value of 100 microns.

The fuel rod oxidation evaluation assesses the cladding corrosion level for the M5 cladding and confirms it is below the limit of 100 microns. Calculations are performed using the USNRC-approved COPERNIC fuel rod analysis code (Reference 7).

[

] While an explicit chemistry assessment would be performed as part of a fuel transition to assess the potential impact of lithium on cladding corrosion, based on operating experience and the calculated design margin for cladding oxidation ([ ]), it can be concluded that Callaway's unique chemistry (elevated lithium concentrations) does not challenge the conclusions above regarding satisfaction of the cladding oxidation design criterion.

2.4.4.3. Internal Pin Pressure

The design criterion for internal pin pressure is defined in Section 8.1.6.1 of Reference 1:

Internal gas pressure of the peak fuel rod in the reactor will be limited to a value below that which would cause (1) the fuel-cladding gap to increase due to outward cladding creep during steady-state operation or (2) reorientation of the hydrides in the radial direction in the cladding.

Per Reference 8, the explicit criteria adopted to satisfy the criterion above are [

].

The fuel rod internal pressure evaluation confirms the maximum internal gas pressure is below the limit of [

]. Calculations are performed using the USNRC-approved COPERNIC fuel rod analysis code (Reference 7). Consistent with Reference 1, DNB propagation is addressed and considered within the plant-specific thermal-hydraulic analyses.

2.4.4.4. Internal Hydriding

The design criterion for internal hydriding is defined in Section 8.2.1.1 of Reference 1:

Internal hydriding shall be precluded by appropriate manufacturing controls.

Fuel rod internal hydriding is precluded by imposing tight controls on hydrogen impurities during fuel rod fabrication and on the fuel rod components, including careful moisture control of the fuel pellets. UO_2 fuel pellets have a total hydrogen content [

].

2.4.4.5. Cladding Creep Collapse

The design criterion for cladding creep collapse is defined in Section 8.2.2.1 of Reference 1:

Predicted creep collapse life of the fuel rod must exceed the maximum expected in-core life.

The fuel rod creep collapse evaluation assesses the predicted creep collapse life of the M5 cladding and confirms it exceeds the maximum expected in-core life. Calculations are performed using the USNRC-approved COPERNIC fuel rod analysis code (Reference 7) and the CROV creep collapse method (Reference 9). The applicability of the CROV method is extended to M5 cladding material through Reference 6.

2.4.4.6. Fuel Centerline Melt

The design criterion for fuel centerline melt is defined in Section 8.2.3.1 of Reference 1:

Fuel melting during normal operation and AOO's is precluded.

The fuel centerline melt (FCM) evaluation assesses normal operation conditions and AOO's. The FCM linear heat rate limits are calculated using the USNRC-approved COPERNIC fuel rod analysis code (Reference 7).

Verification of the FCM limits calculated by COPERNIC is performed in accordance with the USNRC-approved setpoint and transient analysis methods described in Reference 10. The FCM evaluation confirms that there is at least a 95% probability at 95% confidence that fuel melting does not occur during normal operation conditions and AOO's in Section 5.5.5 and 5.5.6.2.

2.4.4.7. Transient Cladding Strain

The design criterion for transient cladding strain is defined in Section 8.1.1.1 of Reference 1:

Maximum uniform hoop strain (elastic plus plastic) shall not exceed 1%.

The transient cladding strain (TCS) evaluation assesses the M5 cladding performance and confirms that the maximum uniform hoop strain (elastic plus plastic) does not exceed 1%. The TCS linear heat rate limits are calculated using the USNRC-approved COPERNIC fuel rod analysis code (Reference 7).

Verification of the TCS limits calculated by COPERNIC is performed using [ ]. The peak linear heat rates for these [ ] are compared to the COPERNIC TCS limits to confirm that the 1% cladding strain limit is not exceeded during AOO events.

2.4.4.8. Cladding Stress and Buckling

The design criterion for normal operation cladding stress is defined in Section 8.1.1.1 of Reference 1:

[ ]

The design criterion for faulted condition cladding stress resulting from SSE and LOCA events is defined in Section 8.3.1.1 of Reference 1:

Fuel rods must be protected against mechanical fracturing. M5 fuel rod cladding stress criteria are in accordance with those defined in Section 8.1.1.1 of Reference 1.

The fuel rod stress evaluations assess the M5 cladding performance under normal operation and faulted conditions and confirm that the stresses are less than the defined stress limits from Section 8.1.1.1 of Reference 1.

In accordance with Section 8.1.1.2 of Reference 1, steady-state cladding buckling is analyzed with the USNRC-approved method described in Section 3.3 of Reference 6. The fuel rod buckling evaluations assess the M5 cladding performance and confirm that the buckling loads do not exceed [

].

2.4.5. Mechanical Analyses Results

The generic criteria (SAFDLs) for the fuel rod and fuel assembly are listed in Table 2-6 along with the corresponding section number from the criteria topical report (Reference 1) and the corresponding results. The LAR representative cycles are analyzed to demonstrate that the fuel design is acceptable and provides typical results showing SAFDL compliance. The specific reload results could be slightly different but will continue to show SAFDL compliance.

Table 2-6: Fuel Mechanical Design Evaluation Results

Criteria Section	Description	Criteria	Results
8.1	Fuel System Damage		
8.1.1	Stress, Strain, Loading Limits	Stress and Load Limits: Stresses and/or loads associated with normal operation, AOO's, shipping, and handling shall be less than limits based on Section III of the ASME Code for all components, unless otherwise specified. • Based on the ASME code, the basic component stress criteria (other than fuel rod cladding) are as follows. - Primary membrane stress intensity, Pm < Sm - Primary membrane plus bending stress intensity, Pm + Pb < 1.5 Sm - Primary and secondary membrane plus bending stress intensity, Pm + Pb + Q < 3Sm Where: Sm = min (1/3Su, 2/3Sy) at the applicable temperature • [ ] : [ ] M5 Fuel Rod Cladding Strain: Maximum uniform hoop strain (elastic plus plastic) shall not exceed 1%.	Guide Tube: Limiting Margin: Normal operation + OBE [ ] Spacer Grid: Limiting Margin: Normal operation + OBE bounded by shipping with margin [ ] . Top Nozzle: Limiting Margin: Normal operation bounded by shipping with margin [ ]. Bottom Nozzle: Limiting Margin: Normal operation bounded by shipping with margin [ ]. M5 Fuel Rod Cladding Stress: Analysis results demonstrate positive margins to the criteria. M5 Fuel Rod Cladding Strain: Calculated transient cladding strain LHGR limits were verified such that maximum uniform hoop strain does not exceed 1%.

Criteria Section	Description	Criteria	Results
8.1.2	Strain Fatigue	For all components other than fuel rod cladding, the CUF shall be less than 1.0. For M5 fuel rod cladding, the CUF shall be less than 0.9.	Guide Tube and Holddown Spring Fatigue: GT CUF **[ ]** Holddown Spring CUF **[ ]** M5 Fuel Rod Cladding Fatigue: The fuel rod cladding cumulative usage factors (UO_2 and gadolinia) are less than the M5 CUF limit.
8.1.3	Fretting Wear	Fuel rod failures due to fretting shall not occur.	Supported by fretting test, evaluation, and operational experience.
8.1.4	Oxidation, Hydriding, and Crud	Cladding peak oxide thickness shall not exceed a best-estimate predicted value of 100 microns.	Maximum best estimate oxide is less than 100 microns. Approved fuel rod performance code accounts for oxidation and crud buildup. Metal loss is accounted for in stress analysis.
8.1.5	Fuel Rod / Fuel Assembly Bow and Growth	Fuel Rod Bow: There is no explicit design criterion for fuel rod bow. Departure from nucleate boiling ratio (DNBR) and linear heat generating rate (LHGR) burnup thresholds and penalties are calculated and considered on a cycle by cycle basis to address the thermal hydraulic limits established in SRP 4.4. Fuel Rod and Fuel Assembly Growth: Fuel rod irradiation growth is addressed by requiring clearance between the fuel rod and nozzles at EOL. Fuel assembly irradiation growth is addressed by requiring clearance between the fuel assembly and reactor core plates at EOL.	Fuel Rod Bow: Fuel rod bow is addressed in Section 5.5.3. Fuel Rod and Fuel Assembly Growth: Clearance between the fuel rods and nozzles and between the fuel assembly and reactor core plates is maintained through EOL up to a maximum fuel rod burnup of 62 GWd/MTU and a maximum fuel assembly burnup of 60 GWd/MTU. Criteria are met throughout design life.

