2807 West County Road 75
Monticello, MN 55089

Attachments 3c, 4c, 5c, 6c, 7c, 8c, 9c, 10c, 11c, 12c, 13c Contain Proprietary Information Withhold in Accordance with 10 CFR 2.390

July 29, 2021

L-MT-21-044
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

License Amendment Request: Application to Adopt Advanced Framatome Methodologies

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," the Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter "NSPM"), requests an amendment to the Technical Specifications (TS) for the Monticello Nuclear Generating Plant (MNGP). NSPM proposes to revise TS 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.

The current long-term reactor stability solution referred to as the Framatome Inc., Enhanced Option III (EO-III) methodology will be upgraded to the Best-estimate Enhanced Option-III (BEO-III) methodology which was recently approved by the U.S. Nuclear Regulatory Commission (NRC). Accordingly, the Extended Flow Window Stability – High function (Function 2.g) contained within TS Specification 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," which is not part of the BEO-III stability methodology, will be removed from the MNGP TS. Also, associated changes to the Technical Requirements Manual will be made to reflect the removal of this function from the MNGP TS.

Enclosed is a description and assessment of the proposed TS changes. Attachment 1 to the enclosure provides the existing TS pages marked up to show the proposed changes. NSPM is not including an attachment with the revised (retyped) TS pages due to the straightforward nature of the proposed TS changes. Providing only mark-ups of the proposed TS changes satisfies the requirements of 10 CFR 50.90 in that the TS mark-ups fully describe the changes desired. Attachment 2 to the enclosure provides the TS Bases pages marked up to show the associated TS Bases changes and is provided for information only.

The enclosure to this letter contains attachments considered proprietary to Framatome Inc. (i.e., Enclosure Attachments 3c, 4c, 5c, 6c, 7c, 8c, 9c, 10c, 11c, 12c, and 13c). Within these attachments, proprietary information has been denoted by brackets. As the owner of the proprietary information, Framatome has executed affidavits for each proprietary document, which identify the information as proprietary, is customarily held in confidence, and should be withheld from public disclosure in accordance with 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." Enclosure Attachments 3a, 4a, 5a, 6a, 7a, 8a, 9a, 10a, 11a, 12a, and 13a provide non-proprietary versions of each proprietary Framatome document. Corresponding affidavits are provided in Enclosure Attachments 3b, 4b, 5b, 6b, 7b, 8b, 9b, 10b, 11b, 12b, and 13b.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), NSPM is notifying the State of Minnesota by providing a copy of this application, with this enclosure and non-proprietary attachments, to the State of Minnesota designated official.

NSPM requests approval of this proposed license amendment within 12 months after completion of the NRC acceptance review. Approval is required to support reactor startup from the spring 2023 Refueling Outage (RFO). Implementation will occur before startup from the spring 2023 RFO.

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

<u>Summary of Commitments</u>

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 July 29, 2021.

for Tom Conboy

Thomas A. Conboy
Site Vice President, Monticello Nuclear Generating Plant
Northern States Power Company – Minnesota

Enclosure

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

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LICENSE AMENDMENT REQUEST

APPLICATION TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

1.0 SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, “Application for amendment of license, construction permit, or early site permit,” the Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter “NSPM”), requests an amendment to the Technical Specifications (TS) for the Monticello Nuclear Generating Plant (MNGP). NSPM proposes to revise 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 ATRIUM™ 11 fuel type into the MNGP core.

Currently, reactor stability protection is provided in accordance with a Framatome enhancement, referred to as Enhanced Option III (EO-III), to the original Boiling Water Reactor Owners Group (BWROG) Option III long-term stability solution described in Reference 1. In March 2021, a further improvement to the original BWROG stability methodology became available with the U.S. Nuclear Regulatory Commission (NRC) approval of the Best-estimate Enhanced Option-III methodology (BEO-III) (Reference 2). With the approval of this license amendment request (LAR), the BEO-III stability methodology will become the long-term reactor stability solution for the MNGP. The existing Extended Flow Window Stability – High scram function (Table 3.3.1.1-1, Function 2.g) contained in TS Specification 3.3.1.1, “Reactor Protection System (RPS) Instrumentation,” is not a part of the BEO-III long-term stability solution methodology and will be removed from the MNGP TS.(1)

2.0 DETAILED DESCRIPTION

2.1 System Design and Operation

Core Operating Limits Report

In accordance with Specification 5.6.3, core operating limits are established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and are documented in the COLR for the listed specifications listed under Item a. The analytical methods used to determine the core operating limits are those previously reviewed and approved by the NRC, specifically those listed under Item b. The core operating limits are determined such that all

1. The associated Technical Requirements Manual Extended Flow Window (EFW) control rod blocks and EFW Stability – High Instrumentation specifications will no longer be applicable and are also being removed from the Technical Requirements Manual as described in Section 3.2 of this LAR.

applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as the shutdown margin, transient analysis limits, and the accident analysis limits) of the safety analysis are met.

