2807 West County Road 75
Monticello, MN 55089

**ENCLOSURE 3 CONTAINS PROPRIETARY INFORMATION
WITHHOLD IN ACCORDANCE WITH 10 CFR 2.390**

June 6, 2022

L-MT-22-024
10 CFR 50.90

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Monticello Nuclear Generating Plant
Docket No. 50-263
Renewed Facility Operating License
No. DPR-22

Response to a Request for Additional Information for the Monticello Nuclear Generating Plant Related to the Amendment to Adopt Advanced Framatome Methodologies (EPID: L-2021-LLA-0144)

References:

1. NSPM letter to NRC, "License Amendment Request: Application to Adopt Advanced Framatome Methodologies," (L-MT-21-044) dated July 29, 2021 (ADAMS Accession No. ML21211A594)
2. NRC letter to NSPM, "Monticello Nuclear Generating Plant – Request for Additional Information Related to the Amendment to Adopt Advanced Framatome Methodologies (EPID: L-2021-LLA-0144)," dated May 6, 2022 (ADAMS Accession No. ML22109A005)

Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter "NSPM"), submitted a license amendment request (LAR) on July 29, 2021, (Reference 1) pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," for the Monticello Nuclear Generating Plant (MNGP). The proposed LAR revises MNGP Technical Specification 5.6.3, "Core Operating Limits Report (COLR)," to allow the application of advanced Framatome, Inc., methodologies for determining the core operating limits in support of the loading of the Framatome, Inc. ATRIUM 11 fuel type into the MNGP core and to incorporate a new long-term reactor stability solution.

On May 6, 2022, the U.S. Nuclear Regulatory Commission (NRC) provided a request for additional information (RAI) (Reference 2). The responses to this RAI are provided in non-proprietary and proprietary versions of a Framatome, Inc. report provided in the following enclosures.

Enclosure 1 provides the non-proprietary version of a Framatome, Inc. report entitled, "Monticello Advanced Methods LAR – Response to Request for Additional Information"

(ANP-4005NP, Revision 0). Within this report the proprietary information has been denoted by brackets.

Enclosure 2 provides the associated affidavit executed by Framatome, Inc., the owner of the proprietary information, that sets forth the basis under which the information within this report may be withheld from public disclosure by the NRC. The affidavit addresses with specificity the considerations listed in 10 CFR 2.390, "Public inspections, exemptions, requests for withholding."

Enclosure 3 provides the proprietary version of this Framatome, Inc. report (i.e., ANP-4005P, Revision 0).

The information provided in this letter does not alter the evaluations performed for Reference 1 in accordance with 10 CFR 50.92, "Issuance of amendment."

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," NSPM is notifying the State of Minnesota of this RAI response by transmitting a copy of this letter and the non-proprietary enclosures to the designated State official.

Should you have any questions or if additional information is needed, please contact Mr. Richard Loeffler at (612) 342-8981 or Rick.A.Loeffler@xcelenergy.com.

Summary of Commitments

This letter makes no new commitments and no revisions to existing commitments.

I declare, under penalty of perjury, that the foregoing is true and correct. Executed on June 6, 2022.

Christopher P. Domingos
Site Vice President, Monticello and Prairie Island Nuclear Generating Plants
Northern States Power Company – Minnesota

Enclosures

cc: Administrator, Region III, US NRC
Project Manager, Monticello, US NRC
Resident Inspector, Monticello, US NRC
State of Minnesota (without proprietary enclosure)

ENCLOSURE 1

RESPONSE TO THE REQUEST FOR ADDITIONAL INFORMATION

FOR THE MONTICELLO NUCLEAR GENERATING PLANT

RELATED TO THE AMENDMENT TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

ANP-4005NP, REVISION 0

MONTICELLO ADVANCED METHODS LAR – RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

(27 pages follow)

Monticello Advanced Methods LAR – Response to Request for Additional Information

ANP-4005NP
Revision 0

May 2022

Nature of Changes

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

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1.0 INTRODUCTION

By letter dated July 29, 2021, the Northern States Power Company, a Minnesota Corporation doing business as Xcel Energy, submitted a license amendment request (LAR) to the U. S. Nuclear Regulatory Commission (NRC) for the Monticello Nuclear Generating Plant (MNGP). The amendment would revise the MNGP Technical Specification 5.6.3, Core Operating Limits Report (COLR), to allow the application of advanced Framatome Inc., methodologies for determining the core operating limits in support of the loading of the Framatome, Inc. ATRIUM 11 fuel type into the MNGP core and to incorporate a new long-term reactor stability solution. Upon review of the submittal, the NRC staff provided requests for additional information (RAI) in a letter dated May 6, 2022 (Reference 6). This report provides responses to these RAIs.