Criteria Section	Description	Criteria	Results
8.1.6	Fuel Rod Internal Pressure	Internal gas pressure of the peak fuel rod in the reactor will be limited to a value below that which would cause (1) the fuel-cladding gap to increase due to outward cladding creep during steady-state operation or (2) reorientation of the hydrides in the radial direction in the cladding.	Acceptable maximum internal rod pressure. Internal rod pressure does not exceed [ ], the pellet-to-clad gap criterion is met. DNB propagation is addressed as part of the plant-specific thermal-hydraulic analyses.
8.1.7	Fuel Assembly Lift-Off	During normal operation conditions and AOO's (with the exception of a pump overspeed transient), the holddown springs shall maintain fuel assembly contact with the lower support plate. Assuming a pump over-speed transient, fuel assembly lift-off can occur, but the fuel assembly top and bottom nozzles shall maintain engagement with reactor internal pins and the holddown springs shall maintain positive holddown margin after the event.	All criteria for fuel assembly lift-off are met, including demonstration of the criteria for pump over speed (POS) transient at 120% flow are met. The assembly will not come off the lower core pins during a POS transient and positive holddown margin is maintained after the event.
8.2	Fuel Rod Failure		
8.2.1	Hydriding	Internal hydriding shall be precluded by appropriate manufacturing controls.	Controlled by manufacturing specifications and verified by Quality Control inspection.
8.2.2	Cladding Collapse	Predicted creep collapse life of the fuel rod must exceed the maximum expected in-core life.	Analysis results demonstrate positive margins to the collapse criteria.
8.2.3	Overheating of Fuel Pellets	Fuel melting during normal operation and AOO's is precluded.	Sections 5.5.5 and 5.5.6.2 demonstrate that the acceptance criteria are met.
8.3	Fuel Coolability		

Criteria Section	Description	Criteria	Results
8.3.1	Structural Deformation	OBE Spacer Grid Acceptance Criteria Spacer grid deformation experienced during an OBE event should not exceed the magnitude of the tolerance band to which the grid was designed. SSE/LOCA Spacer Grid Acceptance Criteria Spacer grid deformation experienced during an SSE/LOCA event [ ] . [ ] . SSE/LOCA Fuel Rod Acceptance Criteria Fuel rods must be protected against mechanical fracturing. M5 fuel rod cladding stress criteria are in accordance with those defined in Section 8.1.1.1 of Reference 1. SSE/LOCA Guide Tube Acceptance Criteria Sudden and severe changes in the geometry of the guide tube (e.g., local collapse or plastic hinge) shall not occur. This acceptance criterion is further delineated by requiring that (1) stresses do not exceed a limit prohibiting local collapse of the guide tube, and (2) the structural stability of the guide tube must be maintained.	Guide Tube: Limiting Margin: SSE + LOCA [ ] Spacer Grids: Limiting Margin: ISG SSE + LOCA [ ] [ ] . IGM SSE + LOCA [ ] Top Nozzle: Limiting Margin: SSE + LOCA [ ] Bottom Nozzle: Limiting Margin: SSE + LOCA [ ] SSE/LOCA Fuel Rod: Analysis results demonstrate positive margins to the criteria.
8.4	Additional Acceptance Criteria		
8.4.1	Overheating of Cladding	The criteria for departure from nucleate boiling are included in the NRC-approved critical heat flux (CHF) correlation topical report for use with the GAIA fuel assembly.	Reference 35, Section 3.9 demonstrates that the acceptance criteria are met.

Criteria Section	Description	Criteria	Results
8.4.2	Excessive Fuel Enthalpy	The criteria for excessive fuel enthalpy during a reactivity initiated accident are included in the NRC-approved control rod ejection methods.	Section 6.1.1.4 demonstrates that this acceptance criteria is met.
8.4.3	Bursting	Cladding swelling and rupture requirements are included in the NRC-approved loss-of-coolant accident (LOCA) evaluation models.	The impact of flow blockage effects via clad swell and rupture model is considered in the LOCA analysis and is reflected in the LOCA Summary Reports (See Section 6.2).
8.4.4	Cladding Embrittlement	The criteria for cladding embrittlement during a LOCA are included in the NRC-approved LOCA evaluation methods.	Cladding embrittlement is satisfied by meeting 10 CFR 50.46(b) acceptance criteria (See Section 6.2).
8.4.5	Violent Expulsion of Fuel	The criteria for violent expulsion of fuel during a reactivity initiated accident are included in the NRC-approved control rod ejection methods.	Section 6.1.1.4 demonstrates that this acceptance criteria is met.
8.4.6	Fuel Rod Ballooning	Fuel rod ballooning requirements are included in the NRC-approved LOCA methods.	The impact of flow blockage effects via clad swell and rupture model is considered in the LOCA analysis and is reflected in the LOCA Summary Reports (See Section 6.2).
8.4.7	Reactivity Coefficients	The Doppler coefficient shall be negative at all operating conditions. The power coefficient shall be negative at all operating power levels relative to hot zero power.	Table 3-1 demonstrates that the acceptance criteria are met.

2.5. *Reconstitutable Fuel*

Framatome's NRC-approved reconstitution topical report (Reference 12) is applicable to Framatome PWR fuel designed to the NRC-approved GAIA topical report (Reference 1). To ensure applicability, application of Reference 12 must be within the following stated limits in the SER:

- Application is intended for reconstituted fuel assemblies (as opposed to fresh fuel assemblies).
- The PWR replacement rods must be comprised of inert zircaloy or stainless-steel slugs housed in standard fuel cladding.
- Reconstituted fuel is limited to two inert replacement rods or one inert rod and a guide tube per subchannel and a maximum limit on the total number of replacement rods per PWR assembly of 26%. The exception to this is for assemblies located on the outer edge of the core where outer rods are used for shielding.

2.6. *Mechanical Design Conclusion*

The Framatome GAIA fuel design is mechanically compatible with the core internals, control components, co-resident fuel assemblies, in-core instrumentation, handling equipment, and RCCAs at Callaway Unit 1. The Framatome GAIA fuel design is analyzed in accordance with USNRC-approved mechanical design criteria using the most limiting of mixed core and full core cycle inputs. All design criteria are met up to the licensed peak UO_2 fuel rod burnup of 62 GWd/MTU and peak Gadolinia fuel rod burnup of 55 GWd/MTU under normal and faulted operating conditions.

3. NUCLEAR DESIGN

3.1. *Description*

The licensing basis for the reload core nuclear design is defined in FSAR SP Section 4.3. The purpose of the core analysis is to verify that the cycle-specific reload design and the key safety parameters are properly addressed in the reload analysis. The effects of transitioning from the co-resident fuel to the Framatome GAIA fuel on the nuclear design bases and methodologies for Callaway Unit 1 are evaluated in this section.

3.2. *Input Parameters*

The Framatome GAIA fuel differs from that of co-resident fuel design as described in Sections 2.1 and 2.3. A discussion of the application of a mixed core penalty to the departure from nucleate boiling (DNBR) safety limits is provided in Reference 35 Section 3.9.

3.3. *Methodology*

The nuclear design codes are based on the ARCADIA code system for the submittal cycles, including the transition cycles, and future operation of Framatome GAIA fuel at Callaway Unit 1. References 17 and 18 are the USNRC-approved ARCADIA Topical Reports outlining the approved Framatome neutronics methodology and codes.

The L&C section of Reference 18 contains 6 L&Cs, with the third L&C that was originally presented in Reference 17 removed. L&C No. 1 restricts the applicability of the ARCADIA methodology to the fuel types discussed in the topical report unless additional analysis and benchmarking is conducted. The GAIA fuel design is equivalent to other fuel designs considered in References 17 and 18. It is a 17x17 assembly design with Zirconium based cladding material loaded with UO_2 pellets consistent with those evaluated in the References 17 and 18. The burnable poison used in GAIA is gadolinia that is mixed in with UO_2 fuel, which is also consistent with the fuel designs evaluated in References 17 and 18. The mechanical aspects of the GAIA design are also consistent with designs of other assemblies evaluated in References 17 and 18. GAIA performs similarly to other fuel designs (e.g., Framatome's Advanced W17 HTP fuel design). All materials and characteristics of the GAIA fuel do not affect the ability to model this assembly with the ARCADIA code system.

L&C No. 2 for Reference 18 requires that Framatome evaluate at least three cycles of data to confirm that uncertainty verification criteria in the topical report are met. Therefore, additional benchmarking analyses based on previous and current Callaway Unit 1 cores were performed. The benchmarking demonstrates acceptable results, which provide evidence that ARCADIA can successfully model the Callaway core and that this ability can also be extended to include the use of GAIA fuel due to its similarities to the other fuel designs used in the TR. This meets the requirements of L&C No. 2 for Callaway Unit 1.