Extended Flow Window Stability – High Function

The Extended Flow Window Stability – High function, contained in the TS, uses the Average Power Range Monitor (APRM) simulated thermal power (STP) trip to provide a programmed set of reactor stability protection from reactor core operation initiated from the EFW operating domain defined in the COLR. Current reactor stability protection is derived from the licensing requirements of the Framatome EO-III long-term stability solution.

2.2 Current Technical Specification Requirements

The COLR provides cycle-specific parameter limits for the current reload cycle determined in accordance with the NRC approved analytical methods listed therein, to ensure the safety analysis is met. Some of these currently applicable analytical methods will no longer be applicable, and several Framatome advanced methods will be added with this amendment to allow for the installation of the ATRIUM 11 fuel type.

In the generic EO-III reactor stability methodology, independent channel oscillations (ICOs) that could exceed the Safety Limit Minimum Critical Power Ratio (SLMCPR) cannot be precluded, and hence the solution was to adopt a channel instability exclusion region in the MNGP TS to protect the safety limit. Implementation was accomplished through an extension of the STP – High Function by adding the EFW Stability – High scram function, Specification 3.3.1.1, (Table 3.3.1.1-1, Function 2.g) to the MNGP TS. This TS function is supported by several conditions presented within the Technical Requirements Manual (TRM), discussed in detail in Section 3.2 of this LAR. With the approval of the BEO-III methodology for the MNGP, this TS function will be removed and the associated TRM conditions are no longer necessary and will also be removed from the TRM.

2.3 Reason for the Proposed Technical Specification Changes

NSPM plans to transition to the Framatome ATRIUM 11 fuel type with the 2023 Refueling Outage (RFO). This proposed license amendment reflects the application of advanced Framatome methodologies for the design and analysis of ATRIUM 11 fuel and for determining the core operating limits for this fuel type. NSPM is pursuing the ATRIUM 11 fuel design due to the improved fuel cycle economics and improved safety margins.

The ATRIUM 11 fuel type consists of an 11 by 11 array of fuel rods; whereas, the current fuel design (i.e., ATRIUM 10 XM) consists of an array of 10 by 10 fuel rods. This increase in the number of fuel rods significantly reduces Linear Heat Generation Rate (LHGR) and the fuel duty, thereby improving safety margins.

The ATRIUM 11 fuel type incorporates enhanced debris protection features which make the fuel design less susceptible to debris-related fuel failures. In addition, the channel design changes incorporated with ATRIUM 11 make the fuel design less susceptible to channel bow.

The BEO-III methodology determines the cycle-specific Operating Limit Minimum Critical Power Ratio (OLMCPR) based upon statistical analyses of recirculation pump trip scenarios and evaluation of the time-dependent local power range monitors (LPRMs) response and core MCPR to determine the most limiting event based on the period-based detection algorithm (PBDA) detect and suppress (D&S) hardware response. The BEO-III methodology discussed within the generic topical report (Reference 2) demonstrates that the conservative channel instability exclusion region required by EO-III is not required to protect the SLMCPR applying the BEO-III methodology. Consequently, the Extended Flow Window Stability – High scram function in TS Specification 3.3.1.1 (Table 3.3.1.1-1, Function 2.g) and associated TRM conditions are no longer required and will be removed from the TS and TRM following approval of this LAR.

As described in Section 3.1.11, a plant-specific evaluation applying BEO-III was performed for the MNGP equilibrium ATRIUM 11 core design (Reference 3 – provided in proprietary Attachment 13c). The evaluation demonstrates that the PBDA D&S hardware can reliably detect and suppress oscillations with a high confidence level for the ATRIUM 11 fuel design.

