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Dominion Energy®

September 16, 2024

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Serial No.	23-251D
NRA/SS:	R0
Docket No.	50-423
License No.	NPF-49

DOMINION ENERGY NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING PROPOSED AMENDMENT TO SUPPORT IMPLEMENTATION OF FRAMATOME GAIA FUEL

By letter dated October 30, 2023 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML23304A047), Dominion Energy Nuclear Connecticut, Inc. (DENC) submitted a License Amendment Request (LAR) to the Nuclear Regulatory Commission (NRC) to revise the Technical Specifications (TS) for Millstone Power Station Unit 3 (MPS3). The proposed LAR would revise the MPS3 TS to support the implementation of Framatome GAIA fuel, which is currently scheduled for onload during the spring 2025 refueling outage. The proposed TS changes include updating the Reactor Core Safety Limits (RCSLs, TS 2.1.1.1), reducing the Reactor Trip System (RTS) instrumentation trip setpoint for the Permissive-8 Interlock (TS Table 2.2-1, Item 18.c), and adding to the list of approved methodologies for the Core Operating Limits Report (COLR) (TS 6.9.1.6.b). Additionally, the LAR requests NRC approval of mixed-core Departure from Nucleate Boiling (DNB) penalties and MPS3-specific Design Basis Limits for a Fission Product Barrier (DBLFPBs) needed to support GAIA fuel implementation.

On July 31, 2024, the NRC transmitted the final version of the Request for Additional Information (RAI, ADAMS Accession No. ML24213A260) related to this LAR. DENC agreed to respond to the RAI within 45 days of issuance, or no later than September 16, 2024.

Proprietary and non-proprietary versions of DENC's response to the RAI are provided in Attachments 1 and 2, respectively. Attachment 3 provides supplemental information related to Table 1 in Attachment 1 of the subject LAR. Attachment 4 provides an application for withholding and affidavit from Framatome Inc.

Attachment 1 contains information proprietary to Framatome Inc. (Framatome) and is supported by an affidavit (Attachment 4) signed by Framatome, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations. Accordingly, it is respectfully requested the proprietary information be withheld from public disclosure in accordance with 10 CFR 2.390.

Attachment 1 contains information that is being withheld from public disclosure under 10 CFR 2.390. Upon separation from Attachment 1, this letter is decontrolled.

If you have any questions or require additional information, please contact Mr. Shayan Sinha at (804) 273-4687.

Sincerely,

James Holloway

James E. Holloway
Vice President – Nuclear Engineering and Fleet Support

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by James E. Holloway who is Vice President – Nuclear Engineering and Fleet Support of Dominion Energy Nuclear Connecticut, Inc. He has affirmed before me that he is duly authorized to execute and file the foregoing document on behalf of that company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 16th day of September, 2024.

My Commission Expires: August 31, 2027.

GARY DON MILLER
Notary Public
Commonwealth of Virginia
Reg. # 7629412
My Commission Expires August 31, 2027

Gary Don Miller
Notary Public

Attachments:

1. Response to Request for Additional Information Regarding Proposed Amendment to Support Implementation of Framatome GAIA Fuel (Proprietary)
2. Response to Request for Additional Information Regarding Proposed Amendment to Support Implementation of Framatome GAIA Fuel (Non-Proprietary)
3. Supplemental Information Related to LAR Attachment 1, Table 1
4. Framatome Application for Withholding and Affidavit

Commitments made in this letter: None

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Attachment 1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING PROPOSED AMENDMENT TO SUPPORT IMPLEMENTATION OF FRAMATOME GAIA FUEL (PROPRIETARY)

Dominion Energy Nuclear Connecticut, Inc.
Millstone Power Station Unit 3

Attachment 1 contains information that is being withheld from public disclosure under 10 CFR 2.390. Upon separation from Attachment 1, this page is decontrolled.

Attachment 2

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING PROPOSED AMENDMENT TO SUPPORT IMPLEMENTATION OF FRAMATOME GAIA FUEL (NON-PROPRIETARY)

Dominion Energy Nuclear Connecticut, Inc.
Millstone Power Station Unit 3

By letter dated October 30, 2023 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML23304A047), Dominion Energy Nuclear Connecticut, Inc. (DENC) submitted a License Amendment Request (LAR) to the Nuclear Regulatory Commission (NRC) to revise the Technical Specifications (TS) for Millstone Power Station Unit 3 (MPS3). The proposed LAR would revise the MPS3 TS to support the implementation of Framatome GAIA fuel, which is currently scheduled for onload during the spring 2025 refueling outage. The proposed TS changes include updating the Reactor Core Safety Limits (RCSLs, TS 2.1.1.1), reducing the Reactor Trip System (RTS) instrumentation trip setpoint for the Permissive-8 Interlock (TS Table 2.2-1, Item 18.C), and adding to the list of approved methodologies for the Core Operating Limits Report (COLR, TS 6.9.1.6.b). Additionally, the LAR requests NRC approval of mixed-core Departure from Nucleate Boiling (DNB) penalties and MPS3-specific Design Basis Limits for a Fission Product Barrier (DBLFPBs) needed to support GAIA fuel implementation.

On July 31, 2024, the NRC transmitted the final version of the Request for Additional Information (RAI, ADAMS Accession No. ML24213A260) related to the subject LAR. DENC agreed to respond to the RAI within 45 days of issuance, or no later than September 16, 2024.

This attachment provides the non-proprietary version of DENC's response to the RAI.

<u>DOM-NAF-2-P-A, Appendix F NRC-Approval</u>

As described in Attachment 1, Section 3.1 of the subject LAR, the requested GAIA implementation changes are dependent upon NRC-approval of Appendix F to DOM-NAF-2-P-A. DOM-NAF-2-P-A, Appendix F (ADAMS Accession No. ML24170B053) qualifies the ORFEO-GAIA and ORFEO-NMGRID Critical Heat Flux (CHF) correlations for thermal-hydraulic analysis of the GAIA fuel product using the VIPRE-D computer code. However, the DOM-NAF-2-P-A, Appendix F final NRC Safety Evaluation (SE) had not been released at the time of LAR submittal on October 30, 2023. Attachment 1, Section 3.1 of the subject MPS3 LAR stated that the LAR submittal and its supporting documentation met the Limitations and Conditions (L&Cs) imposed by DENC in the DOM-NAF-2-P-A, Appendix F submittal, and described DENC's contingency actions should additional L&Cs be imposed by the NRC in their approval of DOM-NAF-2-P-A, Appendix F.

The SE supporting NRC-approval of DOM-NAF-2-P-A, Appendix F for licensing applications was released on December 20, 2023 (ADAMS Accession No. ML23283A305). The L&Cs outlined in the final SE supporting DOM-NAF-2-P-A, Appendix F have been reviewed by DENC. The NRC final SE did not include any L&Cs beyond those proposed by DENC. As presently submitted, the October 30, 2023, LAR and its supporting documentation satisfy the L&Cs of NRC-approved DOM-NAF-2-P-A, Appendix F.

RAI 1 (AQ-A.1)

P-8 Trip Setpoint Analysis

Section 3.3 of Attachment 1 to the LAR dated October 30, 2023, discusses the P-8 trip setpoint analysis. It indicates that a deterministic departure from nucleate boiling ratio (DNBR) analysis for a low flow condition consistent with the P-8 interlock logic (loss of a reactor coolant pump (RCP)) was performed to establish the proposed P-8 trip setpoint. The analysis shows that an upper limit analytical value (AV) of 45 percent power is required to ensure the P-8 setpoint maintains the calculated DNBRs within the acceptable limits for GAIA fuel.

Discuss the methods and computer codes used in the determination of the upper limit AV for P-8 setpoint and address compliance with the conditions and limitations imposed in the NRC safety evaluation report (SER) for the computer codes and methods.

DENC Response to RAI-1

P-8 is a RTS interlock that blocks a single loop low Reactor Coolant System (RCS) flow reactor trip when three out of four power range neutron flux signals are less than the permissive setpoint. When operating below P-8, the loss of a single Reactor Coolant Pump (RCP) will not result in a direct reactor trip.

The DNB analysis that supports the changes to TS Table 2.2-1, Item 18.C is considered a verification of the proposed P-8 setpoint reduction. The setpoint evaluation does not meet the Nuclear Energy Institute (NEI) Document 96-07, Rev. 1 definition of a safety analysis as the event is not described within the MPS3 Final Safety Analysis Report (FSAR). The below described NRC-approved methods, codes, and design limits are used for simplicity in validating the P-8 setpoint change.

The subject LAR, dated October 30, 2023, stated a deterministic DNB evaluation was performed in accordance with the DOM-NAF-2-P-A methodology to establish the proposed P-8 allowable value. DOM-NAF-2-P-A provides the necessary documentation to describe the Dominion Energy fleet use of the VIPRE-D code, including modeling and qualification for Pressurized Water Reactors (PWR) core thermal-hydraulic design. DOM-NAF-2-P-A also demonstrates that the VIPRE-D methodology is appropriate for PWR licensing applications.

The subject LAR further states the MPS3 P-8 evaluation results satisfy applicable Departure from Nucleate Boiling Ratio (DNBR) limits for the ORFEO-GAIA CHF correlation and that positive margins are maintained for resident Westinghouse fuel. The ORFEO-GAIA and ORFEO-NMGRID CHF correlations have been qualified for thermal-hydraulic analysis of the GAIA fuel product using VIPRE-D in Appendix F to DOM-NAF-2-P-A. The recently approved DOM-NAF-2-P-A, Appendix F is proposed for addition to the MPS3 TS 6.9.1.6.b list of NRC-approved COLR methodologies in the subject LAR. MPS3 TS 6.9.1.6.b currently lists DOM-NAF-2-P-A, Appendices C and D as NRC-approved methodologies for the DNBR analysis of resident Westinghouse fuel (MPS3-specific NRC-approval in ADAMS Accession No. ML16131A728). All

MPS3 P-8 DNB calculations were confirmed to remain within the DOM-NAF-2-P-A L&Cs for applicable correlations found in Appendices C, D and F.

Additionally, Dominion Energy's NRC-approved VEP-FRD-41-P-A methodology is cited in the MPS3 P-8 setpoint evaluation. VEP-FRD-41-P-A (ADAMS Accession No. ML19141A148) provides the necessary documentation to describe the Dominion Energy fleet use of the RETRAN computer code to perform transient thermal-hydraulic analyses of the Nuclear Steam Supply System (NSSS). RETRAN calculates general system parameters as a function of time that may be used as boundary conditions for input into more detailed calculations of DNBR. The NRC approved the application of VEP-FRD-41-P-A to MPS3 in ADAMS Accession No. ML16131A728 where it was stated Dominion Energy's use of the RETRAN method at MPS3 is within the NRC-accepted L&Cs of VEP-FRD-41-P-A.

<u>RAI 2 (AQ A.2)</u>

Discuss the bases of the selected values for plant key parameters used in the deterministic DNBR analysis and justify that the selected value of each parameter would result in the lowest margin to the acceptance criteria. The key parameters of concern include the RCS pressure and temperature, RCS flow rate, and axial and radial peaking factors, and axial power shape. Also, discuss the sensitivity study that was used to determine the limiting axial power shape for use in the P-8 setpoint analysis, and show that the limiting power shape bounds the operating plant power shape data in MPS3.

<u>DENC Response to RAI-2</u>

A DNB analysis was performed to demonstrate the adequacy of the MPS3 P-8 setpoint change. As stated above, the MPS3 P-8 verification used NRC-approved methods, codes, and design limits for simplicity. All reactor core thermal-hydraulic calculations were performed using the Dominion Energy fleet VIPRE-D code in accordance with the DOM-NAF-2-P-A methodology. NRC-approved CHF correlations documented in DOM-NAF-2-P-A were applied.