L&C Nos. 4, 5, and 6 pertain to limitations and conditions for changes in the COBRA-FLX module, use of ARTEMIS as a stand-alone evaluation model for non-LOCA SRP Chapter 15 events, and changes made to the ARCADIA code system, respectively. The ARCADIA modules and codes used in the Callaway analyses comply with these L&Cs.

Key safety parameters are calculated and verified as part of the neutronics core design analysis (see Table 3-1) using the ARCADIA code system. These parameters are then conservatively treated in the safety analysis. Validation of key safety parameters is performed by comparing calculated parameters for the cycle-specific reload to the values used in the safety analysis. These cycle-specific parameters will be generated based on Framatome methodology using Framatome codes. If the key parameters are not within the reference safety analysis for a given transient, then the transient will be reanalyzed or reevaluated on a cycle-to-cycle basis using the stated methods.

Table 3-1: Range of Key Safety Parameters

Technical Specification (Reference 42)	Safety Parameter	Transition Analysis Value
Technical Specifications (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) (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) (See Note)	-1.871 to -1.485
Not a TS	Beta-Effective (See Note)	0.0052 to 0.0063
Not a TS	Power Coefficient (See Note)	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)) F_Q^{RTP}	2.5
TS 3.2.2 COLR Section 2.6	Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^{N}$) $F_{\Delta H}^{RTP}$	1.65
TS 3.2.3 COLR Section 2.7	Axial Flux Difference (AFD) at (100% Power) (%ΔI)	See Figure 4-9

NOTE: BETA-effective, DTC, and power coefficients do not have analyses or TS limits directly associated with them. These parameters are major contributors to transient analysis behavior and are good early indicators of significant physics characteristics changes in the core. Current design values for the Beta-effective and DTC parameters are expected ranges only. The power coefficient is inherently negative.

3.4. *Description of Design Evaluations*

Standard nuclear design analytical models and methods (References 17 and 18) accurately describe the neutronic behavior of the Framatome GAIA fuel. The specific design bases and their relation to the General Design Criteria (GDC) in 10 CFR 50, Appendix A for the Framatome GAIA design are discussed in Reference 1.

Two transition core designs followed by the reference cycle (also referred to as representative cycle) with full core of GAIA fuel have been developed for Callaway Unit 1 to model the transition to Framatome GAIA fuel. The loading patterns were developed based on design requirements (e.g., energy, peaking, and pin burnup limits (see References 1 and 7)) specified for Callaway Unit 1. The loading patterns were depleted at a core power of 3565 MWt. These cycles were developed to be bounding of future cycle designs to produce conservative margins to acceptance criteria. 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. Table 3-2 contains key information based on the nominal transition cycle designs and nominal control rod position of 216 steps withdrawn. Key safety parameters for the analyzed core designs were verified against the parameter range specified in Table 3-1.

The combination of fresh fuel enrichment loading and integrated burnable absorber enrichments and loadings are applied to control the peaking factors and maintain compliance with the Technical Specifications requirements. Changes in boron concentration and axial offset are typical of normal cycle-to-cycle variations in the core design.

Table 3-2: Projected Transition Cycle Core Characteristics

Transition Cycle	Cycle Energy (EFPD)	Number of Feed Framatome Assemblies	Maximum HFP $F_{\Delta H}$	Maximum HFP F_Q
N-2	501	84	1.480	1.808
N-1	501	80	1.477	1.769
N	501	77	1.469	1.790

3.5. *Results*

Margin to key safety parameter limits (Table 3-1) is maintained during the transition from co-resident fuel to Framatome GAIA fuel.

The changes in fuel design provide results that are similar to those observed in previous cycles. Power peaking and reactivity parameters remain within the limits specified in the Technical Specifications. Changes to the core power distributions, power peaking factors and reactivity parameters observed for the transition cycles and reference cycle are consistent with and bounding of cycle-to-cycle variations in core loading patterns. These parameters vary from cycle-to-cycle based on energy requirements, that are controlled through the use of feed batch size and enrichment along with the use of fuel rods containing Gadolinia (Gd_2O_3) neutron burnable absorbers.

3.6. *Conclusion*

The nuclear core design analysis for the transition from co-resident fuel to Framatome GAIA fuel confirms that peaking factors and key safety parameters are maintained within their specified limits using Framatome methodologies and codes. Cycle checks are performed against the reference core design key safety parameters including fuel rod peaking factor limits using Framatome's advanced codes (References 17 and 18).

4. POWER DISTRIBUTION CONTROL

4.1. *Description*

Reactors that operate using periodic flux maps to determine the core power distribution, as opposed to reactors that use a continuous measurement system, require a Power Distribution Control (PDC) procedure for reactor power peaking control between flux maps. This ensures that the Technical Specifications on power total peaking limits, mainly dictated by the LOCA, and the Loss-of-Coolant-Flow (LOCF) transient, are not exceeded during the interval between flux maps.

4.2. *Methodology*

A PDC methodology, previously approved by the NRC, is presented in References 19, 20, and 21 (PDC-3). The methodology in these references is based on a conservative approach focused on using a defined set of demanding load follow scenarios. The PDC methodology presented here (termed PDC-A) is a modification of the previously approved PDC-3 methodology.

The basic concept of the PDC-A methodology is to control the variation in core power distribution during reactor operation by controlling the variation in core Axial Flux Difference (AFD), or equivalently core power Axial Offset (AO). AFD and AO are defined as:

$$AFD\ (\%) = (P_T - P_B) / P_{RTP} * 100.0$$

$$AO\ (\%) = (P_T - P_B) / P_{CORE} * 100.0 = AFD\ (\%) * P_{RTP} / P_{CORE}$$

where,

P_T - is power in the top half of the core

P_B - is power in the bottom half of the core

P_{RTP} - is Rated Thermal Power (RTP)

P_{CORE} - is core thermal power

Typical Technical Specifications for plants requiring a PDC methodology are defined in terms of AFD. Conversely, typical core simulator inputs are provided in terms of AO. For this reason, AFD is used in the discussion of the PDC-A procedure while AO is used for analysis purposes.

Under the PDC-A methodology, a target AFD is determined at full power equilibrium conditions. The target AFD band is defined as the allowed AFD variation about the target AFD. The target AFD [

].

The major feature of the PDC-A methodology is the V(z) distribution. The V(z) distribution is the ratio of the maximum associated increase in $F_Q(z)$, total peaking by plane, during non-equilibrium operation when following the PDC-A methodology, to the $F_Q(z)$ corresponding to the target AFD. A comparison of ($F_Q(z)$ at the target AFD) * V(z) to the Technical Specification limit on $F_Q(z)$ ensures that the Technical Specification limit is not exceeded during the time between measurements of $F_Q(z)$ with the flux map system.

The PDC-A methodology is based on the methodology defined in Reference 21 for calculation of the V(z) function. The differences between PDC-A and the previously approved PDC-3 methodology (Reference 21) are:

[ ]

4.2.1. Regulatory Requirements

The regulatory requirements that PDC-A addresses are: 10 CFR 50.36(c)(2)(ii), 10 CFR 50.36(c)(2)(ii)B Criterion 2, and 10 CFR 50.36(c)(3) Surveillance requirements (Reference 22).

10 CFR 50.36(c)(2)(ii) states: "A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria."

10 CFR 50.36(c)(2)(ii)B Criterion 2 states: "a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier."

10 CFR 50.36(c)(3) Surveillance requirements states: "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

4.2.2. The PDC-A Methodology

The PDC-A methodology provides a process to generate the V(z) function and [

]. In PDC-A, the allowable AFD range for operation is expressed as an "AFD barn" which is defined as the AFD limits (AFD band) applied as a deviation about the target AFD. Table 4-1 and Figure 4-1 present a sample set of target AFDs, a target AFD band, and an associated AFD barn in tabular and graphic forms, respectively. The target AFD band is specified in the COLR or Technical Specifications and includes the AFD uncertainty. The target AFD is the AFD at the burnup under consideration and is proportional to the power level. [

]. During operation, the target AFD is set mainly by the measured values obtained from detailed flux/power measurements.