2.4 Description of the Proposed Technical Specification Changes

The following superseded licensing methodologies and an evaluation are proposed to be removed from TS Specification 5.6.3, Item b:

- XN-NF-84-105(P)(A) Volume 1, and Volume 1 Supplements 1 and 2, “XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis,” February 1987
- ANF-913(P)(A) Volume 1 Revision 1, and Volume 1 Supplements 2, 3, and 4, “COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses,” August 1990
- Engineering Evaluation EC 25987, “Calculation Framework for the Extended Flow Window Stability (EFWS) Setpoints,” as docketed in Xcel Energy letter to NRC L-MT-15-065, dated September 29, 2015
- ANP-10262PA, “Enhanced Option III Long Term Stability Solution,” Revision 0, May 2008

These methodologies and an evaluation are no longer applicable with the addition of the advanced methodologies to support the introduction of the ATRIUM 11 fuel type as described in the following sections.

The advanced methodologies listed below are proposed to be added to Specification 5.6.3, Item b:

- 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
- ANP-10340P-A, Revision 0, “Incorporation of Chromia-Doped Fuel Properties in AREVA Approved Methods,” Framatome Inc., May 2018
- ANP-10335P-A, Revision 0, “ACE/ATRIUM 11 Critical Power Correlation,” Framatome Inc., May 2018
- ANP-10333P-A, Revision 0, “AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA),” Framatome Inc., March 2018
- ANP-10300P-A, Revision 1, “AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios,” Framatome Inc., January 2018
- ANP-10332P-A, Revision 0, “AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios,” Framatome Inc., March 2019
- ANP-10344P-A, Revision 0, “Framatome Best-estimate Enhanced Option III Methodology,” Framatome Inc., March 2021

Also, since under the BEO-III stability methodology the Extended Flow Window Stability – High scram function in TS Specification 3.3.1.1 (Table 3.3.1.1-1, Function 2.g) is unnecessary, references to this function will be removed throughout the TS (and TS Bases). Also, the associated EFW TRM conditions will be removed from the TRM (see Section 3.2) with the approval of this amendment.

The following TS changes are proposed:

- Remove reference to Function 2.g in Required Action A.2 and Condition B.
- Remove Action J and relabel following Action K accordingly.
- Remove reference to Function 2.g in Table 3.3.1.1-1.
- Remove statement in Specification 5.6.3, Item a.6, indicating the EFW Stability – High Setpoints are contained in the COLR.

The mark-ups of the proposed TS pages are provided in Attachment 1. Mark-ups of the TS Bases directly related to the proposed change are provided in Attachment 2 for information.

3.0 TECHNICAL EVALUATION

3.1 Discussion of Framatome Boiling Water Reactor (BWR) Methods Being Applied for the MNGP

The new advanced methods listed below are being added to the MNGP licensing basis for this transition to the Framatome advanced methods. They are discussed within the reports referred to within the various sections of this LAR for the plant-specific applications of these advanced methods. The following methods are all approved for Extended Power Uprate/Extended Flow Window (EPU/EFW) applications:

- ACE ATRIUM 11
- RODEX-4 for Chromia doped fuel
- AURORA-B Anticipated Operational Occurrence (AOO)
- AURORA-B Control Rod Drop Accident (CRDA)
- AURORA-B Loss-of Coolant Accident (LOCA)
- Anticipated Transients Without Scram with Instability (ATWS-I) with RAMONA5-FA
- Best-estimate Enhanced Option-III Methodology (BEO-III)

With the approval of this LAR, the currently utilized Enhanced Option III reactor stability methodology will be replaced with BEO-III as the long-term reactor stability solution for the MNGP.

With the transition from ATRIUM 10XM to ATRIUM 11 fuel, the MNGP will transition to RAMONA5-FA as the licensing basis ATWS-I analysis. The current ATWS-I licensing basis analysis was performed with the AISHA/SINANO computer codes as discussed in the NRC safety evaluation for the Extended Flow Window, approved in Amendment No. 191 (Reference 4).

Attachments 3 through 13 provide reports that contain the detailed technical evaluations for the proposed changes to adopt Framatome advanced methods and transition to ATRIUM 11 fuel. The following sections provide a brief summary of what is included in the reports provided within the attachments. The attachments with the ‘a’ designation provide a non-proprietary version of the full report (i.e., proprietary information is redacted). The attachments with the ‘b’ designation provide the affidavit for the proprietary version of the report. The attachments with the ‘c’ designation provide the proprietary version of the full report. For ease of reference throughout this LAR, only the attachments with the ‘c’ designation are referenced within the discussions.