The proprietary information in this document is bold-faced and marked with double brackets such as **[[ ]]**.

2.0 SNSB REGULATORY BASES AND RAIs

SNSB RAI 1:

REGULATORY BASIS:

Proposed General Design Criteria (GDC) were published by the Atomic Energy Commission (AEC) in the *Federal Register* (32 FR 10213) on July 11, 1967. An evaluation comparing the MNGP design basis to the AEC-proposed GDCs of 1967 is presented in the MNGP Updated Safety Analysis Report (USAR), Appendix E, “Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria.” AEC-proposed GDC CRITERION 6 - REACTOR CORE DESIGN (CATEGORY A) states:

> The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all offsite power.

RAI:

In ANP-3925P, “Monticello ATRIUM 11 Transient Demonstration,” (Reference 18 – provided in proprietary Attachment 11c of the LAR) Table 3.1, “Disposition of Events Summary for Introduction of ATRIUM 11 Fuel at Monticello,” lists two events which state the events are expected to be non-limiting. The events are USAR 14A, Section 5.0, “Pneumatic System Degradation (Turbine Trip with Bypass and degraded scram speed)” and USAR 14A, Section 5.0, “Loss of Stator Cooling.” If the events do not prove to be non-limiting, explain the process to ensure protection for each reload.

Response:

The objective of the disposition of events is to identify the limiting events which need to be analyzed to support plant operation. As discussed in ANP-3925P Section 3.2, a cycle specific calculation plan is developed to identify the analyses to be performed as part of the licensing campaign. The calculation plan is based on the results of the disposition of events. All events that are not dispositioned as “no further analysis required” are addressed in the calculation plan. An event for which the disposition status is “address for the initial reload” with the comment that it is expected to be bound by another event, will be addressed in the calculation plan. If the event has been shown to be non-limiting based on a previous cycle analysis, the calculation plan will identify the licensing campaign in which the analysis was performed and state that no further analysis is needed for the upcoming cycle. If the initial or subsequent analysis does not conclude the event is non-limiting, the calculation plan will identify the event as needing analysis for the upcoming cycle.

It is noted that ANP-3925P is a demonstration of the applicability of the AURORA-B methodology to Monticello for transient events that are typically limiting and does not represent licensing analysis results for any particular cycle.

SNSB RAI 2:

REGULATORY BASIS:
See RAI 1 Regulatory Basis.

RAI:

ANP-3934P, “Monticello LOCA Analysis for ATRIUM 11 Fuel,” (Reference 19 – provided in proprietary Attachment 12c of the LAR) Appendix A, limitation and condition 11 states:

> Plant-specific licensing applications referencing the AURORA-B LOCA [loss-of coolant accident] evaluation model **[[**
>
> **]]**.

The LAR disposition is as follows:

> BWR [boiling water reactor] fuel rods are **[[**
>
> **]]**

[[**]]**.

What is the basis for **[[** **]]**

Response:

The basis for **[[** **]]** is based on Figure 2-1 which is from the approved rupture model in XN-NF-82-07(P)(A) and used by the AURORA-B LOCA method as indicated in ANP-3934P Section 4.1.

[[ ]]

Figure 2-1 S-RELAP5 BWR Burst Strain Model

SNSB RAI 3

REGULATORY BASIS:

See SNSB RAI 1. RAI:

ANP-3934P, Section 7.1 states that **[[**

]], and Section 7.2 states the SLO analyses are performed with a 0.8 multiplier applied to the two-loop MAPLHGR limit resulting in a maximum SLO MAPLHGR limit of **[[ ]]** kW/ft. Explain how the multiplier 0.8 was selected to determine the maximum MAPLHGR for SLO.

Response:

The reduction factor is defined so the SLO PCT is bounded by the TLO PCT.

SNSB RAI 4:

REGULATORY BASIS:

Title 10 of the Code of Federal Regulations (10 CFR) Section 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors." requires that ECCS be designed to provide cooling following a postulated LOCA.