Table 4-1: AFD Band, Target AFD, and Target AFD Band (Example)

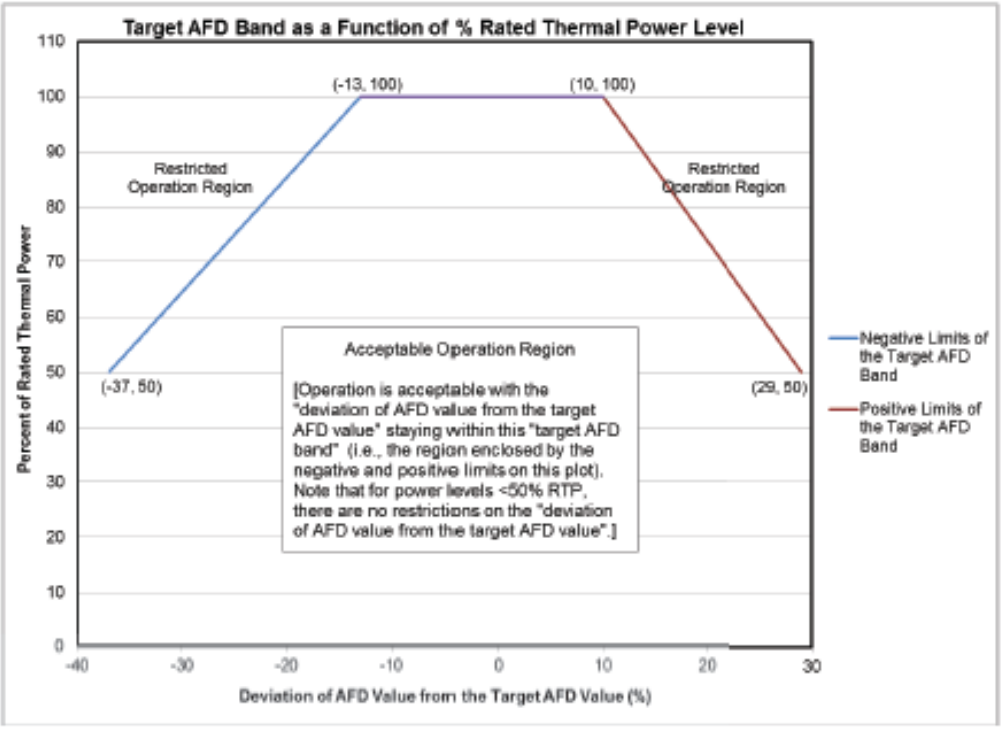
AFD Band (Limits)		
Power Level (%RTP) [a]	**Max Allowable AFD Deviation (%) from the Target AFD Value in the Negative Direction**	**Max Allowable AFD Deviation (%) from the Target AFD Value in the Positive Direction**
100	-13	10
50	-37	29
<50	No Limit	No Limit

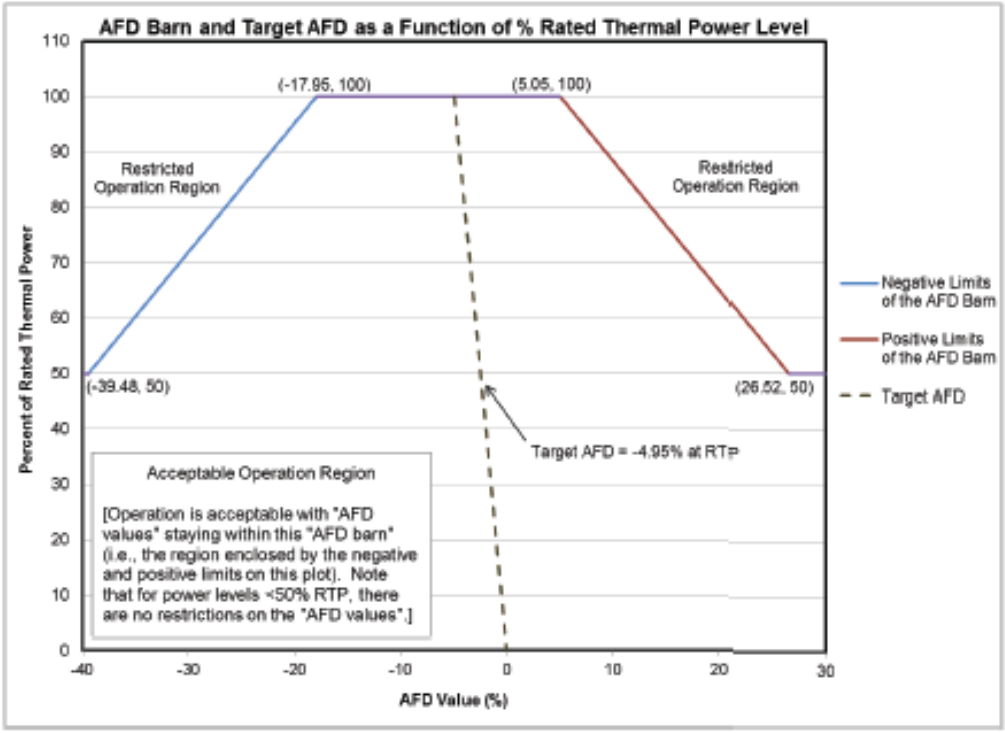
Target AFD	
Power Level (%RTP)	**Target AFD (%) [b]**
100	-4.955
50	-2.477
<50	-4.955 * Power / 100.0

Target AFD Band		
Power Level (%RTP)	**Max Allowable AFD Value (%) in the Negative Direction**	**Max Allowable AFD Value (%) in the Positive Direction**
100	-17.955	5.045
50	-39.477	26.523
<50	No Limit	No Limit

[a] For power levels ≥ 50%RTP, the power level dependencies of the maximum allowable AFD deviations from the target AFD values in both negative and positive directions are linear.
[b] Target AFD = AO at 100% RTP, Target AFD (XX% RTP) = Target AFD (100% RTP) * (XX / 100). AO at 100% RTP is a calculated value for the cycle burnup from simulated design calculations.

Figure 4-1: Graphic Presentation of AFD Band, Target AFD, and Target AFD Band (Example)





4.2.3. Determination of the V(z) Function

All core simulations at the reload design stage are performed with the NRC-approved ARCADIA reactor analysis system (References 17 and 18). All example calculations are based on data from a reload cycle of a Westinghouse plant.

During the reload design process, a burnup dependent V(z) function is conservatively generated to cover the maximum peaking variation out of all power distributions that can occur during the interval between flux maps. This V(z) function is generated such that it is applied to the $F_Q(z)$ at full power equilibrium conditions to check if the $F_Q(z)$ peaking limits are challenged during operation at non-equilibrium conditions prior to the next measurement of $F_Q(z)$. Power distribution calculations assume that all associated continuously measured parameters (such as AFD or control bank positions) are within their corresponding allowable operating ranges and the follow-on burnup period should be consistent with the frequency of flux map surveillance required by Technical Specifications. More explicitly, the V(z) function is generated such that its value is greater than the value of the following mathematical expression:

[ ]

The [ ]. The V(z) function is selected as the maximum peaking variation at each axial level. The V(z) function also depends [ ]. Since no AFD restrictions apply to operation at power levels below 50% RTP, the V(z) function is [ ] for power levels less than 50% RTP.

The process developed in Reference 21 provides a conservative approach in generating the V(z) function. Briefly, the process simulates [ ] load follow scenarios, all starting from the full power equilibrium condition. These load follow scenarios are intentionally designed to be very demanding and [ ]. Together, the [ ] create peaking variations high enough to cover those from power distributions that are observed during normal operation where AFD is maintained within the limits of the allowable operating region. More explicitly, the V(z) function is generated from:

[ ]

The [ ] transient scenarios defined in Reference 21 are used in defining the PDC-A process for the generation of the V(z) function.

Using the V(z) function, compliance with the peaking limits for operation during the cycle of interest is possible. Detailed descriptions for the required verification are given in Section 4.2.4. If the verification indicates that the limits on $F_Q(z)$ may be exceeded, then a method is defined to reduce the potential $F_Q(z)$ during non-equilibrium operation [ ].

As previously mentioned, generation of the V(z) function is performed through the simulation of [ ] load follow scenarios. These are the same load follow scenarios defined in Reference 21. The scenarios are applied through a [ ] as shown in Figure 4-2. These simulations are performed to generate the limiting power distributions that represent the most severe challenges to the power distribution limits. The [ ] load follow scenarios are described below. Discussion of the load follow scenarios are provided in terms of AO, which is consistent with code input requirements:

[ ]

[ ]

In all [ ] load follow scenarios, AO is pushed to extreme values while maintaining the control bank within insertion limits. Additionally, [ ]. The above simulations produce extreme conditions which may exceed the AFD barn limits [ ]. In these cases, the transients are [ ].