The TS Table 2.2-1, Item 18.C changes were validated by a steady-state DNB run performed at the proposed P-8 setting (including a 10% power allowance). Other key inputs corresponding to the P-8 reactor condition were developed from calculations performed using the Dominion Energy fleet system transient code, RETRAN, per the VEP-FRD-41-P-A methodology. For example, a 3-out-of-4 pump flow was obtained from a RETRAN evaluation, as it corresponds to the final steady-state obtained following the loss of an RCP (i.e. lowest anticipated flow). A low RCS pressure and high RCS temperature are input to the analysis which are the limiting directions for DNB analysis. These inputs are based on plant protection setpoints and mitigative equipment settings, respectively. The radial peaking factor reflects anticipated COLR limits and is adjusted by the part power multiplier (TS 3.2.3.1.b) that increases peaking at lower powers.

A sensitivity study examined a wide range of axial skewing to identify an appropriate axial power shape and peak. This input was generated using the Dominion Energy NRC-approved codes and methods currently used to support MPS3 reload analyses (i.e., VEP-FRD-42-A and VEP-NE-1-A, NRC-approved for use at MPS3 via ADAMS Accession No. ML16131A728). A sufficiently limiting power shape was selected and confirmed to bound measured MPS3 flux map data at power levels close to the proposed P-8 setpoint.

Deterministic DNBR methods are used since the plant uncertainties (e.g. power, flow) at the P-8 reactor condition are larger than those incorporated into the MPS3-specific statistical DNBR limit (see Attachment 3 of the subject LAR.) Therefore, the MPS3-specific plant uncertainties are directly incorporated into the VIPRE-D initial condition in the direction that challenges margin to the DNBR design limit.

RAI 3 (AQ B.1)

Methodologies and Acceptance Criteria for the Rod Ejection Analysis

Section 3.4 of Attachment 1 to the LAR discusses the methodologies and acceptance criteria for the MPS3 rod ejection analysis (REA). It indicates that the MPS3 REA analysis for GAIA fuel, using the methodologies in topical report (TR) ANP-10338P-A, was performed against the acceptance criteria in Regulatory Guide (RG) 1.236, "Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents." (ML20055F490). Table 1 on page 8/25 in Attachment 1 to the LAR lists the proposed acceptance criteria for the REA analysis.

High-Temperature Cladding Failure Threshold

The licensee proposed in row 2 of Table 1, the limit in Figure 1 of RG 1.236 as the acceptance criterion for the high-temperature cladding failure threshold. Regulatory Position (RP) 3.1 of RG 1.236 restricts the applicability of the Figure 1 limit to events with prompt critical excursions (i.e., ejected rod worth greater than or equal to $1.00). Provide a discussion to address the compliance with the above restriction for use of the Figure 1 limit.

DENC Response to RAI-3

Per Regulatory Guide (RG) 1.236 (ADAMS Accession No. ML20055F490), page 5, "if the reactivity worth of the ejected control rod is large enough, the reactor will become prompt critical." In this regard, reactivity (in units of $) is defined as rho (i.e., ejected rod worth) divided by β_{eff}.

An event is prompt critical when rho is greater than β_{eff} ("Nuclear Reactor Analysis", by J. J. Duderstadt, L. J. Hamilton, copyright 1976, page 246). This definition of prompt critical is also used in the MPS3 AREA (ARCADIA Rod Ejection Accident) analysis. If the ejected rod worth is equal to β_{eff}, this is equivalent to $1.00 of reactivity. In this context, **[**

] The enthalpy and enthalpy rise information is reported for prompt critical cases in the MPS3 AREA analysis. Therefore, the MPS3 AREA analysis is compliant with the restriction for use of RG 1.236 Figure 1.

RAI 4 (AQ B.3)

Pellet-cladding mechanical interaction (PCMI)Threshold

The licensee proposed in row 4 of Table 1, the limits in Figures 2 and 4 of RG 1.236 as the acceptance criterion for the PCMI threshold. RP 3.2 of RG 1.236 restricts the applicability of Figures 2 and 4 to recrystallized annealed (RXA) cladding type. Address the compliance with the above restriction for use of Figures 2 and 4 limits.

DENC Response to RAI-4

The application of RG 1.236, Figures 2 and 4 for pellet-cladding mechanical interaction (PCMI) failure threshold is restricted to recrystallized annealed (RXA) cladding type. The GAIA fuel design uses M5 cladding as described in Section 2.1 of Topical Report (TR) ANP-10342P-A, Revision 0, “GAIA Fuel Assembly Mechanical Design,” (ADAMS Accession No. ML19309D916). M5 is a RXA cladding type as shown in Section 5.0 of TR BAW-10227P-A, Revision 1, “Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel,” (ADAMS Accession No. ML15162B052, proprietary). Therefore, the use of RG 1.236, Figures 2 and 4 as the acceptance criterion for the PCMI threshold in MPS3 AREA analysis is appropriate.

RAI-5 (AQ B.4)

Fuel Melt Temperature Limits

The licensee proposed in row 5 of Table 1, the limits “less than 4754°F, decreasing linearly by 13.7°F per 10,000 MWD/MTU of burnup; rim melt is precluded,” as the acceptance criterion for the fuel centerline melt temperature limits.

Provide derivation of the proposed fuel melt temperature limits from the best estimate (BE) fuel melt temperature limit relationship with consideration of the data measurement uncertainty and calculational uncertainty. Also, provide evidence showing that the BE fuel melt temperature limit relationship, data measurement uncertainty, and calculational uncertainty were previously approved by NRC. The NRC staff observed that the overall approach discussed in the response to Audit Question 9.B4 seems consistent with Section 6.8.3 of ANP-10338P-A, which was previously approved by the NRC, but the evidentiary trail is not well marked. For example, ANP-10323P-A, Rev. 1 doesn’t contain the melt limit itself or a data-based rationale for it; Section 4.2.4.7.1 of ANP-10339P-A has the melt limit (which the NRC previously accepted) but lacks a clear explanation or basis; and Section 6.8.3 of ANP-10338P-A also appears to lack a basis for its values. Consider the above observations when preparing the response to this RAI. Also, if the calculated fuel melt temperature limits are different from the proposed limits in the LAR, identify the impact of use of the calculated fuel temperature limits on the REA analysis results in the LAR.

DENC Response to RAI-5

1. Evidence Supporting the MPS3 AREA Fuel Melt Limit

The ANP-10338P-A, Revision 0, “AREA-ARCADIA Rod Ejection Accident,” TR was approved by the NRC in December 2017 (ADAMS Accession No. ML18059A782). RG 1.236 was not issued at the time ANP-10338P-A, Revision 0, was approved by the NRC. Thus, no quantitative fuel melt limit was given in ANP-10338P-A. Rather, the process used to calculate the fuel melt temperature limit is described in Sections 6.2.3 and 6.8.3 of ANP-10338P-A. The process consists of adjusting the fuel melt temperature function from GALILEO by the melt temperature uncertainty and temperature prediction uncertainty. The fuel melt limit is established at the time of plant specific application.

As described in Section 2 of this RAI response, the MPS3 AREA fuel melt limit is calculated consistent with Sections 6.2.3 and 6.8.3 of ANP-10338P-A. As stated in Section 6.8.3 of ANP-10338P-A, the process to calculate the fuel melt limit includes three parts: (1) the best estimate fuel melt temperature relationship obtained from GALILEO, (2) a melt temperature uncertainty, and (3) a temperature prediction uncertainty. Each part is discussed below.

1.1. Best Estimate Fuel Melt Temperature Relationship

The AREA TR (ANP-10338P-A) states the fuel melt temperature relationship is obtained from GALILEO and references Revision 0 of the GALILEO TR (ADAMS Accession No. ML13218A015). The clarification regarding the GALILEO code version used for MPS3 AREA application is given in the MPS3 AREA analysis. Revision 1 of GALILEO TR (ADAMS Accession No. ML21005A030) is used as the fuel performance code in the MPS3 AREA analysis.

Revision 1 of the GALILEO TR (ANP-10323P-A) does not explicitly contain the fuel melt temperature relationship. The NRC Final SE for ANP-10323P-A indicates in Section 1.0 and Table 1 (pages 2 through 4) that the methodology used to assess Fuel Centerline Melt (FCM) Analysis is addressed within ANP-10339P-A, Revision 0, “ARITA-ARTEMIS/RELAP Integrated Transient Analysis Methodology.” NRC review of the ARITA TR (ANP-10339P) was underway when the Final SE for GALILEO TR, Revision 1 was released.

NRC-approved ARITA TR (ANP-10339P-A, Revision 0, ADAMS Accession No. ML24026A140) provides the GALILEO fuel pellet melt temperature relationship as a function of burnup in Section 4.2.4.7.1 (page 4-34). **[**

] This is consistent with the **[**
] that was described in the AREA TR, Section 6.8.3. This is further supported by comparing the GALILEO equation in ANP-10339P-A, Section 4.2.4.7.1, to the clearly defined best estimate fuel melt equation in BAW-10231P-A, Revision 1, “COPERNIC Fuel Rod Design Computer Code,” (ADAMS Package Accession No. ML042930233). BAW-

10231P-A, Revision 1, Chapter 12.3.1 provides the best estimate fuel melt equation used within COPERNIC (page 12-6).

There are two (2) differences between the presented equations in the ARITA and COPERNIC TRs. The first difference is the multipliers for the burnup penalty. The difference between the multipliers is due to the use of different burnup units (i.e., GWd/MTU versus MWd/MTU). **[**

].

Thus, the GALILEO fuel temperature relationship (**[** **]**) has been approved by the NRC in ANP-10339P-A. Additional details supporting NRC-review and approval of the COPERNIC fuel melt temperature in BAW-10231P-A are provided below.

Chapter 12 of BAW-10231P was submitted for NRC review and approval by letter dated December 2, 1999 (ADAMS Package Accession No. ML993470325). BAW-10231P-A, Revision 0 (Chapters 1 through 12) received NRC approval by letter dated June 14, 2002 (ADAMS Accession No. ML021360461). At a later time, in 2004, NRC approved Chapter 13 of BAW-10231P. This approved version of COPERNIC TR was published as Revision 1 to distinguish it from BAW-10231P-A, Revision 0, which was issued after NRC approval of Chapters 1 through 12 (ADAMS Accession No. ML042930236).

The background data used to develop the fuel melt temperature relationship in COPERNIC is provided in the Question 24 RAI response in Appendix A (page 14-61) of BAW-10231P-A, Revision 1. The GALILEO code builds on experience from the approved COPERNIC code, **[** **]** The GALILEO TR ANP-10323P-A, Revision 1, Section 2.2.1, states that the fuel rod database used for calibration and validation of the GALILEO code was developed through a merging of databases associated with the COPERNIC, RODEX4, and CARO-E3 fuel thermal-mechanical codes, and supplemented with more recent data as available.

1.2. <u>Melt Temperature Uncertainty</u>

[**]** was approved in ANP-10338P-A, Revision 0 and ANP-10339P-A, Revision 0, as outlined above.

1.3. <u>Temperature Prediction Uncertainty</u>

[**]** in AREA TR (ANP-10338P-A, Revision 0, Section 6.8.3). A deviation from ANP-10338P-A is identified in Attachment 1, Section 3.4 (with Proprietary Insert 1 in Attachment 4) of the subject LAR, dated October 30, 2023. The deviation is the use of an **[**
] in the MPS3 AREA analysis (ANP-4058P, Revision 0). The deviation is justified in

ANP-4058P, Revision 0, “Millstone Unit 3 Rod Ejection Accident Analysis,” and is provided below.

Excerpt from ANP-4058P, Revision 0, Section 1.4, AREA Methodology Deviations, page 1-6

“Section 6.8.3 of the Reference 1 methodology states that a **[**

]”

Excerpt from ANP-4058P, Revision 0, Section 8.0, References, page 8-1

“8.0 REFERENCES

1. ANP-10338P-A, Revision 0, AREA – ARCADIA Rod Ejection Accident, December 2017.

…

3. ANP-10323P-A, Revision 1, GALILEO Fuel Rod Thermal Mechanical Methodology for Pressurized Water Reactors, November 2020.

4. ANP-10323P, Revision 0, Fuel Rod Thermal-Mechanical Methodology for Boiling Water Reactors and Pressurized Water Reactors, July 2013.”