RAI:

ANP-3934P, Table 4.5 states that the low-pressure coolant injection (LPCI) valve stroke time is 35 seconds.

Based on the information in ANP-3934P, Table 6.2, for the single failure case single failure- LPCI (SF-LPCI), the NRC staff understands that one out of the two LPCI loops flow is used in the analysis and the second LPCI loop is assumed to be a single failure.

During normal plant operation, in the scenario in which the residual heat removal (RHR) system is placed in the suppression pool cooling mode, or flow test mode, the motor-operated valve MO-2009 in the suppression pool return line would be open.

(a) What is the automatic closing time of the suppression pool return valve MO-2009 on receiving a LOCA (loss-of coolant accident) signal?

Response:

When the RHR system is placed into the suppression pool cooling mode or flow test mode, motor-operated value MO-2009 in the suppression pool return line is open. Upon receiving a LOCA signal, the valve will close. However, due to the bounding nature of the SF-LPCI LOCA analysis assumptions, its closing time is not a required input parameter; therefore, no specific closing time is defined. See the response to items (b), (c) and (d) below. (provided by Xcel Energy)

During normal plant operation, when the RHR system is placed in suppression pool cooling mode in limiting condition for operation (LCO) 3.6.2.1, when its average temperature is ≥ 90 ºF, the single failure assumption is relaxed. For the following two scenarios there is no associated LCO in the TS:

- RHR system is in placed in the LPCI flow test mode while the Surveillant Requirement
  3.5.1.7 is in progress.
- RHR is placed in the suppression pool cooling mode for any reason while its average temperature is already less than 90 ºF.

In the above scenarios, the return valve MO-2009 should automatically close on receiving a LOCA signal in the closing time in response to question (a) and the LPCI injection valve must fully open in 35 seconds from receiving LOCA signal. If the closing time of the return valve MO-2009 is greater than 35 seconds, some of the LPCI flow will be bypassed to the suppression pool and the reactor may not receive the fully rated LPCI flow credited in the analysis.

(b) Did the SF-LPCI LOCA analysis considered the above two scenarios in which the return valve MO-2009 is partially open for a short period of time (the difference after the time the injection valve fully opens and the time the return valve fully closes) and the reactor received less than the rated LPCI flow?

Response:

The two scenarios were not considered because they are conservatively bounded by not crediting any LPCI flow. Table 5.1 in ANP-3934P shows LPCI flow is not credited in either loop for the limiting failure of the LPCI injection valve (SF-LPCI).

(c) In case a lesser than the rated flow was credited considering the difference between the closing time of MO-2009 and the opening time of the LPCI injection valve, provide the rated and the degraded values of the LPCI flows.

Response:

Zero LPCI flow was used in both loops. Table 4.5 in ANP-3934P shows the rated LPCI flows.

(d) In case the fully rated LPCI flow was credited from the beginning of the fully open LPCI injection valve, and the bypass flow through the return valve MO-2009 for a short time was not considered, provide justification.

Response:

Rated LPCI flow was not credited. Zero LPCI flow was used in both loops.

SNSB RAI 5:

REGULATORY BASIS:

Regulation 10 CFR 50.46(b)(4) requires that calculated changes in core geometry shall be such that the core remains amenable to cooling.

RAI:

MNGP USAR-03, Revision 36, Section 3.6.3.3, “Performance of Reactor Internals,” fourth paragraph states:

> The designs of the reactor internal components are sensitive primarily to the application of single rather than combination loads. Combination loads tend to produce only a secondary effect. For example, the reactor internal component which is most sensitive to the maximum seismic load is the top guide; however, the top guide is not appreciably affected by DBA [design basis accident] loads. On the other hand, the steam dryer is predominantly affected by the blowdown pressure differentials which accompany the steam line rupture DBA whereas seismic loads are negligible in comparison. Therefore, the following discussion of stresses and deformations calculated for the reactor internals are only concerned with the specific locations in the reactor where the maximum stress or strain occurs for a given load combination. The stress or strain at every other location is less than the one cited.

Based on the above statement, the combination of DBA and seismic loads on the reactor internal structures is considered in the current licensing basis for the design of reactor internal components. The information provided in the LAR does not clearly indicate how this applies to the ATRIUM-11 fuel assemblies when loaded in the MNGP reactor core. Provide a discussion of how the combination of DBA and seismic loads is dispositioned in order to confirm that the coolable geometry of the ATRIUM 11 fuel assembly is maintained as required by the 10 CFR 50.46(b)(4) during a DBA along with a worst seismic event.