To account for the uncertainties in the control rod model, [ ], as defined in References 17 and 18.

The above load follow scenarios are simulated at [ ] on core power distribution are adequately covered. The PDC-A methodology defines AFD limits as a deviation about a target. Therefore, as the reactor [ ], the limits also change. This is captured in the above transient simulations by [ ] evaluated.

Generation of V(z) functions is illustrated by a sample problem. The sample problem considers one cycle of a 3-loop Westinghouse PWR. The analysis was performed [ ].

Figure 4-2: Power Level versus Clock Time [ ] for Power Cycling Scenarios

[ ]

Figure 4-3 provides an example of the AFD values (which are directly related to AO values) produced during the transients [

]. This figure provides the limits in terms of AFD to show the coverage of the barn [ ]. As seen in Figure 4-3, [

]. In addition, the V(z) penalty function is largely dependent on the data at 100% RTP. Figure 4-4 shows a comparison between a V(z) penalty function generated with data [

]. The penalty functions in this figure represent [ ]. As can be seen in Figure 4-4, [

].

The [ ] are selected to account for changes in AFD [ ].

If AFD is [

]. Based on experience from use of References 19, 20, and 21, at least one set of simulation calculations is required [

].

Figure 4-3: Coverage of the Barn in Terms of Axial Flux Difference

Figure 4-4: V(z) Function versus Axial Height at EOC for Power Level Comparison

The resulting [ ] database for the V(z) function is converted to a conservative [ ] V(z) penalty function. This is done by assigning the maximum V(z) points [ ].

All V(z) functions have a minimum defined value to ensure conservatism. A typical minimum value of **[ ]** is conservative based on data obtained from previous reload design calculations and experiences accumulated from core follow calculations and core monitoring observations. Figure 4-5 shows V(z) penalty functions for the sample problem **[ ]**.

Figure 4-5: V(z) Function versus Axial Height

[ ]

4.2.4. Decreasing F_Q by [ ] during Surveillance

When the maximum $F_Q(z) * V(z)$ value (with all applicable uncertainties included) during surveillance challenges the Technical Specification local peaking limit, it is possible to reduce the value of F_Q [ ]. The behavior of F_Q with respect to AFD can be defined [ ].

F_Q as a function of AFD is shown in Figure 4-6 for the 100% power data [ ], using the data approaching the AFD limits. The [ ] are used to [ ] for each percent change in F_Q with respect to the F_Q value at the [ ] AFD limit, that exceed the Technical Specification limit. Mathematically, the [ ] can be expressed as:

[ ]

The [ ] determined from the [ ] shown in Figure 4-6 are as follows:

[ ]

Based on these results, the AFD limit is [ ] that the F_Q exceeds the Technical Specification limit. This [ ] is to be verified each cycle and may be adjusted based on the cycle-specific analysis.

Figure 4-6: F_Q versus AFD

4.2.5. Power Distribution Limit Verification

4.2.5.1. Verification of Peaking Limits with Flux Maps

The measured power distribution and associated AFD and $F_Q(z)$ values are obtained by performing a flux map during operation at equilibrium conditions and are used to check $F_Q(z)$ against the limit. The measured power AFD is used to re-initialize the AFD barn (i.e., allowable AFD operating region) for operation until the next flux map at equilibrium conditions is performed. The acceptance criteria are dictated by the power distribution related Limiting Conditions of Operation (LCOs) in the Technical Specifications.

4.2.5.2. Target AFD Considerations

The initial AFD after refueling used to establish the target AFD at RTP may be determined using design predictions. Afterwards, the target AFD at RTP is updated by $\frac{\text{Measured AFD}}{\text{Measured Power in Fraction of RTP}}$ at the time when the flux map is taken at equilibrium conditions. Target AFDs for other power levels are obtained by multiplying the RTP value by the core thermal power level fractions. Flux maps should be taken at high power levels.

The measured target AFD obtained from detailed flux/power measurements under equilibrium conditions reflects the operating history. Its difference relative to the calculated target AFD, as well as the trend of such differences, must be monitored to detect problems that may arise with long-term operations with one or more process variables outside the range recommended by fuel vendor operational guidelines. These operational guidelines are recommendations that provide the plant operations staff with the basis for expected or intended power operation and specify guidance in the event of deviations from expected operation.

4.2.5.3. Performing Checks against Peaking Limits

The peaking values to be compared against their corresponding peaking limits include:

$$F_Q^C(z) = F_Q^M(z) \ * \text{[Measurement Uncertainty]} * \text{[Manufacturing Tolerance]}$$

$F_Q^V(z) = F_Q^C(z) * V(z)$

where

$F_Q^M(z)$ - the measured $F_Q(z)$

$V(z)$ - one of the penalty functions specified in the COLR and selected to be used according to the following rules:

1. Select the V(z) penalty functions specified for the burnup range of the flux map.
2. Apply the V(z) to the $F_Q^C(z)$ value that is based on the measured F_Q.
3. If the $F_{\Delta H}$ (=Measured $F_{\Delta H}$ + Measurement Uncertainty) for the current time point is greater than the $F_{\Delta H}$ for the previous time point, then increase $F_Q^V(z)$ by [ ].
4. If the $F_Q^V(z)$ value exceeds the $F_Q(z)$ limit, [ ].

Values of the measurement uncertainties are specified in the plant's COLR or Technical Specifications. As stated in Section 4.2.3, AFD restrictions are not applicable for operation at power levels below 50% RTP; therefore, the V(z) penalty function is set to [ ] for power levels less than 50% RTP.

4.2.5.4. AFD Barn Initialization for Allowable AFD Operating Region

Using the target AFD and the target AFD band associated with the V(z), the positive and negative AFD limits are calculated at appropriate power levels to form the AFD barn (i.e., the allowable AFD operating region) by:

[ ]

For power levels < 50% RTP where no AFD limits are defined, there are no restrictions on the AFD value and the V(z) is [ ].

This AFD barn accounting for the target AFD is used until the next flux map is taken for $F_Q(z)$ surveillance. An example AFD barn is presented graphically in Figure 4-1.

4.2.6. Technical Specifications

4.2.6.1. PDC-A Based Technical Specification Changes

The methodology described in Sections 4.2 and 4.2.2 are used for the Technical Specification 3.2.1, 3.2.2, and 3.2.3 revisions. The basis for the revisions to Technical Specification 3.2.1 and 3.2.2 are described in Section 4.2.5.1, 4.2.5.2, and 4.2.5.3. Section 4.2.3 describes the methodology used to calculate the V(z) values for each cycle. The basis for the revision to Technical Specification 3.2.3 is described in Section 4.2.5.2 and 4.2.5.4. The surveillance requirements for the revision to Technical Specification 3.2.1 and 3.2.2 are based on the information in Section 4.2.5.3. The surveillance requirements for the revision to the Technical Specification 3.2.3 are based on the information in Section 4.2.5.4.

As described in Section 4.2, PDC-A is mostly focused on the V(z) and F_Q; i.e., mainly focused on Technical Specification 3.2.1 as indicated in Section 4.2.2 and Section 4.2.3. The $F_{\Delta H}$ revision is straight-forward and is affected by PDC-A only by the addition of a conservative factor that is imposed on F_Q if $F_{\Delta H}$ increases between two consecutive measurements (given in the example Technical Specification Surveillance Requirement 3.2.2.1 and described in Section 4.2.5.3). This Surveillance Requirement is similar to Surveillance Requirement 3.2.1.2 in the existing Technical Specifications.

4.2.6.2. Monitoring Framatome Fuel with Westinghouse Methods

Monitoring of power peaking to Technical Specifications 3.2.1 and 3.2.2 requires the ability to predict $F_{\Delta H}$ and F_Q consistently and there are no fuel design changes between the co-resident fuel and GAIA that impact results on determining power distributions and power peaking. Therefore, methodologies for the determination of power distribution control are not impacted by insertion of GAIA fuel assemblies into the Callaway Unit 1 reactor if these assemblies are modeled explicitly in the core design.

The predicted peaking values are expected to remain within the measurement uncertainty to confirm the accuracy of the explicit model. Based on the ARCADIA benchmarking of the Callaway plant, prediction of power peaking is consistent between Westinghouse and Framatome. The existing Technical Specifications 3.2.1 and 3.2.2 remain applicable to GAIA fuel with use of the co-resident fuel vendor's power distribution control methodology.

Replacement Technical Specifications 3.2.1 and 3.2.2 are presented for application to Callaway Unit 1 following full batch transition to GAIA fuel.