The same deviation was applied by Callaway Plant, Unit 1, in their AREA analysis supporting the use of a limited number of Framatome GAIA fuel assemblies. ANP-4012P/NP, Revision 2, “Callaway Rod Ejection Accident Analysis”, specifies the methodology deviation / clarification for the **[ ]** on page 1-2. The ANP-4012P/NP AREA Summary Report was included in Callaway’s LAR supplement submittal dated August 3, 2023 (ADAMS Package Accession No. ML23215A196). The conclusions on pages 74-75 of the SE for Callaway Plant, Unit 1, Amendment 235 (ADAMS Accession No. ML23240A369) support approval of methodologies and analyses that incorporate the same deviation for **[ ]** as was used in the MPS3 AREA analysis.

2. Derivation of the MPS3 AREA Fuel Melt Limit

The fuel melt limit derivation for the MPS3 AREA analysis is included in this section.

2.1. The fuel melt temperature relationship for UO_2 as a function of burnup is given by the following equation, in Celsius, per ANP-10339P-A, Section 4.2.4.7.1. Comparison of Equation 1 to the best estimate fuel melt equation identified in BAW-10231P-A, Revision 1, Chapter 12.3.1, shows that Equation 1 below incorporates the **[** **]**.

[**]** Eq. 1

where

- BU is burnup in MWd/MTU.

2.2. ANP-10338P-A additionally requires application of a temperature prediction uncertainty. The **[** **]** is discussed in the MPS3 AREA analysis (see excerpt in Section 1.3 above) as a deviation to the MPS3 application of ANP-10338P-A. The temperature prediction uncertainty is applied as shown below.

[**]** Eq. 2

[**]** Eq. 3

$$T_{Melt_{MPS3\ AREA}}(\mathrm{UO_2}, BU, °C) = 2623.72\ °C - 7.096 \times 10^{-4} * BU \qquad \text{Eq. 4}$$

$$T_{Melt_{MPS3\ AREA}}(\mathrm{UO_2}, BU, °F) = 4754.7\ °F - 12.8 \times 10^{-4} * BU \qquad \text{Eq. 5}$$

where

- BU is burnup in MWd/MTU.

Thus, the MPS3 AREA fuel melt temperature is given as “4754.7°F (rounded down to 4754°F), decreasing linearly by 12.8°F per 10,000 MWd/MTU”. Adjustments for burnable poison content are not reflected in this derivation. They are established by following ANP-10339P-A, Revision 0, Section 4.2.4.7.1.

3. MPS3 AREA Fuel Melt Limit Correction

As described in Section 2 of this RAI response, the MPS3 AREA fuel melt temperature limit is given as “4754°F, decreasing linearly by 12.8°F per 10,000 MWd/MTU”. The burnup dependent term of “12.8°F” is different from the value of “13.7°F” reported in Attachment 1, Table 1, Row 5 of the subject LAR, dated October 30, 2023. Thus, this RAI response corrects the MPS3 AREA fuel melt limit burnup dependent term from 13.7°F to 12.8°F. A markup of Attachment 1, Table 1 from the subject LAR is provided as Attachment 3 to this letter.

Framatome evaluated the impact of this error on the MPS3 AREA analysis results and concluded that there is no impact on the analysis results due to this error. This error was limited to the reporting of the MPS3 AREA fuel melt limit, specifically to the burnup dependent term of 13.7°F, in the subject LAR, dated October 30, 2023.

RAI-6 (AQ C.1)

Application of ANP-10338P-A to GAIA Fuel in MPS3 Cores

Section 3.4 of Attachment 1 to the LAR, discusses the application of the REA methodologies in ANP-10338P-A to GAIA fuel in MPS3. It addresses the compliance with the limitations and conditions (L&C) imposed in the SEs approving the TRs applicable to REA application.

The licensee indicates in Section 3.4 of Attachment 1 that an L&C of ANP-10342, "GAIA Fuel Assembly Mechanical Design," is applicable to the REA application. The L&C states that the most up-to-date guidance and analytical limits should be considered when demonstrating acceptable performance of GAIA under reactivity-initiated accident conditions and highlights the recently issued REA guidance in RG 1.236. Also, the licensee indicates that the application of the REA methodology is compliant with RG 1.236 and thus, satisfies the L&C. RP 2.2.1 of RG 1.236 provides guidance for selection of PWR REA initial conditions. Specific guidance for selection of the initial conditions important to the REA analysis is discussed in RPs 2.2.1.1 through 2.2.1.13.

Provide a discussion to show that the REA analysis for MPS3 cores with GAIA meets each of the PWR REA initial conditions in RPs 2.2.1.1 through 2.2.1.13 of RG 1.236. Identify the specified initial conditions that are not met and provide rationale to justify each identified non-compliant initial condition.

DENC Response to RAI-6

The MPS3 AREA analysis applied the NRC-approved ANP-10338P-A methodology to demonstrate compliance with the RG 1.236 acceptance criteria for a postulated PWR control rod ejection (CRE) accident. The discussion provided outlines how the NRC-approved ANP-10338P-A methodology and MPS3 AREA analysis address the thirteen PWR CRE initial conditions (Regulatory Position (RP) 2.2.1.1 – 2.2.1.13) given in RG 1.236.

2.2.1.1

Accident analyses should consider the full range of cycle operation from beginning of cycle (BOC) to end of cycle (EOC).

Response

Per methodology (Section 6.4 of ANP-10338P-A, pg. 6-6) the analysis is performed at several times in life (TIL) and power levels. The selection of the TILs is made based on the steady state depletions performed from which the peaking information are extracted and used in the selection process. Power levels are chosen to maximize ejected rod worths. The TILs were chosen to correspond to BOC, EOC, and sufficient Middle of Cycle (MOC) conditions required to ensure a conservative analysis.

2.2.1.2

Accident analyses at hot zero power (HZP) should encompass both (1) BOC following core reload and (2) restart following recent power operation.

Response

In general, the consequences of the rod ejection event are more severe when skewed xenon conditions exist that increase local peaking by creating top or bottom peaked axial power distributions. As such, conservative representation of xenon distribution with respect to restart following recent power operation is addressed explicitly in the AREA method (see Section 7.1.4.6 of ANP-10338P-A, pg. 7-11 for more discussion). Initiation of the rod ejection event from zero power at true BOC (following core reload) without significant xenon present has the effect of producing an initial condition with a more positive or less negative moderator temperature coefficient, because of the higher boron concentration. **[**

]

2.2.1.3

Accident analyses should consider the full range of power operation including intermediate power levels up to hot full-power (HFP) conditions. These calculations should consider power-dependent core operating limits (e.g., control rod insertion limits, rod power peaking limits, axial and azimuthal power distribution limits). At conditions where certain core operating limits do not apply, the analysis should consider the potential for wider operating conditions resulting from xenon oscillations or plant maneuvering.

When properly justified, cycle-independent bounding evaluations that demonstrate that regions of power operation are less limiting are an acceptable analytical approach to reduce the number of cases analyzed. For example, during CRE scenarios initiated from at-power conditions, credit for power-dependent insertion limits in the technical specifications may be used to demonstrate that these particular events are of less significance with respect to coolable geometry.

Response

The power levels selected cover HZP, HFP, and intermediate power levels (Section 6.4 of ANP-10338P-A, pg. 6-6). Based on guidance from the AREA topical report, the operating space as defined by the TS is accounted for in the AREA analyses. The operating space includes, but is not limited, to rod insertion limits, rod position uncertainty, and axial power distribution limits. When conditions exist that are not covered by certain core operating limits the method explicitly includes the potential for wider operating conditions. The main example of this is found in the treatment of **[** **]** This approach is in alignment with RG 1.236.

The RG 1.236 approach for reducing the number of cases analyzed is not applied in the NRC-approved ANP-10338P-A methodology. Rather, the power levels are chosen to maximize the ejected rod worth as stated in the response to Item 2.2.1.1. This includes the use of the Rod Insertion Limits to determine the limiting power levels. Additionally, the times in cycle evaluated are chosen to maximize peaking. This combination of maximum ejected rod worth and high peaking provides for a conservative analysis.

<u>2.2.1.4</u>
Uncontrolled worth of an ejected rod should be calculated based on the following conditions: (1) the range of control rod positions allowed at a given power level and (2) additional fully or partially inserted misaligned or inoperable rod(s) if allowed. Applicants do not need to consider dropped or misaligned rods which are being recovered within technical specifications limiting conditions for operation (TS LCO) completion times.

Sufficient parametric studies should be performed to determine the worth of the limiting control rod for the allowed configurations highlighted above. The evaluation methodology should account for (1) calculation uncertainties in neutronic parameters (e.g., neutron cross sections) and (2) allowed power asymmetries.

<u>Response</u>
As discussed in Section 6.4 of ANP-10338P-A, pg. 6-7, the selection of the rod to be ejected is the **[** **]** The process for the determination of the rod with the **[** **]** includes consideration for the range of allowed initial positions for the power level of interest. The ejected rod worth calculation within the AREA analysis includes the control rod position uncertainty and is **[** **]** as discussed in Section 7.1.4.1 of ANP-10338P-A, pg. 7-7. Cross section adjustment factors (Section 6.5.3 of ANP-10338P-A, pg. 6-12) **[** **]**

For rods in the **[** **]** All other rods are at the All Rods Out (ARO) position, and a misalignment does not impact the AREA analysis.

Inoperable rods do not impact rod ejection but would impact the scram worth. This impact is small and has no consequence to the ability to reduce power following scram.

The AREA methodology does not consider power **[** **]**

The core biasing strategies which include ejected rod worth are shown in Table 7-2 (pg. 7-25) of ANP-10338P-A.

<u>2.2.1.5</u>
Because of burnup-dependent and corrosion-dependent factors that tend to reduce cladding failure thresholds and allowable limits on core coolability during fuel rod lifetime, the limiting initial conditions may involve locations other than the maximum uncontrolled rod worth defined in Regulatory Position C.2.2.1.4 (e.g., uncontrolled rod motion at a core location adjacent to higher burnup fuel assemblies). For this reason, a more comprehensive search for the limiting conditions may be necessary to ensure that the total number of fuel rod failures is not underestimated, and allowable limits are satisfied. Applicants may need to survey a larger population of PWR ejected rod core locations and exposure points to identify the limiting scenarios.

When properly justified, combining burnup-dependent parameters to create an artificial, composite worst time-in-life (e.g., end-of-life cladding hydrogen content combined with maximum ejected worth) is an acceptable analytical approach to reducing the number of cases analyzed.

<u>Response</u>
As discussed in Section 6.4 of ANP-10338P-A, pg. 6-7 the AREA method selects the **[** **]** in the simulation. The discussion in Section 6.4 acknowledges that **[** **]** This philosophy is applied to other acceptance criteria via investigation of parameters such as **[** **]** where conditions may exist that have a **[** **]**

A comprehensive search for the limiting conditions was **[** **]**

<u>2.2.1.6</u>
The reactivity insertion rate should be determined from differential control rod worth curves and calculated transient rod position versus time curves.

Response
The AREA analysis does not use the maximum differential rod worth or a calculated transient rod position versus time curve to calculate the reactivity insertion rate. This is because the AREA analysis uses a full core transient 3D nodal solver (ARTEMIS) that calculates the core reactivity as the rod is ejected rather than relying on a differential control rod worth.

ARTEMIS is one of the constituent codes used within the ARCADIA Reactor Analysis System. ARCADIA is the NRC-approved code suite used in the MPS3 AREA analysis (ANP-10297P-A, Rev. 0 (ADAMS Package Accession No. ML14195A145) and ANP-10297, Supplement 1P-A, Rev. 1, (ADAMS Accession No. ML21071A064.))

For a scram, the calculated reactivity insertion rates are based upon control rod scram position versus time curves provided by the licensee (Section 6.5.1 of ANP-10338P-A, pg. 6-9).

The approach used to calculate the reactivity insertion rate is compliant with the AREA approved method. The approach used in the reactivity insertion rate calculation is different from the RG 1.236 but meets its intent.