3.1.1 Applicability of Framatome BWR Methods to the MNGP (ANP-3924P)

Plant specific report, ANP-3924P, “Applicability of Framatome BWR Methods to Monticello with ATRIUM 11 Fuel,” (Reference 5 – provided in proprietary Attachment 3c) reviews the Framatome approved licensing methodologies to demonstrate their applicability to the licensing and operation of the MNGP with ATRIUM 11 fuel in the EPU/EFW operating domain with a representative power/flow operating map.

ANP-2637P, “Boiling Water Reactor Licensing Methodology Compendium,” (Reference 6) is a compendium of the Framatome methodologies and design criteria, which are described in various Topical Reports (TRs) that the NRC has found acceptable for referencing in Boiling Water Reactor (BWR) licensing applications. This report is periodically updated by Framatome and provided directly to the NRC and is not included herein.

This compendium provides a concise, organized source for NRC-approved BWR topical reports. It presents information about the application of each topical report, the associated safety evaluation report (SER) and its conclusions and restrictions/limitations for each topical report, the relationships among the topical reports, and, for certain methodologies, descriptions of their unique characteristics or applications. Compliance with the SER restrictions/limitations is assured by implementing them within the engineering analysis guidelines or by incorporating them into the computer codes.

The Framatome licensing methodologies presented within the BWR licensing methodology compendium are applicable to the ATRIUM 11 fuel type and for the operation of the MNGP in the EFW operating domain as indicated within the plant-specific report.

3.1.2 Mechanical Design of MNGP ATRIUM 11 Fuel Assemblies (ANP-3882P)

Licensing report ANP-3882P, “Mechanical Design Report for Monticello ATRIUM 11 Fuel Assemblies,” (Reference 7 – provided in proprietary Attachment 4c) documents completion of all licensing analyses and related testing necessary to verify that the mechanical design criteria are met for the ATRIUM 11 fuel assemblies supplied by Framatome for insertion into the MNGP reactor. This report also provides a description of the mechanical design and licensing methods applied for the ATRIUM 11 fuel. The scope of this report is limited to an evaluation of the mechanical design of the fuel assembly and fuel channel.

The fuel assembly structural design evaluation is not cycle-specific so this licensing report is intended to be referenced for each cycle where the fuel design is in use. Minor changes to the fuel design and cycle-specific input parameters will be dispositioned for future reloads. Framatome confirms the continued applicability of this licensing report prior to delivery of each subsequent reload of ATRIUM 11 fuel at the MNGP in a cycle-specific compliance document.

The fuel assembly design was evaluated according to the Framatome BWR generic mechanical design criteria. The fuel channel design was evaluated to the criteria given in the fuel channel topical reports. The generic design criteria have been approved by the NRC and the criteria are applicable to the subject fuel assembly and fuel channel design. Mechanical analyses have been performed using NRC-approved design analysis methodology.

3.1.3 T-H Design of MNGP ATRIUM 11 Fuel Assemblies (ANP-3893P)

Report ANP-3893P, "Monticello Thermal-Hydraulic Design Report for ATRIUM 11 Fuel Assemblies," (Reference 8 – provided in proprietary Attachment 5c) presents the results of the MNGP thermal-hydraulic (T-H) analyses to demonstrate that the Framatome ATRIUM 11 fuel is hydraulically compatible with the previously loaded ATRIUM 10 XM fuel design. This report also provides the hydraulic characterization of the ATRIUM 11 and the co-resident ATRIUM 10 XM design for the MNGP.

The generic T-H design criteria applicable to the design have been reviewed and approved by the NRC in topical report ANF-89-98(P)(A) Revision 1 and Supplement 1. In addition, the thermal-hydraulic criteria applicable to the design have also been reviewed and approved by the NRC in topical report XN-NF-80-19(P)(A) Volume 4, Revision 1.

ATRIUM 11 fuel assemblies have been determined to be hydraulically compatible with the co-resident ATRIUM 10 XM fuel design in the MNGP reactor for the entire range of the licensed power-to-flow operating map.