Response:

The cited USAR section, Section 3.6.3.3, addresses “Other Reactor Vessel Internals” and applies ASME code requirements for the variety of components listed in USAR Section 3.6.1, which does not include reactor fuel. USAR Section 3.4.1 describes the design basis for fuel mechanical characteristics. This includes the ability to withstand both an operating basis earthquake and a safe shutdown earthquake during normal operation. The ability to withstand a pipe break inside containment (i.e., LOCA) is listed separately and does not include the requirement to combine LOCA loads with seismic loads. Therefore, the combination of seismic + LOCA loads is not required to be analyzed for Monticello.

3.0 SFNB RAI REGULATORY BASES AND RAIs

The Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (SRP, NUREG-0800), Section 4.2, "Fuel System design" (ADAMS Accession No. ML070740002), Section 4.3, "Nuclear Design", and Section 4.4, "Thermal and Hydraulic Design" (ADAMS Accession No. ML070740003), provide regulatory guidance for the review of fuel rod cladding materials, the fuel system, the design of the fuel assemblies and control systems, and thermal and hydraulic design of the core.

According to SRP, Section 4.2, the fuel system safety review provides assurance that:

- The fuel system is not damaged as a result of normal operation and Anticipated Operational Occurrences (AOOs)
- Fuel system damage is never so severe as to prevent control rod insertion when it is required,
- The number of fuel rod failures is not underestimated for postulated accidents, and Coolability is always maintained.

AEC-proposed GDC CRITERION 6 - REACTOR CORE DESIGN (CATEGORY A) – See SNSB RAI 1.

SFNB RAI 1:

Section 3.0 of ANP-3924P/ANP-3924NP states, **[[**

]] Explain how this adjustment to the channel achieves thermal-hydraulic stability.

Response:

The thermal-hydraulic stability of a channel is sensitive to the two-phase to single-phase pressure drop ratio with higher ratios resulting in higher decay ratios. **[[**

]]

[[**]]** and improving the overall stability performance of the ATRIUM 11 bundle.

SFNB RAI 2:

According to Section 5.1 of ANP-3924P, "Applicability of Framatome BWR Methods to Monticello with ATRIUM 11 Fuel," (Reference 5 – provided in proprietary Attachment 3c of the LAR) the **[[** **]]** void quality correlation was developed by Framatome against both the FRIGG void measurements, ATRIUM-10, and ATRIUM 10XM fuel designs. However, it is applied to ATRIUM 11 fuel design which has a different geometric configuration from the ATRIUM-10 and ATRIUM 10XM fuel designs.

(a) Provide justification for how this correlation can be applied to the ATRIUM 11 fuel design.

Response:

The **[[** **]]** void quality correlation was successfully benchmarked by Framatome against both the FRIGG, and ATRIUM-10 /ATRIUM 10XM void measurements which covers a broad range of fuel geometries. The Marviken assembly of FRIGG features 36 heated rods arranged in a cylindrical bundle with a single unheated central rod. The ATRIUM-10 and ATRIUM 10XM bundles are representative of more modern BWR designs with part length rods and modern spacers. The ATRIUM 11 bundle shares the ATRIUM 10XM design feature of a **[[**

]] and justifies the use of the **[[** **]]** correlation with ATRIUM 11 fuel.

(b) Provide justification for the use of the ATRIUM-10 and ATRIUM 10XM uncertainties for use with ATRIUM 11 fuel design.

Response:

As discussed in the response to Part (a), the **[[** **]]**. The standard deviation for the FRIGG tests was shown to be **[[** **]]** while the standard deviation for the ATRIUM-10 and ATRIUM 10XM tests was found to be **[[** **]]** the use of the ATRIUM-10 and ATRIUM 10XM uncertainties is reasonable.

(c) Even though ATRIUM 11 **[[** **]]** void fraction measurements, S-RELAP5 has been used against previous measurements, justify the use of 2-Sigma error of **[[** **]]** for the ATRIUM 11 fuel design.

Response:

Similar to the response in Part (b), the **[[** **]]**. The 2-sigma error for the FRIGG tests was shown to be **[[** **]]** while the 2-sigma error for the ATRIUM-10 tests was found to be **[[** **]]** the use of a 2-sigma error of **[[** **]]** is reasonable.