2.2.1.7
The rate of ejection should be calculated based on the maximum pressure differential and the weight and cross-sectional area of the control rod and drive shaft, assuming no pressure barrier restriction.

Response
The AREA analysis does not use the maximum pressure differential and the weight and cross-sectional area of the control rod and drive shaft to calculate the ejection rate. The AREA analysis assumes that a fully inserted control rod is ejected within 0.1 seconds consistent with the approved methodology (see Section 6.5 of ANP-10338P-A, pg. 6-8). This is a typical value used for rod ejection analyses in the industry (see MPS3 FSAR Section 15.4.8.2, Callaway FSAR Section 15.4.8.2, Byron and Braidwood FSAR Section 15.4.8.2).

The assumed rate of ejection is compliant with the AREA approved method. The approach taken for quantifying the rate of ejection is different from the RG 1.236 but meets its intent.

2.2.1.8
The initial reactor coolant pressure, core inlet temperature, and flow rate used in the analysis should be conservatively chosen, depending on the transient phenomenon being investigated. The range of values should encompass the allowable operating range and monitoring uncertainties.

Response
Per methodology (Section 7.1.4 of ANP-10338P-A, pg. 7-7), the initial reactor coolant pressure (Section 7.1.4.15, pg. 7-18), core inlet temperature (Section 7.1.4.14, pg. 7-17), and flow rate

(Section 7.1.4.13, pg. 7-16) were considered in the case matrix. These parameters include uncertainties and conservative biases. The core biasing strategy for these parameters is shown in Table 7-2 (pg 7-25) of ANP-10338PA.

The AREA analysis used conditions for pressure, flow rate, and core inlet temperatures that encompass the operating range and monitoring uncertainties. The plant operating ranges are within the range of applicability of the thermal hydraulic methods and codes. These initial plant conditions were conservatively biased (including monitoring uncertainties) in accordance with requirements of ANP-10338P-A, to generate conservative results depending on the transient phenomenon being investigated.

2.2.1.9

Fuel thermal properties (e.g., fuel-clad gap thermal conductivity, fuel thermal conductivity) should cover the full range over the fuel rod's lifetime and should be conservatively selected based on the transient phenomenon being investigated. Time-in-life specific fuel properties may be used for a given burnup-specific statepoint analysis.

Response

Per methodology (Section 7.1.4 of ANP-10338P-A, pg. 7-7), the fuel thermal properties (i.e., gap conductivity (Section 7.1.4.8, pg. 7-12), fuel conductivity (Section 7.1.4.9, pg. 7-14), and heat capacity (Section 7.1.4.10, pg. 7-14)) were considered in the case matrix. These parameters include conservative biases consistent with the core biasing strategies shown in Table 7-2 (pg. 7-25) of ANP-10338P-A.

The AREA analysis employed fuel thermal properties that are within the range of applicability of the ARTEMIS fuel rod model, which is based upon GALILEO. The fuel properties were conservatively biased in accordance with requirements of ANP-10338P-A, to generate conservative results depending on the transient phenomenon being investigated. TIL-specific fuel properties were considered for the burnup-specific TIL analyzed.

2.2.1.10

The moderator reactivity feedback resulting from voids, coolant pressure changes, and coolant temperature changes should be calculated based on the various assumed conditions of the fuel and moderator using standard transport and diffusion theory codes. If boric acid shim is used in the moderator, the highest boron concentration corresponding to the initial reactor state should be assumed. If applicable, the range of values should encompass the allowable operating range (i.e., technical specifications in the core operating limits report) and any applicable analytical uncertainties.

Response

In the context of the AREA analysis, the ARCADIA 3D solver (ARTEMIS) performs an explicit Moderator Temperature Coefficient (MTC) calculation that captures the effects of voids, coolant pressure, and temperature changes based on the conditions of the transient.

The AREA analysis assumed MTCs that encompass the operating range. The bias applied to the moderator temperature feedback is discussed in Section 7.1.4.4 of ANP-10338P-A, pg. 7-9. The moderator temperature feedback biasing is part of the core biasing strategies shown in Table 7-2 (pg. 7-25) of ANP-10338P-A.

The MTC analytical uncertainty used in the AREA analysis is applied according to Section 7.1.4.4 of ANP-10338P-A, pg. 7-9.

The AREA analysis uses a boron concentration that is consistent with the TIL and power level analyzed. The effect of the **[ ]** bias applied in the calculation.

2.2.1.11

Calculations of the Doppler reactivity feedback should be based on and compared with available experimental data. Since the Doppler feedback reflects the change in reactivity as a function of fuel temperature, uncertainties in predicting the coefficient, as well as in predicting fuel temperatures at different power levels, should be reflected by conservative application of Doppler feedback.

Response

Doppler reactivity was considered during the evaluation of the ARCADIA topical report (Section 11.2.2 of ANP-10297P-A, Revision 0 (ADAMS Package Accession No. ML14195A145) and SE Section 3.5.3 of ANP-10297, Supplement 1P-A, Revision 1 (ADAMS Accession No. ML21071A064)). The benchmarks demonstrate that the ARCADIA 3D solver (ARTEMIS) can accurately predict Doppler Temperature Coefficient (DTC). The uncertainty applied to the Doppler temperature feedback is discussed in Section 7.1.4.3 of ANP-10338P-A, pg. 7-8. The Doppler temperature feedback biasing is part of the core biasing strategies shown in Table 7-2 (pg. 7-25) of ANP-10338P-A.

2.2.1.12

Control rod reactivity insertion during trip versus time should be obtained by combining the differential rod worth curve with a rod velocity curve based on maximum design limit values for scram insertion times. Alternatively, reactivity may be calculated using control rod velocity during trip based on maximum design limit values for scram insertion times. Any loss of available scram reactivity resulting from allowable rod insertion should be quantified.

Response

The AREA analysis uses the ARCADIA 3D solver (ARTEMIS) that calculates the negative reactivity insertion during the trip based on a scram curve consistent with the plant licensing basis. Rod position with respect to time is used in the core simulator (Section 6.5.1 of ANP-10338P-A, pg. 6-9).

The scram curve input to the MPS3 AREA analysis is given as a function of control rod drop distance versus time (control rod velocity). The TS maximum allowed rod drop time from the fully withdrawn position to dashpot entry is modeled in the curve, producing maximum scram times.

The loss of available scram reactivity due to partially inserted control rods is explicitly handled by the 3D neutronic simulator, ARTEMIS.

2.2.1.13

The reactor trip delay time, or the amount of time that elapses between the instant the sensed parameter (e.g., pressure, neutron flux) reaches the level for which protective action is required and the onset of negative reactivity insertion, should be based on maximum values of the following: (1) time required for the instrument channel to produce a signal, (2) time for the trip breaker to open, (3) time for the control rod motion to initiate, and (4) time required before control rods enter the core if the tips lie outside the core. The response of the reactor protection system should allow for inoperable or out-of-service components and single failures.

Response

Per methodology (Section 6.5.1 of ANP-10338P-A, pg. 6-8), the excore detector model, the signal processing (based on the excore flux signal), and the control rod scram model are implemented in the ARCADIA 3D core simulator (ARTEMIS) trip function.

The AREA analysis uses a high flux trip setpoint. Furthermore, a trip signal must occur for two consecutive time steps before trip will occur. The AREA analysis uses a 2 out of 4 trip logic for excore trip. A single failure is assumed for the first detector with a second detector out of service (being set as tripped), such that trip signal occurs on the third detector. In the 3D core solver, this is the equivalent of having 3 out of 4 detectors at the trip condition. The analysis includes conservative inputs for reactor protection system setpoints and delay times.

The RTS setpoints input to the MPS3 AREA analysis reflect the TS values with uncertainty applied to delay trip actuation.

Additionally, a time delay is employed before the control rods are dropped (Section 7.2.1.3 of ANP-10338P-A, pg. 7-21). The TS defines RTS response time as the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The acceptance criteria for RTS response times contained in the licensee-controlled Technical Requirements Manual (TRM) are input to the MPS3 AREA analysis. The input captures the (1) time required for the instrument channel to produce a signal, (2) time for the trip breaker to open, and (3) time for the control rod motion to initiate.

Item (4) is handled by the neutronic simulator, ARTEMIS, which tracks the position of the control rods explicitly.

RAI-7 (AQ C.2)

Compliance with L&Cs of the AREVA TR Related to REA Analysis

The licensee indicates on page 15 of 25 of Attachment 1 that one deviation was taken from the NRC-approved ARCADIA rod ejection accident (AREA) methodology. The deviation is included in Attachment 4 of the LAR as "Proprietary INSERT 1". Discuss the effect of the adopted deviation on the proposed melt temperature limits and provide bases to justify that the adopted deviation is acceptable for the REA analysis.

DENC Response to RAI-7

The deviation included in Attachment 4 of the October 30, 2023, LAR as "Proprietary INSERT 1" is discussed and justified in DENC's response to RAI-5, Section 1.3. The RAI-5 response includes the effect of the deviation on the fuel melt temperature limit and illustration of a previous NRC approval of the same deviation for AREA application to Callaway. The MPS3 AREA fuel melt temperature limit derivation presented in DENC's response to RAI-5, Section 2, uses the deviation included in Attachment 4 of the subject LAR.

RAI-8 (AQ C.3)

Calculated Values for MPS3 AREA Analysis

The licensee states on page 15 of 25 of Attachment 1 that "MPS3 AREA analysis results demonstrate margin to the limits for fuel temperature, fuel rim temperature, enthalpy, and enthalpy rise. DNB fuel rod failures are predicted, but the failure total remains within the assumptions of the current REA radiological analysis described in MPS3 FSAR (final safety analysis report) Chapter 15.4.8".

Provide the values of the calculated peak fuel temperature, fuel rim temperature, enthalpy, and enthalpy rise for the applicable limiting cases to show that the values meet the associated limits for the REA analysis. Also, discuss how the total number of DNB failure rods was calculated and specify the calculated total number of failure rods to show it is less than the number assumed in the current REA radiological analysis described in MPS3 FSAR.

DENC Response to RAI-8

MPS3 AREA transient case results for (1) fuel temperature, (2) fuel rim temperature, (3) enthalpy, (4) enthalpy rise, and (5) DNB are provided in Figures 1-5, respectively. A limiting transient case for each AREA acceptance criterion is provided. The result figures are normalized with respect to the parameter limits. A negative value for a relative result indicates the transient response does not violate the acceptance criterion. Positive relative values indicate a violation of the limit. Figures 1-4 illustrate the MPS3 rod ejection accident (REA) analysis meets the associated limits for fuel temperature, fuel rim temperature, enthalpy, and enthalpy rise through the reporting of negative relative values.

Sections 6.8 and 6.9 of ANP-10338P-A describe the AREA methodology approach to calculating fuel failure. The MPS3 AREA analysis results demonstrated margin to the limits for fuel temperature, fuel rim temperature, enthalpy, and enthalpy rise. Five transient cases predicted DNB fuel failure. Therefore, DNB fuel failure is the sole contributor to the MPS3 total fuel failure summation.

The MPS3-specific total number of DNB failed rods was calculated using a penalized DNBR Specified Acceptable Fuel Design Limit (SAFDL) of 1.22 for the COBRA-FX/ORFEO-GAIA CHF code/correlation pair. The applicable **[ ]** (XN-NF-82-06(P)(A), Rev.1 & Supplements 2, 4, and 5) is applied to the transient case result **[ ]** A maximum of four assemblies are assumed to experience clad failure equating to 2.1% (= 4 / 193 assemblies * 100%) of all fuel pins in the core. The current MPS3 REA radiological analysis assumes 10% of the rods experience clad failure and 0.25% experience core melt (MPS3 FSAR Section 15.4.8.4).