4.2.7. Summary

The PDC-A methodology provides a means for ensuring that Technical Specification power peaking limits are not exceeded during the intervals between periodic flux map measurements. Using the AFD limits contained in the TS/COLR, a series of predefined load follow scenarios are simulated with ARTEMIS. A **[** **]** V(z) function is conservatively generated to cover the maximum peaking variation out of all power distributions that can occur during intervals between flux maps. Flux maps obtained during operation at equilibrium conditions determine the measured power distribution and associated AFD and $F_Q(z)$ values. The measured power AFD is used to re-initialize the allowable AFD operating region for use until the next flux map at equilibrium conditions is performed. The V(z) function is applied to the $F_Q(z)$ measured at full-power equilibrium conditions to check if the $F_Q(z)$ peaking limits are challenged during operation at non-equilibrium conditions prior to the next measurement of $F_Q(z)$ using the flux map system. If $F_Q(z)$ peaking limits are challenged, a method is provided to **[** **]**, which reduces the F_Q such that Technical Specification limits are met.

4.3. *Description of Design Evaluations*

The ARCADIA implementation of the Power Distribution Control methodology (PDC-A) is performed using a [ ] power cycle simulation. This cycle is a [ ] power. This maneuver is performed [ ]. In addition, a demonstration for reducing F_Q by narrowing the AFD band is provided for the situation in which the F_Q value exceeds the Technical Specification limit during operation.

At each of the six burnups considered, the calculated AO values obtained from 18 test cases were compared against their associated AO barn. These comparisons are provided in Figure 4-7. As can be seen from the plots in this figure, the core cannot reach the negative limits at lower powers. The V(z) curves generated using the data are conservative for the overall operations range when applied to F_Q for operation between flux maps. Since the reactor cannot physically operate at the conditions defined by the low power negative limits, using a conservative operational scheme means the generated V(z) values are conservative multipliers for expected operation conditions.

Figure 4-7: Calculated AFD for Load Follow Scenarios (Power (%) vs AFD (%))

4.3.1. F_Q Reduction

If the predicted F_Q exceeds the Technical Specification limit, it is desired to [ ]. The [ ] was determined based on following steps:

[ ]

The F_Q vs. AFD with appropriate bounding lines, based on only the 100% RTP conditions for all burnups, are provided in Figure 4-8.

[ ]

The calculation for the [ ] is shown in Table 4-2 and for each [ ].

Table 4-2: Calculation of Limiting Line and %AFD/%F_Q (see Figure 4-8)

[ ]

The [ ]

As can be seen, the change in FQ is converted to percent using the FQ value corresponding to the reduced AFD limit.

Figure 4-8: F_Q vs. AFD

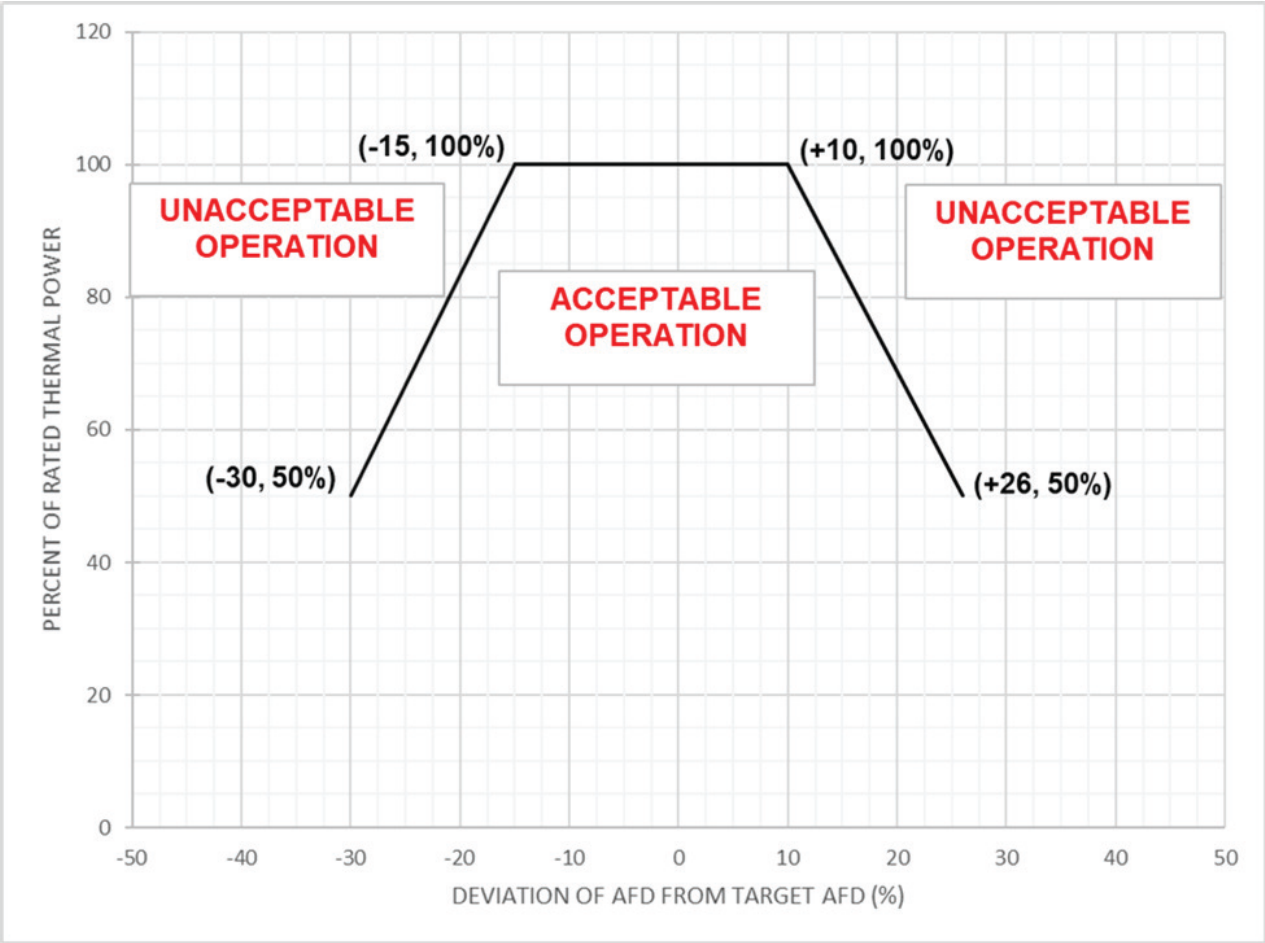
4.3.2. Results

This analysis provides a barn (Figure 4-9) that can be incorporated into the Callaway Unit 1 COLR. Additionally, a set of V(z) values as a function of axial height were also generated (Table 4-3) that are typical COLR values.

Table 4-3: Calculation VQP V(z) vs Core Height

The PDC-A based AFD barn is shown in Figure 4-9.

Figure 4-9: Callaway Unit 1 AFD Limits as a Function of Rated Thermal Power



4.4. *Conclusion*

This report shows acceptability of the power distribution control methodology for the application of the Framatome GAIA fuel design at Callaway Unit 1. Additionally, PDC-A provides a methodology to meet the regulatory requirements, 10 CFR 50.36(c)(2)(ii), 10 CFR 50.36(c)(2)(ii)B Criterion 2, and 10 CFR 50.36(c)(3) Surveillance requirements (Reference 22).

5. THERMAL AND HYDRAULIC DESIGN

5.1. *Description*

This section describes the Thermal-Hydraulic (T-H) analyses supporting the transition to Framatome GAIA fuel at Callaway Unit 1.

5.2. *Input Parameters and Assumptions*

The input parameters used to perform the T-H analyses include fuel design information derived from design documents, fuel assembly and component characteristics established by mechanical / hydraulic testing, and plant parameters provided by Ameren. Additional input parameters are taken from the COLR, Technical Specifications, and FSAR SP. Neutronic input parameters are generated with a representative core design to demonstrate that the fuel design and analysis methodologies are acceptable at Callaway Unit 1.

5.3. *Acceptance Criteria*

The reactor core is designed to meet the following T-H acceptance criteria. The classification of events is described in Reference 35, Table 3-10:

- There is at least a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience Departure from Nucleate Boiling (DNB) during Condition I or II events.
- There is at least a 95% probability at a 95% confidence level that the hot fuel rod in the core does not melt during Condition I or II events.

The CHF resulting in Departure from Nucleate Boiling (DNB) at a particular core location, as predicted by a CHF correlation, to the actual heat flux at the same core location is the DNB Ratio (DNBR). 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 peak linear heat generation rate (PLHGR) is compared to the linear heat generation rate (LHGR) corresponding to fuel centerline melt (FCM) to assess margin to fuel melting. Analytical 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.