SFNB RAI 3:

ANP-3882P, "Mechanical Design Report for Monticello ATRIUM 11 Fuel Assemblies," (Reference 7 – provided in proprietary Attachment 4c of the LAR) defines criteria for fuel assembly lift-off as, "The fuel shall not levitate under normal operating or AOO conditions. Under postulated accident conditions, the fuel shall not become disengaged from the fuel support. These criteria assure control blade insertion is not impaired."

(a) Provide a summary of key steps in calculations of assembly lift-off during normal operating conditions for both ATRIUM 11 core and mixed core conditions.

Response:

In summary, the key steps to ensure the fuel does not separate from the fuel support during normal operation for a full core of ATRIUM 11 fuel assemblies or a mixed core are listed below.

1) **[[**

]]

2) The downward forces are summed. The downward forces include the fuel assembly weight, the weight of the fluid inside the fuel channel **[[** **]]** and the downward effect due to a change in momentum of the fluid inlet and outlet flow rate.
3) The upward forces are calculated based on the fuel assembly inlet and bypass pressure differential provided by **[[**

]]

4) **[[**

]]

5) **[[**

]] and mixed

core of the ATRIUM 11 and co-resident fuel or full core of the ATRIUM 11 fuel assembly.

6) Liftoff conditions are confirmed each Framatome fuel assembly reload.

(b) For faulted or accident conditions, such as a LOCA, provide a summary of procedures with a typical calculation describing how the criteria for assembly lift-off is satisfied.

Response:

For faulted or accident conditions, the ATRIUM 11 fuel assembly **[[**

]] The fuel is confirmed to not disengage from the fuel assembly support piece during faulted or accident conditions and is verified each reload of Framatome fuel.

SFNB RAI 4:

Section 3.1 of ANP-3903P, "ATRIUM 11 Fuel Rod Thermal-Mechanical Evaluation for the Monticello LAR," (Reference 9 – provided in proprietary Attachment 6c of the LAR) states, **[[**

]]

(a) What, if any, is the neutronic impact of Cr in the fuel?

Response:

This item has been addressed in the approved topical report ANP-10340P-A (Reference 1).

[[

]]

(b) What is the impact of Cr in fuel on fission gas release, fuel densification and swelling, corrosion and fuel creep?

Response:

The fuel thermal-mechanical processes mentioned in the question, together with all other material properties have been addressed in ANP-10340P-A (Reference 1), as follows:

- [[

]]

[[

]]

SFNB RAI 5:

Section 3.2 of ANP-3903P describes application of RODEX4 and statistical methodology for thermal-mechanical response of the fuel rod surrounded by coolant.

(a) Explain how [[

]]

Response:

The radial depression of the thermal flux is one component of the radial power profile model of RODEX4 and is described in [[

]]

[[

]] The volumetric thermal power at any location in the fuel rod is the product of the value of the radial power profile factor at that radius, the input linear power at the axial location and the volume of **[[**

]]

(b) Explain the term **[[** **]]**

Response:

Neutronic fuel assembly typing is an identification scheme that groups fuel assemblies by the enrichment and gadolinia distribution within the fuel rods that comprise the assembly. Thus, all fuel assemblies within a given type have the same distribution of these two characteristics. For a given type, the mechanical fuel assembly designs are identical (e.g., number of fuel rods; number, location, and length of part-length fuel rods; plenum volumes for each fuel rod; spacer grid design, water channel design, etc.). A reload batch of BWR fuel usually consists of fuel assemblies with identical mechanical designs, but with two or three neutronic types—and occasionally more.

(c) How are **[[**

]] calculated?

Response:

[[

]]

[[

]]

(d) Discuss the methodology used for power measurement and operational uncertainties, manufacturing uncertainties, and model uncertainties. Provide a summary of these uncertainties.

Response:

[[

]]

[[

]]

SFNB RAI 6:

Section 3.3.7 of ANP-3903P states that **[[** **]]** at MNGP. Provide details of how this regulatory limit is implemented at MNGP.

Response:

[[

]]

SFNB RAI 7:

The NRC staff requests the following information related to rod bow:

(a) Section 3.4 of ANP-3893P, "Monticello Thermal-Hydraulic Design Report for ATRIUM 11 Fuel Assemblies," (Reference 8 – provided in proprietary Attachment 5c of the LAR) provides a brief description of how rod bow impacts the thermal margin. Provide details of the analysis to determine the impact of rod bow on thermal margin at lower and higher exposures of ATRIUM 11 fuel at MNGP.