3.1.4 ATRIUM 11 Fuel Rod T-M Evaluation for the MNGP (ANP-3903P)

Licensing report ANP-3903P, "ATRIUM 11 Fuel Rod Thermal-Mechanical Evaluation for the Monticello LAR," (Reference 9 – provided in proprietary Attachment 6c) documents the results of thermal-mechanical (T-M) analyses for the performance of the ATRIUM 11 fuel assemblies inserted into to an equilibrium cycle for the MNGP unit and demonstrates that the design criteria relevant to T-M performance are satisfied. These analyses include the use of a chromia additive in the urania portions of the fuel and operation in the EFW operating domain. Both the design criteria and the analysis methodology have been approved by the NRC.

The analysis results are evaluated according to the generic fuel rod thermal and mechanical design criteria contained in ANF-89-98(P)(A) Revision 1 and Supplement 1 along with design criteria provided in the RODEX4 fuel rod thermal-mechanical topical report, as modified for the MNGP ATRIUM 10 XM fuel transition.

The approved methodology for the inclusion of chromia additive in the fuel pellets discussed in ANP-10340P-A, "Incorporation of Chromia-Doped Fuel Properties in AREVA Approved Methods," (Reference 10) was also applied.

3.1.5 MNGP ATRIUM 11 Equilibrium Fuel Nuclear Fuel Design (ANP-3877P)

Design report ANP-3877P, “Monticello ATRIUM 11 Equilibrium Fuel, Nuclear Fuel Design Report,” (Reference 11 – provided in proprietary Attachment 7c) provides the results from the neutronic design analyses performed by Framatome for the MNGP ATRIUM 11 equilibrium design. The methodology, design criteria, and general assumptions used in the fuel design are also provided.

Applicable neutronic design criteria are provided in the approved topical report ANF-89-98(P)(A) Revision 1 and Supplement 1. Neutronic design analysis methodology used to determine conformance to design criteria has been reviewed and approved by the NRC in the topical report EMF-2158(P)(A).

The fuel design general assumptions include the use of advanced fuel channels made of the materials as described within this report. The neutronic component of this fuel design includes axially-varying enrichment and Gadolinia and natural uranium dioxide blankets at the top and bottom of the assembly.

3.1.6 MNGP ATRIUM 11 Equilibrium Cycle Fuel Cycle Design (ANP-3881P)

Design report ANP-3881P, “Monticello ATRIUM 11 Equilibrium Cycle, Fuel Cycle Design Report,” (Reference 12 – provided in proprietary Attachment 8c) documents the equilibrium cycle design and the results from a representative Cycle N for the MNGP. This design analysis uses the ATRIUM 11 fuel design and has been performed with the NRC-approved Framatome neutronics methodology.

The CASMO-4 lattice depletion computer code was used to generate nuclear data including cross-sections and local power peaking factors. The MICROBURN-B2 Version 2 three-dimensional core simulator computer code, combined with the application of the applicable critical power correlation, was used to model the core. The specific MICROBURN-B2 Version 2 modeling features utilized are described within the report.

3.1.7 ATWS-I Evaluation with ATRIUM 11 Fuel for MNGP (ANP-3933P)

The Framatome generic methodology for the evaluation of Anticipated Transients Without Scram with Instability (ATWS-I) is presented in (Reference 13). The plant-specific report discussed in this section ANP-3933P, “Monticello ATWS-I Evaluation for ATRIUM 11 Fuel,” (Reference 14 – proprietary Attachment 9c), applies this methodology and documents a RAMONA5-FA based method for evaluating the fuel-specific portion of the event for the MNGP. This method is intended to cover the initial ATWS-I event through the time that operator actions suppress core oscillations. The scope of this analysis covers the MNGP EPU/EFW operating domain using an equilibrium ATRIUM 11 core.

The NRC SE for the Framatome generic ATWS-I evaluation methodology report lists seven limitations and conditions that applications of the methodology must satisfy. Attachment 9c provides a discussion for how each of those limitations and conditions was addressed.

Results of the MNGP ATRIUM 11 ATWS-I analyses for a simulated Turbine Trip With Bypass (TTWB) and the Two Recirculation Pump Trip (2RPT) events demonstrate that the limiting PCTs remain below the 2200°F temperature acceptance criteria.

3.1.8 MNGP ATRIUM 11 Control Rod Drop Analyses (ANP-3929P)

The Framatome generic methodology for the evaluation of BWR CRDA was approved in March 2018 (Reference 15). Plant-specific report ANP-3929P, “Monticello ATRIUM 11 Control Rod Drop Accident Analyses with the AURORA-B CRDA Methodology,” (Reference 16 – provided in proprietary Attachment 10c) summarizes the initial application of the generic AURORA-B CRDA methodology on the MNGP ATRIUM 11 equilibrium cycle.