5.4. *Methodology*

The thermal-hydraulic analysis of the GAIA fuel assembly is based on the methodologies described in Section 3.9 of Reference 35. The USNRC-approved computer codes used are described in Section 3.10 of Reference 35. The COBRA-FLX code (Reference 23) is also used to perform thermal-hydraulic compatibility analysis and guide tube boiling analysis. The solver options and empirical models approved in the limitations and conditions of Reference 23 are selected for COBRA-FLX analyses. Additionally, the fuel rod model and post-CHF heat transfer models within COBRA-FLX are not used. The remainder of this section describes the additional methods used for the Callaway Unit 1 VQP.

The approved methodology for verifying the applicability of the Over-Temperature Delta-Temperature (OTΔT) and Over-Power Delta-Temperature (OPΔT) trips is described in Reference 10. Additionally, Reference 10 describes the approved methodology for verifying the Core Safety Limit Lines (CSLLs). The setpoint verifications are confirmed each cycle. The setpoint verification analyses are performed in accordance with the L&Cs specified in Reference 10. As specified in the SER conditions, the statistical treatment of specific variables analyzed are consistent with Reference 10. Additionally, the steam generator safety valve limit line is set based on the steam generator tube plugging assumed.

The impact of rod bowing on the MDNBR and PLHGR is evaluated using the rod bow methodology described in Reference 25. This methodology is an update to the rod bow methodology in Reference 24 using additional fuel rod water channel gap measurement data and was submitted as a response to a request for additional information as part of a revision to Reference 6.

5.5. *Results*

5.5.1. Thermal-Hydraulic Compatibility

T-H compatibility analysis evaluates the relative T-H performance of hydraulically dissimilar fuel types in the core. This section demonstrates that the GAIA fuel assembly and the co-resident fuel assembly are thermal-hydraulically compatible.

5.5.1.1. Core Pressure Drop

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. An analysis was performed with the COBRA-FLX code to assess the change in core pressure drop associated with the fuel transition.

The core pressure drop for a full core of Framatome GAIA fuel assemblies is **[ ]** psi.

The total pressure drop associated with the full core of Framatome GAIA is **[ ]** psi lower than the total pressure drop of the co-resident core. The pressure drop profile between the two assembly types is shown in Figure 5-1.

Figure 5-1: Pressure Drop Profiles

[ ]

5.5.1.2. Total Bypass Flow

The change in total bypass flow was examined to determine if the active heat transfer coolant flow will be adversely impacted by the fuel transition. The bypass flow includes the following flow paths: guide tubes, vessel upper head, inlet-to-exit nozzle, and core barrel/baffle. The change in total bypass flow was determined by examining the change due to the non-guide tube paths and guide tube paths. Bypass flow for the non-guide tube paths is affected by changes in core pressure drop, while the guide tube bypass flow is dependent on both core pressure drop and assembly geometry.

The core pressure drop for a full core of Framatome GAIA fuel assemblies is lower than the core pressure drop 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 Framatome GAIA fuel assemblies. [

].

5.5.1.3. Crossflow Velocity

The inter-assembly crossflow velocities affecting the Framatome GAIA fuel assemblies were analyzed with the COBRA-FLX code to ensure satisfactory performance during transition. A bounding core configuration was considered for this analysis to cover mixed-core configurations associated with the transition. The results are representative of anticipated operating conditions and are used to develop bounding inputs for mechanical analyses.

5.5.1.4. Reactor Coolant System Flow Rate

An evaluation was performed to assess the change in primary system loop flow attributed to the fuel transition. The evaluation indicates that the transition from a full core of co-resident fuel to a full core of Framatome GAIA results in an increase in the RCS loop flow due to the lower pressure drop in the GAIA fuel assembly. [

].

5.5.1.5. Transition Core Departure from Nucleate Boiling 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. A matrix of boundary conditions, power levels, and power distributions were considered.

[

].

5.5.1.6. Control Rod Drop Times

An assessment was performed to validate that the Technical Specification requirement for the control rod drop time is not challenged as a result of 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.

Due to the similarities between the co-resident and Framatome guide tube designs, the technical specification 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.

5.5.2. Thermo-Hydrodynamic Instability

Framatome has evaluated the Callaway Unit 1 reactor for its susceptibility to a wide range of potential thermo-hydrodynamic instabilities. The evaluation demonstrates that Callaway Unit 1 with GAIA fuel assemblies will not experience thermo-hydrodynamic instabilities during normal operation and AOO's.

5.5.3. Rod Bow

The impact of rod bowing on the MDNBR and PLHGR was evaluated using the updated rod bow methodology described in Reference 25. The rod bow assessment has determined DNBR and LHGR penalties for rod bow and the exposure thresholds beyond which the DNBR and PLHGR penalties are applied. The calculated rod bow penalties shown in Table 5-1 are applied to DNBR and PLHGR for fuel assemblies exceeding the exposure thresholds in the appropriate analyses. [ ]. The DNBR rod bow penalty is applied to the calculated value of MDNBR using the following equation:

[ ]

Table 5-1: GAIA Rod Bow Penalties

[ ]

5.5.4. Guide Tube Heating

Boiling of coolant within the guide tubes has the potential to increase corrosion rates and be detrimental for neutron moderation. An 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. For conservatism, biased operating conditions were used in the analysis.

[2] The DNBR rod bow penalty is only applicable beyond the DNBR penalty threshold exposure

[

]. The analysis demonstrates that for all calculated control rod heating rates, boiling is precluded within the guide tube.

5.5.5. Event Specific Analysis

Margin to the acceptance criteria in Section 5.3 for event specific analysis is confirmed in Reference 35.

5.5.6. Setpoint Analysis

5.5.6.1. Over-Temperature Delta-Temperature (OTΔT) Verification

The thermal over-temperature trip is designed to protect the fuel SAFDL on DNB for normal operating conditions and for trip basis AOO's. The implementation of this trip is that if a core ΔT in excess of a calculated maximum value is indicated, automatic shutdown of the reactor via the RPS is initiated. The OTΔT works in conjunction with the OPΔT trip, the high flux trip, and the steam generator safety valves to ensure that the DNB SAFDL is protected, and hot leg saturation is precluded.

The VQP setpoints analyses confirm that neither DNB nor hot leg saturation are predicted to occur within the operating space permitted by the OTΔT and associated trips for the GAIA fuel assembly design.

5.5.6.2. Over-Power Delta-Temperature (OPΔT) Verification

This setpoint analysis ensures there is sufficient margin for the Over-power Delta-Temperature (OPΔT) trip. The thermal over-power trip is designed to protect the fuel SAFDL on centerline melt for normal operating conditions and for trip basis AOO's. The implementation of this trip is that if a core ΔT in excess of a calculated maximum value is indicated, automatic shutdown of the reactor via the reactor protection system is initiated. The maximum OPΔT trip ΔT is set such that the FCM LHGR is protected, based upon a correlation of core thermal power to core ΔT.

This VQP analysis confirms that the OPΔT trip protects the Framatome GAIA fuel assembly from FCM.

5.5.6.3. Core Safety Limits Verification

The CSLLs are a set of safety limits which establish frontiers in reactor operating space such that core exit saturation and DNB are both avoided at a 95/95 probability/confidence level. In the Callaway Unit 1 Technical Specifications, a set of limiting saturation and DNB lines are required to be documented as part of the core safety limits. For the VQP, a verification analysis is required to demonstrate that positive power margin exists between the documented CSLLs and the saturation/DNB frontiers corresponding to the core configuration featuring Framatome GAIA fuel.

The VQP setpoints analyses confirm that neither DNB nor hot leg saturation are predicted to occur within the operating space permitted by the existing CSLLs for the GAIA fuel assembly design.

5.6. *Conclusion*

This section describes the methodology used to analyze the T-H design of the GAIA fuel assembly at Callaway Unit 1. The GAIA fuel assembly is thermal-hydraulically compatible with the co-resident fuel assembly. Additionally, the methods described are used to demonstrate the GAIA fuel assembly is protected from DNB and FCM.

6. ACCIDENT AND TRANSIENT ANALYSES

6.1. *Non-LOCA Analyses*

The Chapter 15 non-LOCA analyses supporting the introduction of the Framatome GAIA fuel assembly design into Callaway Unit 1 are summarized within a separate Licensing Summary Report in Reference 35. The analyses are performed in accordance with the USNRC-approved method described in Reference 26 with any changes to the approved method and additional supporting methods identified in the Licensing Summary Report in Reference 35. The results of the Chapter 15 non-LOCA analyses demonstrate that acceptance criteria are met for the Framatome GAIA fuel assembly design. The analysis for spectrum of rod cluster control assembly ejection accidents is addressed separately from the remaining FSAR SP Chapter 15 events and is discussed in Section 6.1.1.