Response:

The Critical Power Ratio (CPR) penalty associated with rod bow is determined **[[**

]]

Framatome's BWR rod bow CPR penalty was derived using open literature data (Reference 5, Attachment 1). Based on the available data, a conservative model to predict the reduction of CPR as a function of rod spacing was developed. This correlation was recast in terms of CPR penalty as a function of % gap closure and is presented in Figure 3-1. The correlation is applicable to ATRIUM fuel types. To confirm the conservatism of the model, Framatome ran a critical power test on an ATRIUM-10 assembly where two rods were bowed to touch each other, (i.e., 100% gap closure). The maximum measured CPR penalty in the test was **[[ ]]** as shown in Figure 3-1. The conservatism in the model was confirmed.

[[

]] This relationship is applicable for the ATRIUM 11 fuel at MNGP. As indicated, the rod bow penalty on CPR does not begin until exposure exceeds **[[ ]]** By the time the ATRIUM 11 assemblies reach this exposure level, they are typically not MCPR limiting, even with the rod bow penalty applied.

[[ ]]

Table 3-1 CPR Penalty as a Function of Assembly Exposure

[[ ]]

Figure 3-1 MCPR Penalty versus Gap Closure with Test Data Comparison

(b) Section 3.3.5 of ANP-3882P, "Mechanical Design Report for Monticello ATRIUM 11 Fuel Assemblies," (Reference 7 – provided in proprietary Attachment 4c of the LAR) states that **[[**

]] Provide details of how this correlation is developed. Also describe how this correlation is used to quantify the creep as a function of fuel exposure.

Response:

The full description is given in BAW-10247P-A, Supplement 2P-A, Revision 0, “Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods,” Framatome Inc., Reference 2 Section 4.1.5.1 and Appendix A (Reference 4 within Section 3.3.5 of ANP-3882P). Framatome uses an empirical model to quantify the creep versus exposure.

SFNB RAI 8:

Section 3.2 of ANP-3893P states that the ATRIUM 11 fuel assemblies are hydraulically compatible with the co-resident ATRIUM 10XM fuel design for the entire range of licensed power-to-flow operating map. Clarify this statement and elaborate whether the hydraulic compatibility between ATRIUM 10XM and ATRIUM 11 fuel designs for all combinations of power-to-flow operating map and extended flow window during transition cycles as well as for full-core ATRIUM 11 at MNGP.

Response:

The Monticello Thermal-Hydraulic Design Report for ATRIUM 11 Fuel Assemblies (ANP-3893P) presents results for two power-to-flow state points (100%P/100%F and 59.2%P/43.3%F) and various core loadings as the plant transitions from a full core of ATRIUM 10XM fuel to a full core of ATRIUM 11 fuel. These results are a subset of all the analyses performed to support the conclusion that the ATRIUM 11 and ATRIUM 10XM fuel are thermal-hydraulically compatible. Similar thermal-hydraulic compatibility analyses were also performed at the following power-to-flow state points: 100%P/105%F, 100%P/80%F, 82.5%P/54.7%F, 25%P/100%F and 25%P/41%F. The results from all the analyses, including the additional power-to-flow state points, show that the thermal-hydraulic design criteria are met. Comparing all the state points analyzed with the power-to-flow operating map shown in Figure 3-2, shows that analyses were performed across the entire power-to-flow operating domain, including the extended flow window.