Figure 1 – MPS3 RCCA Ejection Accident for GAIA Fuel:
Fuel Temperature Relative to Fuel Melt Temperature
(Fuel Temperature Case: **[ ]**)

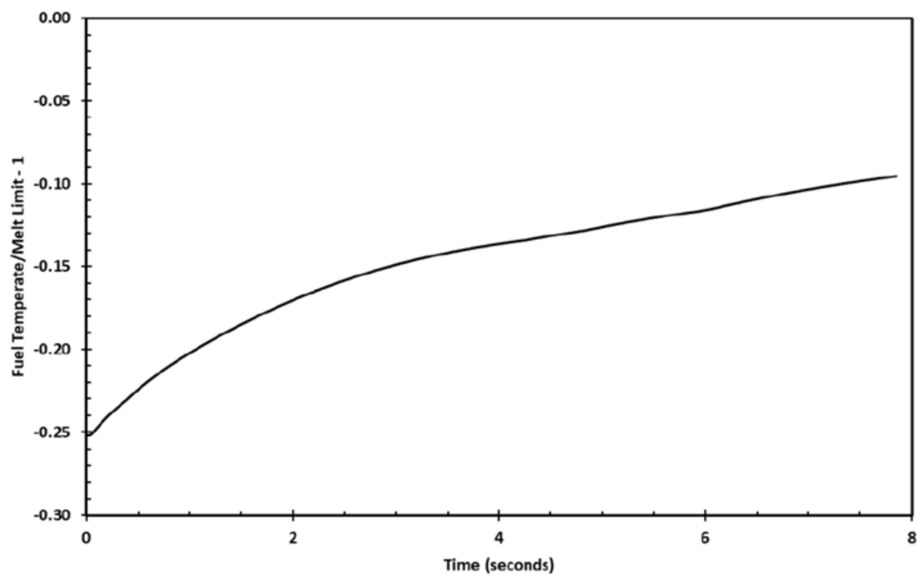


Figure 2 – MPS3 RCCA Ejection Accident for GAIA Fuel:
Fuel Rim Temperate Relative to Fuel Melt Temperature
(Fuel Rim Temperature Case: **[ ]**)

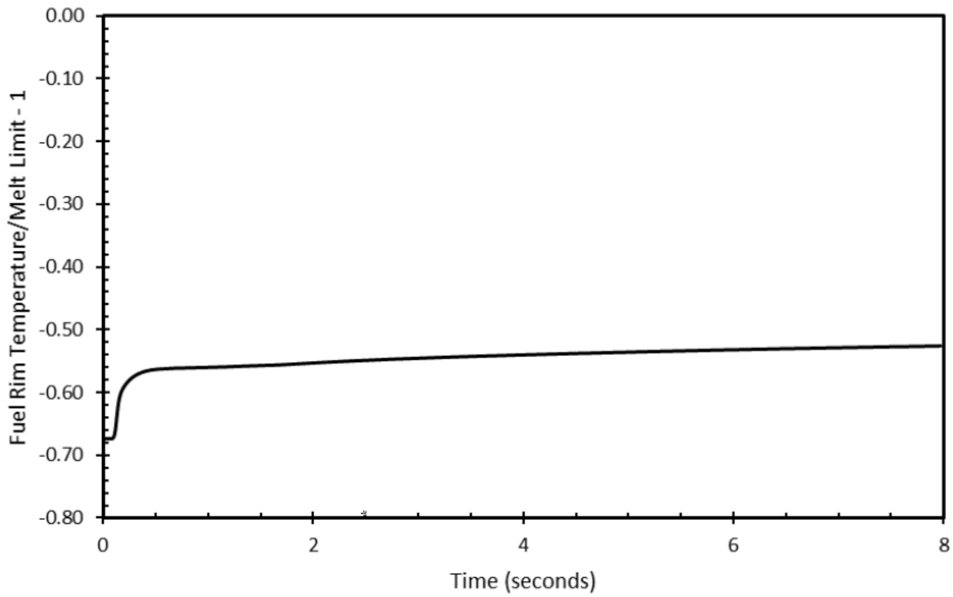


Figure 3 – MPS3 RCCA Ejection Accident for GAIA Fuel:
Fuel Enthalpy Relative to High Temperature Cladding Failure Threshold
(Fuel Enthalpy Case: **[ ]**)

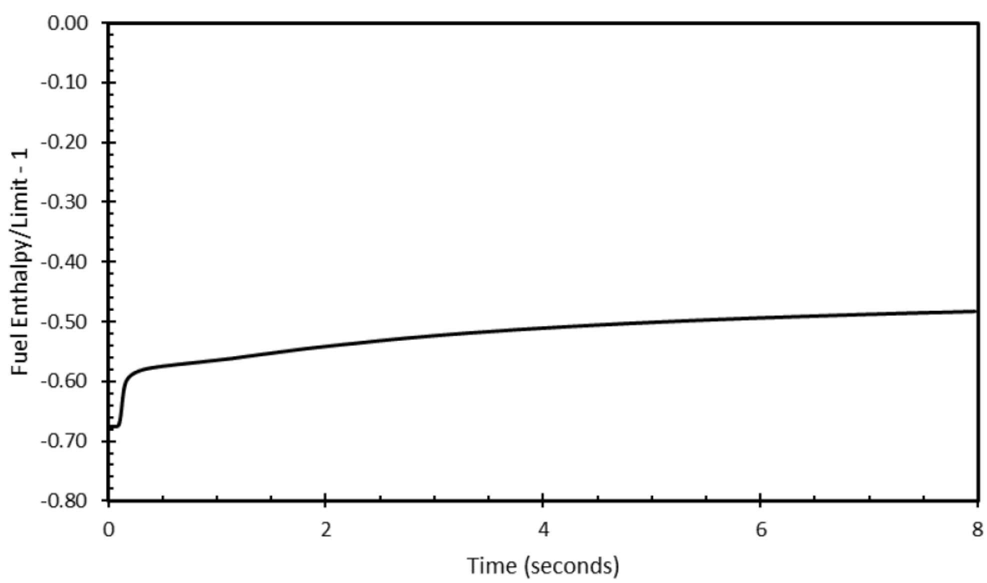
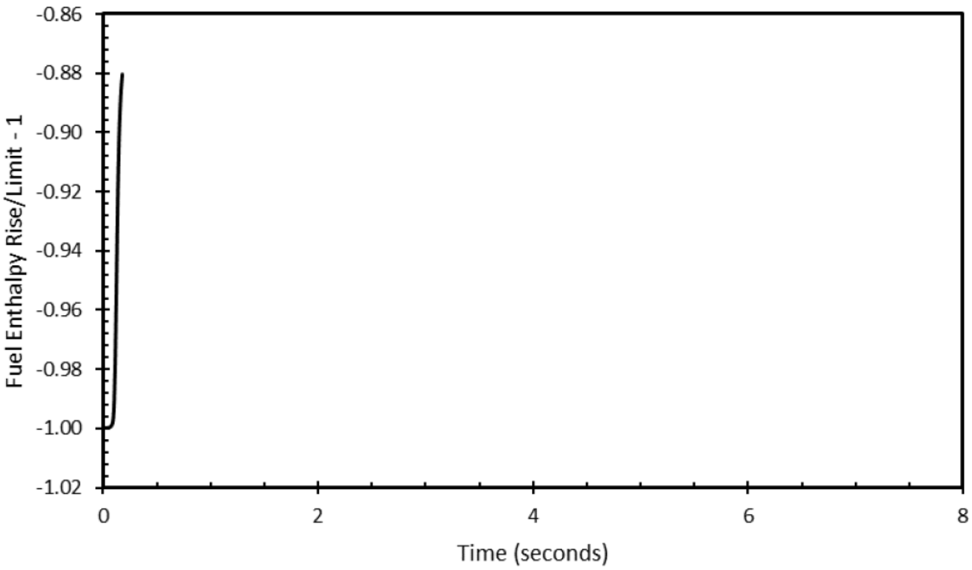
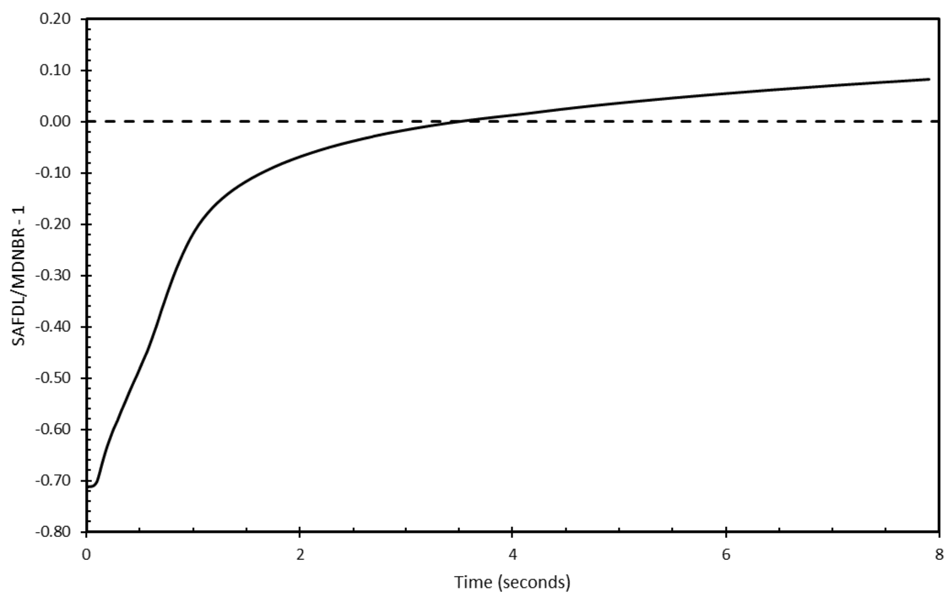


Figure 4* – MPS3 RCCA Ejection Accident for GAIA Fuel:
Fuel Enthalpy Rise Relative to PCMI Cladding Failure Threshold
(Fuel Enthalpy Rise Case: **[** **]**)



* Enthalpy rise is defined as the radial average fuel enthalpy increase from initial conditions to the time corresponding to one pulse width after the peak of the prompt pulse. Thus, the enthalpy increase edit is taken for one pulse width after the time of the peak pulse.

Figure 5 – MPS3 RCCA Ejection Accident for GAIA Fuel:
MDNBR Relative to the DNBR SAFDL
(MDNBR Violation Case: **[** **]**)



RAI-9 (AQ C.4)

REA Overpressure Analysis

The licensee states on page 15 of 25 of Attachment 1 that "if AREA overpressure analysis were performed, it would produce a reduced pressure response compared to the existing analysis described in the FSAR." Please provide rationale (such as the effects of the fuel type and computer codes or methods for use in the analysis on the pressure response), and/or examples of applicable overpressure analysis to support the quoted statement related to the AREA overpressure response bounded by that in the FSAR.

DENC Response to RAI-9

The AREA topical report includes an NRC-approved method for performance of a rod ejection accident RCS overpressure analysis. A MPS3 rod ejection accident overpressure analysis with GAIA fuel was not performed using the NRC-approved AREA method. Rather, an engineering assessment was performed to demonstrate the conservatism of the existing MPS3 analysis of record methodology, WCAP-7588, Rev. 1-A (ADAMS Accession No. ML20330A094, proprietary), for use of GAIA fuel to MPS3.

Key GAIA fuel and reactivity parameters calculated under the NRC-approved AREA TR were compared to those considered in the WCAP-7588, Rev. 1-A analysis of record. The conclusion that the WCAP-7588, Rev. 1-A results were conservative for implementation of GAIA to MPS3 is based upon this comparison. WCAP-7588, Rev. 1-A presents analysis results for both one-dimensional and three-dimensional analysis and concludes that the one-dimensional kinetics method greatly over-predicts the severity of the transient. This same expectation of conservatism applies to AREA's three-dimensional analysis relative to the one-dimensional method of WCAP-7588, Rev. 1-A. Application of three-dimensional kinetics in the NRC-approved AREA topical report would generate a significantly reduced core power response, and therefore a reduced system pressure response relative to the current MPS3 FSAR one-dimensional methodology. Calculated ejected rod worths from the AREA analysis for use of GAIA at MPS3 are much lower than those considered in WCAP-7588, Rev. 1-A. Because of (1) the margins provided by AREA's lower predicted ejected rod worths and (2) the AREA three-dimensional methodology, the primary system pressure analysis results using GAIA at MPS3 would be less severe than the results of WCAP-7588, Rev. 1-A. Thus, it is concluded that an AREA overpressure analysis would produce a reduced pressure response compared to that of the WCAP-7588, Rev. 1-A analysis of record. Therefore, the existing MPS3 FSAR analysis remains conservative considering GAIA fuel.