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

6.1.1.1. Accident Description

This event is initiated by a postulated rupture of a control rod drive mechanism housing. Such a rupture allows the full system pressure to act on the drive shaft, which ejects its control rod from the core. The consequences of the mechanical failure are a 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 reactor protection system with a reactor trip in response to changes in neutron flux or system pressure. Although the initial increase in power occurs too rapidly for control rod scram to affect the power increase, the negative reactivity inserted during scram does affect the fuel temperature and fuel rod cladding surface heat flux.

6.1.1.2. Acceptance Criteria

The acceptance criteria were defined in Reference 28. The relevant criteria are:

- **Fuel Rod Cladding Failure Thresholds** –
  - **High Temperature Cladding Failure Threshold** – For prompt critical scenarios, the failure threshold provided in Reference 28, Figure 1 is used for the first 3 seconds. For non-prompt critical scenarios and **[**

    ] are used as the threshold.
  - **Pellet Clad Mechanical Interaction (PCMI) Cladding Failure Threshold** – Pellet Clad Mechanical Interaction failure thresholds (Δcal/g) from Reference 28, Figures 2 and 3.
  - **Molten Fuel Cladding Failure Threshold** – Fuel cladding failure is presumed if predicted fuel temperature anywhere in the pellet exceeds incipient fuel melting conditions.
- **Allowable Limits on Radiological Consequences** – Reference 27 is used to determine the number of fuel rod failures but does not address the radiological consequences of the Rod Ejection Accident. The use of Reference 27 to determine the number of failed fuel rods has no effect on the existing downstream methodology used to determine the radiological consequences. RG 1.183 (Reference 29) and RG 1.195 (Reference 30) contain the accident dose radiological consequences criteria for control rod ejection accidents.
- **Allowable Limits on Reactor Coolant System Pressure** – The reactor coolant system pressure was not evaluated using Reference 27. No aspect of the Framatome fuel affects the severity of the rod ejection overpressure analysis and does not require reanalysis.

- **Allowable Limits on Damage Core Coolability** –
  - Peak radial average fuel enthalpy must remain below 230 cal/g (Reference 28 Section 6.0).
  - A limited amount of fuel melting is acceptable provided it is restricted to the fuel centerline region and is less than 10 percent of the pellet volume. The peak fuel temperature in the outer 90 percent of the pellet's volume must remain below incipient fuel melting conditions (Reference 28 Section 6.0).

6.1.1.3. Analysis Method

The analysis follows the USNRC-approved AREA methodology described in Reference 27 using the GALILEO fuel rod thermal mechanical methodology as described in Reference 31 with the criteria defined in Reference 28. In this application of the USNRC-approved AREA methodology, [

] to be consistent with the final approved version of GALILEO in Reference 31.

6.1.1.4. Conclusion

Key parameters listed for this event, such as ejected rod worth and Doppler reactivity feedback are potentially impacted by the transition to Framatome fuel. As such, the fuel related acceptance criteria for this event are evaluated to support the fuel transition. The event is analyzed using Framatome's USNRC-approved AREA methodology described in Reference 27 to the acceptance criteria in Reference 28. The use of Framatome's AREA methodology coupled evaluation model permits the use of conservative but [ ] the previous analysis of record. The results of the Rod Cluster Control Assembly Ejection event analysis will be available for USNRC audit.

6.2. *Loss-of-Coolant Accident Analyses*

The LOCA analyses to support the introduction of the Framatome GAIA fuel design for the Callaway Unit 1 plant are summarized within separate Licensing Summary Reports in Reference 36 (SBLOCA) and 37 (RLBLOCA). The licensing analyses are performed in accordance with the USNRC-approved methods of References 32, 33, and 34. Any changes to the approved methods and any additional supporting methods are identified within Reference 36 and 37. The detailed evaluation of the margins to 10 CFR 50.46(b) acceptance criteria is reported for SBLOCA and RLBLOCA in Reference 36 and 37, respectively.

7. METHODOLOGY APPLICABILITY

Table 7-1 provides a list of each of the NRC-approved topical reports that will be added to the Callaway Core Operating Limits Report. A cross-reference is provided for each topical report that points to the section within the LAR and supporting licensing documents where the applicability of the topical report is discussed.

Table 7-1: Core Operating Limits Report Methodologies

Methodology Title	Methodology Reference	Applicability Reference
EMF-2103(P)(A), Revision 3, “Realistic Large Break LOCA Methodology for Pressurized Water Reactors”	Reference 34	Section 3.3 of Reference 37
EMF-2328(P)(A), Revision 0, “PWR Small Break LOCA Evaluation Model, S-RELAP5 Based”	Reference 32	Section 3.3 of Reference 36
EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, “PWR Small Break LOCA Evaluation Model, S-RELAP5 Based”	Reference 33	Section 3.3 of Reference 36
EMF-2310(P)(A), Revision 1, “SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors”	Reference 26	Section 3.9 of Reference 35
XN-NF-82-21(P)(A), Revision 1, “Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations”	Reference 38	Section 3.9 of Reference 35
EMF-92-081(P)(A), Revision 1, “Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors”	Reference 10	Section 5.4 Section 3.9 of Reference 35
ANP-10341(P)(A), Revision 0, “The ORFEO-GAIA and ORFEO-NMGRID Critical Heat Flux Correlations”	Reference 39	Section 3.9 of Reference 35
XN-75-21(P)(A), Revision 2, “XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Core Operation”	Reference 40	Section 3.9 and 3.10 of Reference 35

Methodology Title	Methodology Reference	Applicability Reference
ANP-10311P-A, Revision 1, “COBRA-FLX: A Core Thermal-Hydraulic Analysis Code”	Reference 23	Section 5.4 Section 3.9 and 3.10 of Reference 35
XN-NF-82-06(P)(A), Revision 1, Supplement 2, 4, and 5, “Qualification of Exxon Nuclear Fuel for Extended Burnup”	Reference 41	Section 3.9 of Reference 35
XN-75-32(P)(A) Supplements 1, 2, 3, and 4, “Computational Procedure for Evaluating Fuel Rod Bowing”	Reference 24	Section 5.4
ANP-10297P-A, Revision 0, “The ARCADIA Reactor Analysis System for PWRs Methodology Description and Benchmarking Results”	Reference 17	Section 3.3 Section 3.9 and 3.10 of Reference 35
ANP-10297P-A, Revision 0, Supplement 1PA, Revision 1, “The ARCADIA Reactor Analysis System for PWRs Methodology Description and Benchmarking Results”	Reference 18	Section 3.3 Section 3.9 and 3.10 of Reference 35
ANP-10338P-A, Revision 0, “AREA – ARCADIA Rod Ejection Accident”	Reference 27	Section 6.1.1.3
ANP-10323P-A, Revision 1, “GALILEO Fuel Rod Thermal-Mechanical Methodology for Pressurized Water Reactors”	Reference 31	Section 6.1.1.3
BAW-10231P-A, Revision 1, “COPERNIC Fuel Rod Design Computer Code”	Reference 7	Section 2.4.4 Section 3.9 and 3.10 of Reference 35
BAW-10227P-A, Revision 1, “Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel”	Reference 6	Section 2.4.4
ANP-10342P-A, Revision 0, “GAIA Fuel Assembly Mechanical Design”	Reference 1	Section 2.4.3.1

8. SUMMARY AND CONCLUSION

This report shows acceptability for the application of the Framatome GAIA fuel design at Callaway Unit 1. The results displayed within the report show that Framatome GAIA fuel design complies with USNRC-approved topical reports regarding mechanical and structural analyses, nuclear design analyses, thermal-hydraulic analyses for steady-state and transient performance, and non-LOCA/LOCA safety analyses addressing transient and accident conditions. Alternate methods to the approved topical reports are applied with appropriate justifications and are clearly described within the document in the appropriate section.

Note that demonstration of the evaluation methodologies has been performed with a representative core design. The representative core design was developed to provide key safety parameters to support the transition from co-resident fuel to Framatome GAIA fuel prior to the development of cycle-specific designs. This provides assurance that the plant licensing bases will be met for the anticipated operation of the Framatome GAIA fuel during the transition and full core cycles.

In conclusion, this report supports the use of Framatome GAIA fuel at Callaway Unit 1.

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