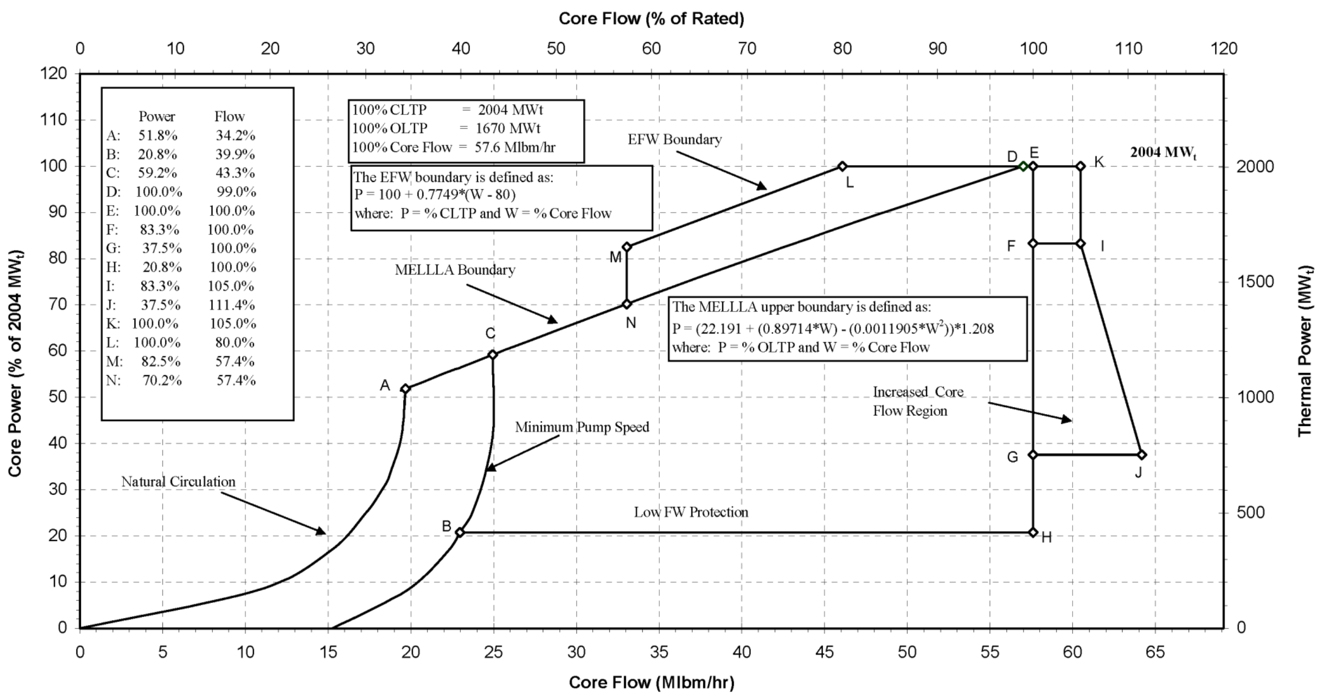

Figure 3-2 Monticello Power / Flow Operating Domain

4.0 REFERENCES

1. ANP-10340P-A, Revision 0, “Incorporation of Chromia-Doped Fuel Properties in AREVA Approved Methods,” Framatome Inc., May 2018.

2. BAW-10247PA, Revision 0, “Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors,” Framatome Inc., February 2008.

3. ANP-3903P, Revision 0, ATRIUM 11 Fuel Rod Thermal-Mechanical Evaluation for Monticello LAR, Framatome Inc., March 2021.

4. BAW-10247P-A Supplement 2P-A Revision 0, “Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods,” Framatome Inc., August 2018.

5. XN-NF-82-06(P)(A) Supplement 1 Revision 2, “Qualification of Exxon Nuclear Fuel for Extended Burnup,” Supplement 1, “Extended Burnup Qualification of ENC 9x9 BWR Fuel,” Advanced Nuclear Fuels Corporation, May 1988.

6. Letter, Robert F. Kuntz (USNRC) to Christopher P. Domingos (MNGP), “Monticello Nuclear Generating Plant – Request for Additional Information Related to the Amendment to Adopt Advanced Framatome Methodologies (EPID L-2021-LLA-0144),” May 6 2022.

7. ANP-10300P-A Revision 1, “AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios,” Framatome Inc., January 2018.

ENCLOSURE 2

RESPONSE TO THE REQUEST FOR ADDITIONAL INFORMATION

FOR THE MONTICELLO NUCLEAR GENERATING PLANT

RELATED TO THE AMENDMENT TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

AFFIDAVIT FOR ANP-4005P, REVISION 0

MONTICELLO ADVANCED METHODS LAR – RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

(3 pages follow)

A F F I D A V I T

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for Framatome Inc. 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 the report ANP-4005P, Revision 0 “Monticello Advanced Methods LAR – Response to Request for Additional Information,” dated May 2022 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 the Document is considered proprietary for the reasons set forth in paragraphs 6(b), 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 have 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: May 27, 2022

MEGINNIS Alan Digitally signed by MEGINNIS Alan
Date: 2022.05.27 16:11:15 -07'00'

Alan B. Meginnis