MPS3 FSAR Section 15.4.8.2.1 presents the results of the analysis supported by WCAP-7588, Rev. 1-A, indicating that "peak pressure does not exceed that which would cause stress to exceed the faulted condition stress limits". The GAIA fuel assembly design has no impact on the overpressure consequences of this event. Therefore, the transition to Framatome fuel does not change the conclusions of the existing overpressure condition described in MPS3 FSAR Section 15.4.8. Relative to the results from the WCAP-7588, Rev. 1-A methodology, application of the ANP-10338P-A methodology would be expected to reduce the magnitude of the pressure response.

A sample AREA overpressure analysis was performed in Appendix A, Section A.4 of ANP-10338P-A for a representative Westinghouse 4-Loop plant. The results of the sample peak RCS pressure transient in ANP-10338P-A support the assessment above.

In summary, the rod ejection accident overpressure conclusions reached in WCAP-7588, Rev. 1-A remain applicable to GAIA fuel design planned for implementation at MPS3.

RAI-10 (AQ C.5)
Mixed-Core Application of MPS3 REA Analysis

The licensee states on page 16 of 25 of Attachment 1 that "the MPS3 REA analysis incorporates a conservative thermal-hydraulic penalty that accommodates mixed-core changes in flow distribution."

Provide the value of the conservative mixed-core penalty factor and discuss how the value of the conservative penalty factor was determined and justify the use of the value of the penalty factor in the REA analysis.

DENC Response to RAI-10
ANP-10338P-A, Section 6.6.3 states an NRC-approved mixed-core methodology can be used to address the thermal-hydraulic effects resulting from mixed fuel types in AREA methodology application. Section 3.2 of Attachment 1 of the subject LAR submittal, dated October 30, 2023, describes DENC's development of mixed-core DNBR penalties supporting the transition from MPS3's resident Westinghouse fuel product to Framatome GAIA fuel. The GAIA Transition Core Penalty (TCP) requested for NRC-approval and applicable to AREA analysis is 2.4% for the VIPRE-D/ORFEO-GAIA code/correlation pair.

The MPS3 AREA analysis retains 8.19% DNBR margin for the application of generic DNBR penalties. As described in RAI-8, the total number of DNB fuel failures in MPS3's AREA analysis was calculated using a penalized COBRA-FLX/ORFEO-GAIA DNBR limit of 1.22. The modified limit is treated in a manner analogous to the DENC-controlled DNBR Safety Analysis Limit (SAL) described in Attachment 3, Section 4.3 of the subject LAR submittal. The delta between the (1) NRC-approved COBRA-FLX/ORFEO-GAIA DNBR design limit of 1.12 (ANP-10341P-A, Rev. 0, ADAMS Accession No. ML18284A031) and (2) modified COBRA-FLX/ORFEO-GAIA DNBR limit of 1.22 applied for MPS3 AREA determines the available retained DNBR margin:

$$Retained\ Margin\ (\%) = \frac{Modified\ Limit - Design\ Limit}{Modified\ Limit} * 100 \qquad \text{Eq. 6}$$

$$Retained\ Margin\ (\%) = \frac{1.22 - 1.12}{1.22} * 100 = 8.19\% \qquad \text{Eq. 7}$$

For each reload cycle, a thermal-hydraulic calculation determines which generic DNBR penalties apply to the MPS3 FSAR Chapter 15 events and confirms positive DNBR margin exists after application of all penalties. This is done in accordance with the requirements of VEP-FRD-42-A (ADAMS Accession No. ML18012A098) which describes the Dominion Energy fleet reload methodology. The VIPRE-D/OREFO-GAIA TCP of 2.4% will be applied against the 8.19% retained DNBR margin from the AREA analysis for MPS3 cycles that contain both Framatome GAIA and resident Westinghouse RFA-2 fuel. The bounding approach used in TCP development described in Attachment 1, Section 3.2 of the subject LAR justifies the use of the specified ORFEO-GAIA TCP penalty factor in MPS3 REA analysis.

It is noted that two different thermal-hydraulic codes are in use. COBRA-FLX is used by Framatome to calculate DNBR within the AREA analysis, and VIPRE-D is used by DENC to develop the TCPs. These calculations apply the same CHF correlations: ORFEO-GAIA above the first mixing vane grid and ORFEO-NMGRID below. Both codes have received NRC-approval for nuclear core thermal-hydraulic analysis, specifically (1) COBRA-FLX in ANP-10311P-A, Rev. 1, (ADAMS Accession No. ML18103A141), and (2) VIPRE-D in DOM-NAF-2-P-A, Rev. 0 (ADAMS Accession No. ML24170B053).

To be licensed for use, a CHF correlation must be tested against experimental data that span the anticipated range of conditions over which the correlation will be applied. As part of the MPS3 GAIA fuel transition, the experimental test sections used by Framatome to develop the ORFEO-GAIA and ORFEO-NMGRID CHF correlations were analyzed in VIPRE-D, to qualify these CHF correlations for use with Dominion Energy's DOM-NAF-2-P-A methodology. The results of the VIPRE-D data analysis were compared to the equivalent data analysis performed with COBRA-FLX. Tables 1 and 2 summarize the code-to-code comparison performed.

Table 1 – Overall Statistics of the Combined Data Set for the ORFEO-GAIA Correlation

VIPRE-D *		**COBRA-FLX ANP-10341P-A, Table 7-3**	
Average M/P	Standard Deviation	Average M/P	Standard Deviation
1.012	0.052	**[ ]**	**[ ]**

Table 2 – Overall Statistics of the Combined Data Set for the ORFEO-NMGRID Correlation

VIPRE-D *		**COBRA-FLX ANP-10341P-A, Table 7-15**	
Average M/P	Standard Deviation	Average M/P	Standard Deviation
1.004	0.070	**[ ]**	**[ ]**

* Summary of data analysis performed using the VIPRE-D ORFEO-GAIA and ORFEO-NMGRID data sets submitted to the NRC in letter dated July 26, 2023 (ADAMS Accession No. ML23208A092) in support of the DOM-NAF-2, Appendix F approval.

The results show agreement in ORFEO-GAIA and ORFEO-NMGRID CHF predictions between the VIPRE-D and COBRA-FLX thermal-hydraulic codes. The difference in the average measured to predicted (M/P) CHF is **[ ]** for ORFEO-GAIA and is **[ ]** for ORFEO-NMGRID. The standard deviations demonstrate comparable data spreads are obtained.

The agreement in ORFEO-GAIA and ORFEO-NMGRID CHF test results between VIPRE-D and COBRA-FLX indicate code-to-code differences are negligible for these correlations. Therefore, it is reasonable to apply VIPRE-D calculated DNBR penalties to COBRA-FLX DNBR results.

<u>RAI-11</u>

Section 3.2, "DNBR Penalties Supporting the MPS3 Transition to GAIA Fuel," of Attachment 1 to the LAR, states:

> *Specifically, a TCP of 2.4% is proposed for application to DNBR analysis results calculated using the VIPRE-D/ORFEO-GAIA code/correlation pair. A penalty of 2.7% is similarly proposed when using the VIPRE-D/ORFEO-NMGRID code/correlation pair above the GAIA first mixing vane spacer grid. Approval of the penalties allows allocation of retained DNBR margin to address the thermal hydraulic effects of adjacent dissimilar fuels during the transition cycles.*
>
> *DENC investigated the thermal-hydraulic effects from the physical differences between the resident Westinghouse and Framatome GAIA fuel products in mixed core configurations at MPS3. DNBR penalties were developed for both the Westinghouse and Framatome fuel products using the VIPRE-D code and applicable CHF correlations in accordance with the USNRC-approved DOM-NAF-2-P-A methodology in support of mixed-core analysis. TCPs will be accommodated within the retained DNBR margin described in Attachment 3.*

a. *Equation 1 uses the values of $MDNBR_{Transition\text{-}Core}$ and $MDNBR_{Full\text{-}Core}$ to calculate the values of TCP for VIPRE-D/ORFEO-GAIA and VIPRE-D/ORFEO-NMGRID code/correlation pair. Provide details on the calculation of $MDNBR_{Transition\text{-}Core}$ for the two cases.*

b. *Clarify if the values of $MDNBR_{Full\text{-}Core}$ used in Equation 1 for VIPRE-D/ORFEO-GAIA and VIPRE-D/ORFEO-NMGRID code/correlation pair are those listed in Tables 3.6-1 and 3.6-2.*

c. *Provide clarification as to why the MDNBR values listed in Tables 3.6-1 and 3.6-2 are the same for all 9 statepoints.*

d. *Provide the value of $MDNBR_{Full\text{-}Core}$ for resident Westinghouse fuel which is used in this calculation.*

DENC Response to RAI-11

DENC's mixed-core DNBR evaluation supporting the GAIA fuel transition is described within Attachment 1, Section 3.2 of the subject LAR. The mixed-core DNBR evaluation is unrelated to Attachment 3 of the subject LAR which describes VEP-NE-2-A (ADAMS Accession No. ML101330527, proprietary) methodology application to MPS3 cores containing GAIA fuel. Subparts A, B, and D to RAI-11 are addressed below. RAI-11, Subpart C is addressed in further detail in the RAI-12 response.

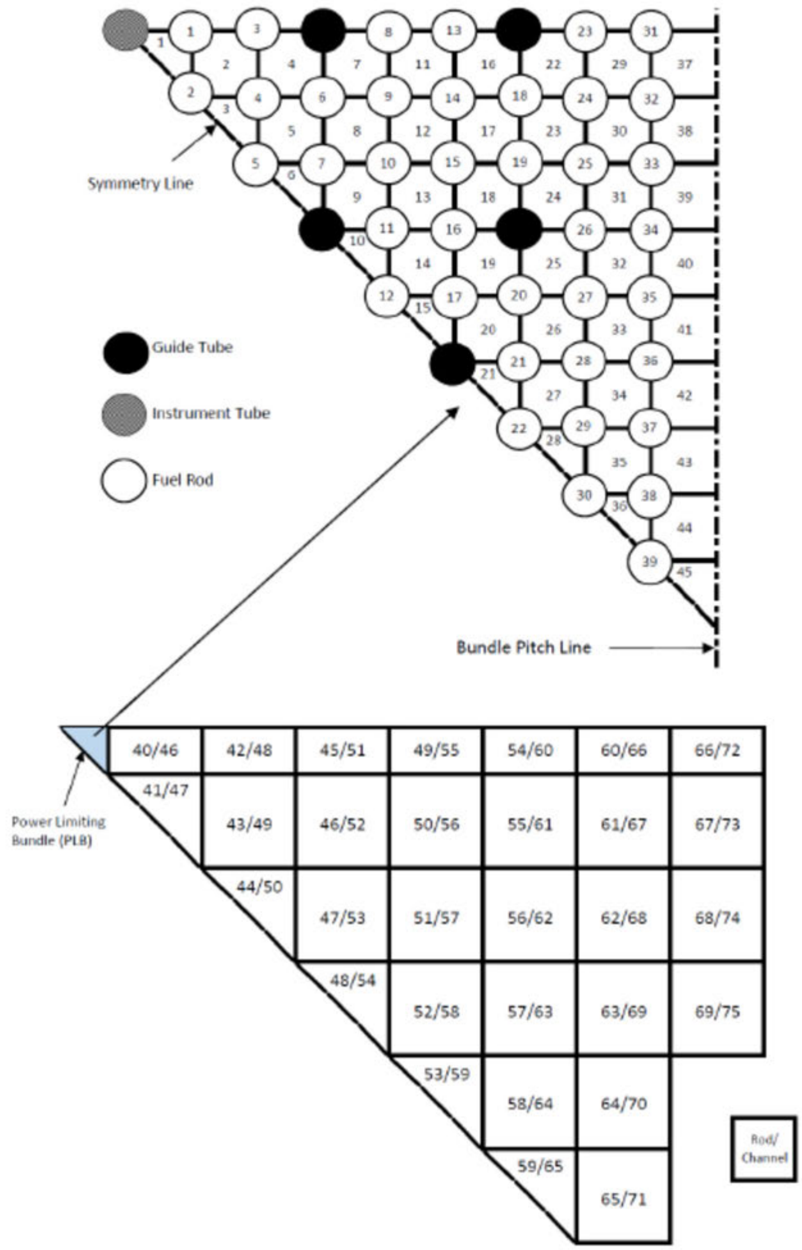
The general approach to the calculation of $MDNBR_{Transition\text{-}Core}$ (Subpart A) and $MDNBR_{Full\text{-}Core}$ is outlined in Section 3.2.1 of the subject LAR. Two sets of VIPRE-D models were developed to perform the MPS3 GAIA fuel transition mixed-core evaluation:

(1) Framatome GAIA fuel model pair
   a. Full-core VIPRE-D model of GAIA
   b. Mixed-core VIPRE-D model; one GAIA fuel assembly is placed in a core of RFA-2 fuel with the single GAIA assembly modeled as the power limiting bundle (PLB). Minimum DNBR (MDNBR) is calculated within the PLB.