The Framatome AURORA-B CRDA methodology has been used to evaluate an MNGP ATRIUM 11 equilibrium fuel cycle. The methodology includes the use of a nodal three-dimensional kinetics solution with both T-H and fuel temperature feedback. These models provide more precise localized neutronic and thermal conditions than previous methods to show compliance with regulatory guidance criteria as presented in NRC Regulatory Guide 1.236, “Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents,” (Reference 17).

The NRC SE for the Framatome generic AURORA-B CRDA evaluation methodology report lists limitations and conditions that applications of the methodology must satisfy. Attachment 10c provides a discussion for how each of those limitations and conditions was addressed.

Attachment 3c discusses the CRDA and an adjustment that was made concerning the hydrogen uptake model applied to the MNGP for the CRDA.

3.1.9 MNGP ATRIUM 11 Transient Demonstration (ANP-3925P)

This report, ANP-3925P, “Monticello ATRIUM 11 Transient Demonstration,” (Reference 18 – provided in proprietary Attachment 11c) demonstrates that the AURORA-B AOO methodology is applicable to the MNGP through the results from the analysis of a small subset of plant transients. The transient events chosen to demonstrate the application of the AURORA-B AOO method are typical limiting events for the MNGP as determined from previous cycle analyses and a review of Chapter 14 of the Updated Safety Analysis Report (USAR).

AURORA-B is a comprehensive evaluation model (EM) developed for predicting the dynamic response of BWRs during transient, postulated accidents, and beyond design-basis accident scenarios. The EM contains a multiphysics computer code system with the flexibility to incorporate all the necessary elements for analysis of the full spectrum of BWR events that are postulated to affect the nuclear steam supply system of the BWR plant. The analysis of plant transients presented in Section 4.0 of this report is based upon a representative equilibrium cycle of ATRIUM 11 fuel. A variety of power/flow state points are performed at a cycle exposure and scram speed as discussed in each subsection of Section 4.0.

For a typical reload, a full assessment of the power/flow operating map, cycle exposure, and scram speed is done on a cycle-specific basis for the actual core configuration to develop thermal limits. This report intends to demonstrate the applicability of the AURORA-B AOO methodology to the MNGP for the transient analyses that are typically limiting on a cycle-specific basis. This document is a subset of transient analyses typically performed for each cycle.

The AURORA-B AOO analysis is used to calculate the change in the minimum critical power ratio (ΔMCPR) during the anticipated operational occurrence. The ΔMCPR is combined with the safety limit MCPR to establish or to confirm the plant operating limits for MCPR. The ACE/ATRIUM 11 critical power correlation was used to evaluate the thermal margin of the ATRIUM 11 fuel.

The AURORA-B AOO analysis is also used to calculate the maximum reactor vessel pressure and the maximum dome pressure during the American Society of Mechanical Engineers (ASME) Code and ATWS events. The calculated maximum reactor vessel pressure is compared to the ASME acceptance criterion (110% of the vessel design pressure) and the calculated maximum steam dome pressure is compared to the pressure safety limit within the plant TS.

For the ATWS event, the calculated maximum reactor vessel pressure is compared to ASME Service Level C (120% of design pressure) criteria to demonstrate that the event acceptance criterion is met. Meeting the acceptance criteria confirms that the plant safety valve performance (number of valves available, capacity per valve, and setpoints) is acceptable.

Section 6.3 of Attachment 3c discusses an adjustment made to address Limitation and Condition 12 from the AURORA-B AOO which requires plant-specific approval for any changes made to transient coolant mixing. It provides a description of the method used to evaluate the amount of mixing to be credited during inadvertent startup of a High Pressure Coolant Injection (HPCI) pump. A similar adjustment was made for this transient and approved as indicated in Section 3.5.2.2.3 of the NRC SE (Reference 22) for the Susquehanna Steam Electric Station adoption of advanced methods supporting their adoption of ATRIUM 11 fuel.

Also, Section 6.6 of Attachment 3c discusses an adjustment made modifying the allowable time step sizes for the AURORA-B transient analyses.

