From: Richard Guzman
To: Shayan.Sinha@dominionenergy.com
Cc: julie.h.hough@dominionenergy.com; Hipo Gonzalez
Subject: Millstone Power Station, Unit 3 - Request for Additional Information Re: LAR to Support Implementation of Framatome GAIA Fuel (EPID L-2023-LLA-0150)
Date: Wednesday, July 31, 2024 2:49:25 PM

Mr. Sinha,

On July 19, 2024, the U.S. Nuclear Regulatory Commission (NRC) staff sent Dominion Energy Nuclear Connecticut, Inc. (DENC, the licensee) the subject Request for Additional Information (RAI) as a draft message via the NRC Box Enterprise File Sync and Share (EFSS) application. The RAI supports the staff's review of the licensee's October 30, 2023 (ADAMS Accession No. ML23304A047), license amendment request (LAR) to revise the technical specifications (TSs) for Millstone Power Station, Unit No. 3 (MPS3). The LAR proposes to revise the Millstone 3 TSs 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 (TS 2.1.1.2), reducing the reactor trip system instrumentation trip setpoint for the Permissive-8 Interlock (TS Table 2.2-1), and adding to the list of approved methodologies for the Core Operating Limits Report (TS 6.9.1.6.b).

As discussed with you on July 26, 2024, the licensee confirms there are no sensitive/proprietary markings that are needed on the draft RAI and that a conference call to discuss clarifications on the questions were not necessary. Shown below is the official RAI. Please provide a response to these questions no later than September 16, 2024, which is approximately 45 days from the date of this communication.

This communication will be placed in ADAMS as a publicly available, official agency record. Please contact me if you have any questions concerning this request.

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REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING PROPOSED LICENSE AMENDMENT TO SUPPORT IMPLEMENTATION OF FRAMATOME GAIA FUEL MILLSTONE POWER STATION, UNIT NO. 3 DOCKET NO. 50-423

By letter dated October 30, 2023 (ADAMS Accession No. ML23304A047), Dominion Energy Nuclear Connecticut, Inc. (DENC, the licensee) submitted a license amendment request (LAR) to revise the technical specifications (TSs) for Millstone Power Station, Unit No. 3 (Millstone 3). The LAR proposes to revise the Millstone 3 TSs 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 (TS 2.1.1.2), reducing the reactor trip system instrumentation trip setpoint for the Permissive-8 Interlock (TS Table 2.2-1), and adding to the list of approved methodologies for the Core Operating Limits Report (TS 6.9.1.6.b). The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the information provided in the submittal and has determined that additional information as requested below is needed to complete its review.

Regulatory Basis

The regulatory requirements and guidance on which the NRC staff based its acceptance are:

- General Design Criterion (GDC) 10, Reactor Design, in 10 CFR 50 Appendix A requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs). SAFDLs are established to ensure fuel is not damaged (i.e., fuel rods do not fail, fuel system dimensions remain within operational tolerances, and functional capabilities are not reduced below those assumed in the safety analysis). Compliance with GDC 10 provides assurance that the integrity of the fuel and cladding will be maintained, thus preventing the potential for release of fission products during normal operation or AOOs.

- GDC 28, Reactivity Limits, requires reactivity control systems be designed with appropriate limits to assure that the effects of the postulated reactivity accidents, including a rod ejection, can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. Compliance with GDC 28 provides assurance that the integrity of the reactor coolant pressure boundary and core coolability will be maintained.

- Regulatory Guide (RG) 1.236, “Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accident”, provides guidance for the methods and procedures that the NRC considers acceptable for use in the PWR rod ejection and BWR rod drop analysis. It describes analytical limits for analyzing the short-term reactivity insertion and demonstrating compliance with GDC 28. It also defines fuel cladding failure thresholds to support radiological consequence assessments.

- Chapters 4 and 15 of the NUREG-0800, Standard Review Plan (SRP) are relevant to the LAR. Specifically, SRP Section 4.2 describes fuel damage criteria. Section 4.3 describes reactivity criteria with respect to the analysis of reactivity events. Section 4.4 describes the thermal-hydraulic criteria for the core and reactor coolant system. Section 15.4.8 discusses the postulated control rod ejection accident and associated criteria for evaluation of provided the review guidance for the PWR rod ejection accident and associated criteria for evaluation reactor coolant pressure boundary damage and cooling flow impairment.

Attachments 1 and 3 to the LAR provide the thermal-hydraulic basis for the proposed GAIA fuel SAFDLs, setpoint changes, and generic penalties. The information addresses the departure from nucleate boiling ratio (DNBR) acceptance criteria defined in SRP Section 4.4, which includes GDC 10 requirements. The guidance in SRP Sections 4.2, 4.3, 15.4.8 and requirements GDC 28 are discussed since they are related to the rod ejection analysis for the GAIA fuel implementation. The licensee proposed to apply the Regulatory Guide (RG) 1.236 guidance to demonstrate the relevant requirements of GDC 28 and Sections 4.2, 4.3, and 15.4.8 are met.

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.

RAI-2 (AQ A.2)

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.

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.

<u>RAI-4 (AQ B.3)</u>
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.

<u>RAI-5 (AQ B.4)</u>
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.

<u>RAI-6 (AQ C.1)</u>
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.

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.

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.

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.

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.

RAI-11

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/ORFEONMGRID 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.

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)).

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.

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.

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.