(2) Westinghouse RFA-2 fuel model pair
   a. Full-core VIPRE-D model of RFA-2
   b. Mixed-core VIPRE-D model; one RFA-2 fuel assembly is placed in a core of GAIA fuel with the single RFA-2 assembly modeled as the PLB. MDNBR is calculated within the PLB.

An example of these models is provided in Figure 6 which shows a 75-channel model cross section for VIPRE-D. For the figure, channels 1 through 45 represent channels within the PLB fuel assembly and channels 46 through 75 represent lumped fuel assemblies. For a full core model, all channels represent the same fuel assembly type (i.e., Framatome GAIA or Westinghouse RFA-2). For a mixed core model, the PLB is designated as the fuel type of interest (either GAIA or RFA-2) and channels 46 through 75 are modeled as the opposing fuel type.

Figure 6 – Example 75-Channel, 1/8th Core VIPRE-D Model for a 4-Loop, 17 x17 Fuel Plant



The models were developed in accordance with Dominion Energy's NRC-approved DOM-NAF-2-P-A methodology. To ensure predictions of crossflow from higher pressure drop fuel regions to lower pressure drop fuel regions were correctly simulated, the additional considerations outlined in Attachment 1, Section 3.2.2 of the subject LAR were applied in mixed-core models. The limiting mixed-core loading pattern described above was analyzed to provide TCPs that bound the entire core transition period and provide loading flexibility for core designers.

Using a pair of VIPRE-D models above, two DNB cases are run at the same statepoint condition (power, RCS pressure and temperature, RCS flow rate, radial peaking, and axial shape). $MDNBR_{Full\text{-}Core}$ is extracted from the full core case results and $MDNBR_{Transition\text{-}Core}$ is extracted from the mixed-core case results to calculate a TCP using Equation 1 from Attachment 1 of the subject LAR. Equation 1 from the subject LAR is repeated below for reference herein.

$$TCP(\%) = \frac{[(MDNBR_{Transition-Core}) - (MDNBR_{Full-Core})]}{(MDNBR_{Full-Core})} * 100 \qquad \text{Eq. 8}$$

The process was repeated across a range of statepoints. The selected statepoints for the GAIA fuel transition included representative Core Thermal Limits (CTLs) and FSAR Chapter 15 events. The CTLs define the DNB limiting portion of the MPS3 TS 2.1.1 RCSL figure and are designed to meet the DENC-controlled DNBR SAL. DNB results from the FSAR Chapter 15 events vary but must satisfy NRC-approved DNBR design limits. Therefore, $MDNBR_{Full\text{-}Core}$ is not a constant value across all evaluated statepoints (Subpart C).

The statepoints shown in Attachment 3, Tables 3.6-1 and 3.6-2 of the subject LAR support VEP-NE-2-A methodology application and are not utilized within the mixed-core evaluation (Subpart B). See the response to RAI-12 for further clarification of the MDNBR values in Tables 3.6-1 and 3.6-2.

TCPs are calculated for both Framatome and Westinghouse fuel products during the transition period. Therefore, the CHF correlations associated with each fuel type must be evaluated in the PLB where MDNBR is calculated. In mixed-core models, Framatome's CHF correlations are not extended to the DNB evaluation of Westinghouse fuel or vice versa as the DNBR results for the non-PLB assemblies do not factor into the TCP determination. Both $MDNBR_{Full\text{-}Core}$ and $MDNBR_{Transition\text{-}Core}$ are calculated from the same correlation in Equation 8. Further, CHF correlations are applied within their NRC-approved applicability ranges provided in DOM-NAF-2-P-A.

The largest penalty calculated using Equation 8 for each CHF correlation across the applicable subset of statepoints determined the TCP for application against cycle-specific retained DNBR margin. Select TCPs calculated in support of the GAIA fuel transition are outlined below to illustrate the approach described above (Subpart D).

For the GAIA fuel transition, a TCP of 2.4% was derived for application to DNB analysis results calculated using the VIPRE-D/ORFEO-GAIA code/correlation pair. The value is obtained from a representative FSAR Chapter 15 dropped rod case which gives the largest TCP penalty of the statepoints analyzed with ORFEO-GAIA. An MDNBR of 1.682 is obtained from the full-core GAIA model analysis ($MDNBR_{Full\text{-}Core}$) and an MDNBR of 1.642 is obtained from the mixed-core model containing a single GAIA assembly ($MDNBR_{Transition\text{-}Core}$). Since ORFEO-GAIA is only applicable to the mixing vane region of the fuel, the MDNBR location was verified to correspond with a GAIA subchannel above the first mixing vane grid prior to use in Equation 8. The TCP is calculated in Equation 9 as:

$$TCP(\%) = \frac{[(1.642)-(1.682)]}{(1.682)} * 100 = -2.38\% \qquad \text{Eq. 9}$$

A penalty of 2.7% was similarly derived for application when using the VIPRE-D/ORFEO-NMGRID code/correlation pair above the GAIA first mixing vane spacer grid. ORFEO-NMGRID is typically used in (1) the non-mixing grid fuel region and (2) the mixing grid fuel region when core conditions outside of the ORFEO-GAIA applicability range are present. Therefore, the ORFEO-NMGRID CHF correlation was selected for application above the first mixing vane grid for applicable statepoint runs. The penalty is obtained from the same representative FSAR Chapter 15 dropped rod statepoint, as it also yields the largest TCP of all statepoints analyzed with ORFEO-NMGRID in the mixing grid fuel region. An MDNBR of 1.308 is obtained from the full-core GAIA model analysis ($MDNBR_{Full\text{-}Core}$) and an MDNBR of 1.273 is obtained from the mixed-core model containing a single GAIA fuel assembly ($MDNBR_{Transition\text{-}Core}$). Prior to use in Equation 8, the MDNBR location was verified to correspond with a GAIA subchannel above the first mixing vane grid. The TCP is calculated in Equation 10 as:

$$TCP(\%) = \frac{[(1.273)-(1.308)]}{(1.308)} * 100 = -2.68\% \qquad \text{Eq. 10}$$

If mixed-core DNBR calculations show improved CHF performance for an applicable code/correlation pair compared to full core DNB calculations, no TCP is applied. The mixed-core DNB analysis results for the VIPRE-D/WRB-2M code/correlation pair provide an example. A positive DNBR differential was obtained for all statepoints analyzed with the WRB-2M CHF correlation when RFA-2 fuel was placed in the PLB. The smallest positive differential corresponds with a representative CTL statepoint. An MDNBR of 1.581 was obtained from the full-core RFA-2 model analysis ($MDNBR_{Full\text{-}Core}$), and an MDNBR of 1.597 was obtained from the mixed-core model containing a single RFA-2 fuel assembly ($MDNBR_{Transition\text{-}Core}$). Prior to use in Equation 8, the MDNBR location was verified to correspond with an RFA-2 subchannel above the first mixing-vane grid, where the WRB-2M CHF correlation is applicable. The positive TCP is calculated in Equation 11 as:

$$TCP(\%) = \frac{[(1.597)-(1.581)]}{(1.581)} * 100 = +1.01\% \qquad \text{Eq. 11}$$

Mixed-core calculations for the resident Westinghouse fuel showed improved DNBR performance for all applicable code/correlation pairs when compared to full-core calculations of the Westinghouse fuel. Improved DNBR performance was also observed for ORFEO-NMGRID below the GAIA first mixing vane spacer grid.

In summary,

- For Subpart A, details are provided on the calculation of $MDNBR_{Transition-Core}$. The VIPRE-D models created and general analysis approach are described. The applicable CHF correlations for each fuel type (ORFEO-GAIA and ORFEO-NMGRID for GAIA fuel) are evaluated in the PLB where MDNBR is calculated. Each correlation is applied within its NRC-approved applicability range provided in DOM-NAF-2-P-A.
- For Subpart B, the MDNBR values from Tables 3.6-1 and 3.6-2 are not used in TCP calculations. Rather, details on the calculation of $MDNBR_{Full-Core}$ using the VIPRE-D/ORFEO-GAIA and VIPRE-D/ORFEO-NMGRID code/correlation pairs have been provided.
- For Subpart C, detailed discussion is provided in the RAI-12 response. Note, $MDNBR_{Full-Core}$ is not a constant value across all evaluated statepoints for a given CHF correlation.
- For Subpart D, $MDNBR_{Full-Core}$ is a variable value. Limiting TCP calculations are summarized to provide example values of $MDNBR_{Transition-Core}$ and $MDNBR_{Full-Core}$ for both Framatome and Westinghouse fuel products.

RAI-12

Section 3.6, "Monte Carlo Calculations," of Attachment 3 to the LAR, states:

> *In order to perform the Monte Carlo analysis, nine Nominal Statepoints covering the full range of normal operation and anticipated transient conditions were selected for both the ORFEO-GAIA and ORFEO-NMGRID CHF correlations. These statepoints must span the range of conditions over which the statistical methodology will be applied. The Nominal Statepoints were selected to cover the DNB limiting range of the Reactor Core Safety Limits (RCSL) and within the validation range of applicability of the associated correlations. In order to apply the methodology to low flow events, a low flow statepoint is also included. The selected Nominal Statepoints are listed in Tables 3.6-1 and 3.6-2.*

In section 3.8, "Verification of Nominal Statepoints," it is stated that:

> *Condition 1 of the USNRC's SER for VEP-NE-2-A (Reference 11) requires that the Nominal Statepoints be shown to provide a bounding DNBR standard deviation for any set of conditions to which the methodology may potentially be applied.*

Provide clarification on the significance of each of the nine statepoints and the reason they are chosen for the Monte Carlo analysis. Also, provide explanation on why the selected statepoints vary from those in Table 2.2.1 of Reference 11 (VEP-NE-2-A, Revision 1, "Statistical DNBR Evaluation Methodology," June 1987 (ML101330527)).

DENC Response to RAI-12

The selection of each statepoint for MPS3 Framatome GAIA fuel application was performed following the NRC-approved methodology in VEP-NE-2-A.

The selected statepoints represent conditions across the range of power, flow, temperature, and pressure for which the DNB statistical methodology is applied. Eight of the nine statepoints represent the DNB-limiting portion (higher power portion) of the TS 2.1.1 Reactor Core Safety Limit (RCSL) lines as they are established to preclude DNB from occurring. VEP-NE-2-A refers to the RCSLs as the Core Thermal Limit (CTL). Two statepoints were selected for each pressure: one statepoint at high power condition and one statepoint at lower power condition (typically based on the intercept of the DNB-limiting portion of the CTL with the vessel exit boiling lines). These statepoints were analyzed at full flow conditions. These represent statepoints A to H in Tables 3.6-1 and 3.6-2 for MPS3 Framatome GAIA fuel application.