3.1.10 MNGP LOCA Analysis for ATRIUM 11 Fuel (ANP-3934P)

This report, ANP-3934P, “Monticello LOCA Analysis for ATRIUM 11 Fuel,” (Reference 19 – provided in proprietary Attachment 12c) demonstrates the AURORA-B Evaluation Model application for the LOCA break spectrum and ECCS analysis for the MNGP. One purpose of the break spectrum analysis is to identify the break characteristics that result in the highest calculated peak cladding temperature (PCT) during a postulated LOCA. The results also provide the maximum average planar linear heat generation rate (MAPLHGR) limit for the ATRIUM 11 fuel as a function of exposure for normal (two-recirculation loop) operation.

The analyses discussed in this report are performed in accordance with the models and computer codes collectively referred to as the NRC approved AURORA-B LOCA Evaluation Model (Reference 20).

The calculations described in this report are performed in conformance with 10 CFR 50 Appendix K requirements and satisfy the event acceptance criteria identified in 10 CFR 50.46.

3.1.11 BEO-III Application to the MNGP (ANP-3932P)

Currently, reactor stability protection at the MNGP is provided in accordance with a Framatome enhancement to the original BWROG Option III long-term stability solution, referred to as Enhanced Option III (EO-III), described in Reference 1. This EFW operating domain long-term stability solution was authorized for application at the MNGP in February 2017, with the NRC approval of Amendment No. 191 (Reference 4). TS and several TRM changes were required to implement the EO-III stability solution for the EFW operating domain in accordance with the amendment. A description of these TS and TRM changes and the basis for their removal from the proposed MNGP licensing basis with the approval of BEO-III for the MNGP are described in the following section.

In March 2021, a further improvement to the BWROG Option III stability methodology became available with the NRC approval of the Best-estimate Enhanced Option-III methodology (BEO-III) topical report (Reference 2). The BEO-III methodology presented in this topical report is a RAMONA5-FA method for determining the operating limit minimum critical power ratio (OLMCPR) with the Option III PBDA. A key improvement over EO-III is the use of cycle-specific best-estimate evaluations of recirculation pump trip scenarios from the initial operating state point through the simulation termination due to limit cycle conditions or reactor trip due to the PBDA. This realistic evaluation of the recirculation pump trip scenarios replaces the compounding of conservatisms inherent in both the original BWROG LTSS Option III topical report and the currently utilized EO-III topical report.

The BEO-III methodology demonstrates that the conservative channel instability exclusion region required by EO-III is not required to protect the SLMCPR with the BEO-III stability methodology.

ANP-3932P, “Application of BEO-III Methodology with Period-Based Detection Algorithm at Monticello,” Reference 3 (proprietary Attachment 13c) applies the BEO-III methodology to the MNGP and establishes the cycle-specific OLMCPR based upon statistical analyses of recirculation pump trip scenarios and evaluation of the time-dependent LPRMs response and core MCPR to determine the most limiting event based on the PBDA D&S hardware response. The MNGP fully follows and applies only the Framatome BEO-III methodology and utilizes the PBDA as the primary stability protection feature (i.e., no hybrid approaches are applied).

As discussed in Reference 2, when the OPRM system is declared inoperable, Backup Stability Protection (BSP) is provided as discussed in Section 7.3 of this NRC approved topical report. The resultant manual BSP exclusion regions on the power/flow map and associated operator guidance will be employed.

Section 5.0 of Attachment 13c discusses compliance of the methodology to the NRC Limits and Conditions associated with the approved BEO-III methodology.

3.2 Removal of the EFW Stability – High Function (EO-III Methodology)

The EO-III methodology employs the same process as the BWROG Option III long-term stability solution for determining the core MCPR response during anticipated oscillations. As described in Section 3.7.2 of the NRC SE for BEO-III (Reference 2), independent channel oscillations (ICOs) are significantly more likely for recirculation pump trips starting from the EFW operating domain, as these oscillations typically only occur deeper into the unstable region (upper left corner) of the power-flow map relative to core-wide oscillations. In the generic EO-III methodology, which is approved for the EFW operating domain, ICOs were precluded by establishing a channel instability exclusion region. This was done because ICOs lead to a breakdown of the relationship between delta-MCPR and oscillation magnitude (i.e., DIVOM [Delta over Initial Versus Oscillation Magnitude]), which forms a central component of that methodology. Thus, it was determined that the methodology could not be guaranteed to protect the SLMCPR in the presence of ICOs. Hence, in conjunction with the normal DIVOM approach, the EO-III methodology required the implementation of a scram region, known as the channel instability exclusion region, to ensure that power would be suppressed before ICOs could develop.