The inlet temperatures used in the calculation of the Statistical Design Limit (SDL) differ from the RCSLs presented in the MPS3 COLR, as they are based upon the Framatome GAIA fuel and result in a target DNBR representative of the SDL. This target DNBR is listed in Attachment 3, Table 3.6-1 (for VIPRE-D/ORFEO-GAIA code/correlation pair) and Table 3.6-2 (for VIPRE-D/ORFEO-NMGRID code/correlation pair) of the subject LAR, dated October 30, 2023. The RCSLs presented in the COLR are developed at a DNBR value equal to the Safety Analysis Limit (SAL). DENC will update the COLR to reflect operation with Framatome GAIA fuel under the provisions of 10 CFR 50.59, coincident with the loading of Framatome GAIA fuel.

The ninth statepoint represents a low flow condition typical of the Complete Loss of Flow and/or the Locked Rotor limiting DNB conditions. The inclusion of the ninth statepoint is to allow the methodology to be applied to low flow conditions. The addition of a low flow statepoint was included in accordance with the NRC-approved methodology in VEP-NE-2-A. This is statepoint I in Tables 3.6-1 and 3.6-2 for MPS3 Framatome GAIA fuel application.

The MPS3 GAIA statepoints in Tables 3.6-1 and 3.6-2 vary from those presented in Table 2.2.1 of VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," June 1987 as they represent conditions (pressure, inlet temperature, power, flow, and $F_{\Delta H}^{N}$) for a different plant and different fuel application. The statepoints listed in Table 2.2.1 of VEP-NE-2-A represent statepoints for North Anna Power Station. Attachment 3, Tables 3.6-1 and 3.6-2 of the subject LAR, dated October 30, 2023, include statepoints for MPS3 using Framatome GAIA Fuel.

RAI-13

Section 2.3.1 of Attachment 1 to the LAR, DOM-NAF-2-P-A and VEP-NE-2-A Application to GAIA Fuel in MPS3 Cores, indicates that TS 2.1.1.1 is modified to include the VIPRE-D/ORFEO-GAIA DDL of 1.13.

Provide clarification as to why the VIPRE-D/ORFEO-NMGRID value of 1.21 which was provided in Table 4.2-1, "DNBR Limits for ORFEO-GAIA and ORFEO-NMGRID" of Attachment 3 is not added to TS 2.1.1.1.

DENC Response to RAI-13

The changes to MPS3 TS 2.1.1.1 are limited to the inclusion of the VIPRE-D/ORFEO-GAIA Deterministic Design Limit (DDL) of 1.13. The following considerations establish why inclusion of the VIPRE-D/ORFEO-NMGRID DDL to MPS3 TS 2.1.1.1 is not required.

The TS 2.1.1 listing of the DNB correlation and fuel centerline melt limits were added to the TSs as part of the justification to allow relocation of the RCSL figure to the COLR. WCAP-14483-A (ADAMS Accession No. ML020430092) provides the technical bases supporting Technical Specification Task Force (TSTF) Traveler 339-A, Rev. 2 which relocated the RCSL figure to the COLR. The NRC SE on WCAP-14483-A discussed how the (1) DNB correlation and fuel centerline melt limits, (2) RCSL figure, and (3) Reactor Protection System (RPS) work together to ensure the requirements of 10 CFR 50.36 are met.

OREFO-GAIA is the primary CHF correlation used in DNB analysis of Framatome GAIA fuel. Alternate CHF correlations are used in DNB analysis when conditions outside the applicability range of the primary CHF correlation are present. ORFEO-NMGRID is the alternate CHF correlation used in DNB analysis of the GAIA fuel. DNB calculations that support Framatome GAIA RCSL development remain within the OREFO-GAIA applicability range described in DOM-NAF-2-P-A, Appendix F.

Given the basis of the NRC approval that added the DNB and fuel centerline melt limits to the TS, only the DNB limits supporting RCSL development (OREFO-GAIA for Framatome fuel) are added to the TS. The VIPRE-D/OREFO-GAIA DDL limit of 1.13 is commensurate with the VIPRE-D/WRB-2M CHF correlation limit currently listed in TS 2.1.1.1. WRB-2M establishes the COLR RCSL figure for the resident Westinghouse RFA-2 fuel in cores at MPS3.

RAI-14

Section 3.10, "Summary of Analysis," of Attachment 3 to the LAR, states:

> *This value was then combined Root Sum Square with code and model uncertainty standard deviations to obtain a total DNBR standard deviation, listed in Tables 3.6-3 and 3.6-4. The use of total DNBR standard deviation in Equation 3.1 yields a peak pin DNBR limit of 1.243 for VIPRE-D/ORFEO-GAIA and 1.261 for VIPRE-D/ORFEO-NMGRID with at least 95% probability at a 95% confidence level. The total DNBR standard deviation was then used to obtain 99.9% DNB protection in the full core DNBR limit of 1.251 for VIPRE-D/ORFEO-GAIA and 1.298 for VIPRE-D/ORFEO-NMGRID.*

Therefore, the VIPRE-D/ORFEO-GAIA code/correlation pair SDL for MPS3 Framatome 17x17 GAIA Fuel is set to 1.26 and the VIPRE-D/ORFEO-NMGRID code/correlation pair SDL for MPS3 Framatome 17x17 GAIA Fuel is set to 1.31.

Provide explanation on how the values of 1.26 and 1.31 are obtained for VIPRE-D/ORFEO-GAIA and VIPRE-D/ORFEO-NMGRID code/correlation pair SDL for MPS3 Framatome 17x17 GAIA Fuel.

DENC Response to RAI-14

The methodology described in the NRC-approved VEP-NE-2-A TR determines a peak pin DNBR limit which provides DNB protection with at least 95% probability at a 95% confidence level and a full core DNBR limit such that no more than 0.1% of the rods in the core are expected to be in DNB if the plant were to operate at the statistical limit. The statistical DNBR limit is defined by the most limiting (higher) of the two DNBR limits (peak pin DNBR limit and full core DNBR limit). The final statistical DNBR limit, which has been submitted for NRC approval, is then rounded up to two decimal places.

For MPS3 Framatome 17x17 GAIA fuel application, the peak pin DNBR limit is 1.243 and the full core DNBR limit is 1.251 for VIPRE-D/ORFEO-GAIA code/correlation pair, which was rounded up to 1.26. For VIPRE-D/ORFEO-NMGRID code/correlation pair, the peak pin DNBR limit is 1.261 and the full core DNBR limit is 1.298, which was rounded up and increased to 1.31. The increase of 0.01 to the VIPRE-D/ORFEO-NMGRID code/correlation limit provides additional margin to accommodate potential plant changes (i.e., instrumentation changes that cause parameter uncertainties to increase) without invalidating the VIPRE-D/ORFEO-NMGRID code/correlation limit.

RAI-15

In Section 4.2, "Safety Analysis Limits (SAL)" and 4.3, "Retained DNBR Margin," of Attachment 3 to the LAR,

a. *The SAL for ORFEO-GAIA and ORFEO-NMGRID for both Deterministic and Statistical DNB analysis (SAL_{DET} and SAL_{STAT}) is listed as 1.45. How is this value obtained? MPS3 FSAR Table 4.4-4 lists the SAL limit value as 1.45 for Westinghouse 17x17 RFA-2 Fuel. Is the same value being imposed on the Framatome GAIA fuel? And if so, what is the reason for that?*

b. *Provide clarification on why the values for the Deterministic Applications are higher than the ones for Statistical Applications for the retained DNB margin (%) listed in tables 4.3-1 and 4.3-2.*

DENC Response to RAI-15

a. The SAL is a self-imposed, licensee-controlled limit. The SAL is set to a DNBR value that is higher than the design limits and provides margin to the design limits. The SAL for ORFEO-GAIA and ORFEO-NMGRID for both Deterministic and Statistical DNB analysis (SAL_{DET} and SAL_{STAT}) of 1.45 is self-imposed by DENC. It is desirable from a human performance perspective to have the same SAL for all the various correlations used at a given site. Thus, the SAL of 1.45 for Framatome GAIA fuel was chosen consistent with the current SAL for Westinghouse 17x17 RFA-2 fuel. The SAL for deterministic and statistical DNB analyses (SAL_{DET} and SAL_{STAT}, respectively) may be changed without prior NRC review and approval, provided the changes meet the criteria established in NEI Document 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Implementation."

b. The retained DNBR margin is the percent difference between the SAL (self-imposed, licensee-controlled limit) and the design limit (DBLFPB). The deterministic design limit (DDL) is the design limit for deterministic DNBR applications, and the statistical design limit (SDL) is the design limit for statistical DNBR applications. As the SAL for both deterministic and statistical DNBR analysis (SAL_{DET} and SAL_{STAT}) is chosen as 1.45, the retained DNBR margin values are higher for deterministic applications compared to statistical applications, as the DDL is lower than the SDL. The SDL is higher than the DDL as it accounts for the uncertainties of the parameters treated statistically in the Statistical DNBR Evaluation Methodology (VEP-NE-2-A).

Attachment 3

SUPPLEMENTAL INFORMATION RELATED TO LAR ATTACHMENT 1, TABLE 1

Dominion Energy Nuclear Connecticut, Inc.
Millstone Power Station Unit 3

2.3.4. <u>ANP-10338-P-A Application to GAIA Fuel in MPS3 Cores</u>

DENC is requesting USNRC review and approval of the DBLFPBs associated with MPS3-specific application of ANP-10338-P-A, "AREA-ARCADIA Rod Ejection Accident," [Reference 3]. The MPS3-specific AREA analysis for GAIA fuel was performed against the acceptance criteria defined in USNRC Regulatory Guide (RG) 1.236 [Reference 7]. The proposed DBLFPBs are outlined in Table 1.

Framatome performed an MPS3 Rod Ejection Analysis (REA) for GAIA fuel. This included the use of Framatome's COBRA-FLX thermal-hydraulics code with the ORFEO-GAIA CHF correlation as part of the ANP-10338-P-A methodology. Therefore, the USNRC-approved COBRA-FLX/ORFEO-GAIA DDL is provided in Table 1 to correspond with the Framatome codes and methods used in the MPS3 REA for GAIA fuel.

Table 1: MPS3-Specific AREA Acceptance Criteria

RG 1.236 Fuel Cladding Failure Mechanism	**Parameter**	**Limit**
High Temperature	Peak Radial Average Enthalpy (cal/g) vs. Cladding Pressure Differential (MPa)	RG 1.236, Figure 1
High Temperature	DNBR Design Limit	1.12 for COBRA-FLX/ ORFEO-GAIA [Reference 8]
Pellet Clad Mechanical Interaction (PCMI)	Peak Radial Average Fuel Enthalpy Rise (Δcal/g) vs. Excess Cladding Hydrogen (wppm)	RG 1.236, Figures 2 and 4
Molten Fuel and Core Coolability	Fuel Centerline Melt (F)	Less than 4754°F, decreasing linearly by 12.8 ~~13.7~~°F per 10,000 MWD/MTU of burnup; rim melt is precluded
Core Coolability	Peak Radial Average Enthalpy (cal/g)	Less than 230 cal/g

Attachment 4

FRAMATOME APPLICATION FOR WITHHOLDING AND AFFIDAVIT

Dominion Energy Nuclear Connecticut, Inc.
Millstone Power Station Unit 3

AFFIDAVIT

1. My name is Gayle Elliott. I am Director, Licensing and Regulatory Affairs, for Framatome Inc. (Framatome) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in Attachment 1 of a letter from Dominion Energy Nuclear Connecticut, Inc., Millstone Power Station Unit 3, "Response to Request for Additional Information Regarding Proposed Amendment to Support Implementation of Framatome GAIA Fuel," Serial No. 23-251D, Docket No. 50-423, and referred to herein as "Document." Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in this Document is considered proprietary for the reasons set forth in paragraphs 6 (d) and 6(e) above.

7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: September 5, 2024

ELLIOTT Gayle Digitally signed by ELLIOTT Gayle Date: 2024.09.05 10:16:18 -04'00'

Gayle Elliott