To address this issue, the TS and TRM changes described below were made to implement the channel instability exclusion region for EO-III.

- Addition to TS Specification 3.3.1.1, “Reactor Protection System (RPS) Instrumentation,” Table 3.3.1.1-1, of the Extended Flow Window Stability – High (Function 2.g). OPERABILITY of the Extended Flow Window Stability trip is currently required when in the EFW operating domain above the Maximum Extended Load Line Limit Analysis (MELLLA) line defined in the COLR.

- Addition of a statement to Item a.6 under the COLR specification (TS 5.6.3) indicating that the EFWS – High Setpoints, TS Table 3.3.1.1-1, Function 2.g are specified in the COLR.

- Addition of a reference to engineering evaluation EC 25987 to Item b.4 under the COLR defining the "Calculation Framework for the Extended Flow Window Stability (EFWS) Setpoints."

- Addition to TRM Specification 3.3.2.1, "Control Rod Block Instrumentation," Table 3.3.2.1-1, of the "Extended Flow Window Stability – High" function (Function 3.d). This function provides the corresponding control rod blocks to the RPS scrams required under TS Specification 3.3.1.1 for Function 2.g when the EFWS is enabled.

- Addition of a new TRM specification (i.e., TRM Specification 3.3.6.1, "Extended Flow Window Stability – High Instrumentation") to establish the power level and enabling points on the power/flow map for which the EFW Stability-High Trip must be enabled under different operational conditions. This provides redundant assurance that the EFWS is enabled prior to entering the operating domain at which these instruments are required to ensure adequate stability protection but still provides for maneuverability on the power/flow map for various conditions such as plant startup or shutdown.

However, the BEO-III methodology discussed in Reference 2, demonstrates that the conservative channel instability exclusion region required by EO-III is not required to protect the SLMCPR when applying the BEO-III stability methodology. Therefore, these previously required TS and TRM requirements are now no longer needed and should not be continued as they are not part of the BEO-III methodology. Therefore, NSPM proposes to remove the existing TS and TRM requirements, described above. Removal of these TS and TRM conditions aligns the MNGP licensing basis with the long-term reactor stability licensing basis to be implemented following NRC approval of this LAR for application of the BEO-III methodology at the MNGP.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50.36(a)(1) states that "Each applicant for a license authorizing operation of a production or utilization facility shall include in its application proposed technical specifications in accordance with the requirements of this section." Appropriate TS changes described in this LAR are supported by the use of the proposed methods.

10 CFR 50.36(c)(5) states that the TS will include administrative controls that address the provisions relating to organization and management, procedures, record keeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. The COLR is required as a part of the reporting requirements specified in the MNGP TS Administrative Controls section. The core operating limits must be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and be documented in the COLR. In addition, the COLR specification requires the analytical methods used to determine the core operating limits to be those that have been previously reviewed and approved by the NRC, and specifically to be those described in TS 5.6.3 Item b. The proposed TS changes ensure that these requirements are met.

10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water power reactors," establishes the acceptance criteria for the design basis LOCA. Paragraph (b)(1) requires the calculated maximum fuel element peak cladding temperature (i.e., PCT) to not exceed 2200°F. 10 CFR Part 50, Appendix K, establishes the required and acceptable features of evaluation models for heat removal by the ECCS after the blowdown phase of a LOCA. The use of the proposed analytical methods to determine core operating limits will continue to ensure that fuel performance during normal, transient, and accident conditions complies with these requirements. Specific Average Planar Linear Heat Generation Rate (APLHGR) limits are determined in conformance with 10 CFR 50 Appendix K requirements and documented in the COLR to ensure compliance with 10 CFR 50.46(b)(1).

10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," defines an ATWS as an anticipated operational occurrence followed by the failure of the reactor trip portion of the protection system specified in 10 CFR 50, Appendix A, GDC 20. During an ATWS the potential exists for thermal-hydraulic instability to develop. The analyses presented in this LAR demonstrate ATWS regulatory criteria are satisfied, including those specifically applicable to ATWS-I (i.e., demonstrating core coolability is maintained).

Appendix A to Title 10, Part 50 of the Code of Federal Regulations (CFR), provides various general design criteria (GDCs) providing guidance for the design of nuclear power plants, including GDC 10, “Reactor design,” GDC 12, “Suppression of reactor power oscillations,” GDC 27, “Combined reactivity control system capability,” and GDC 35, “Emergency core cooling. Appendix A to 10 CFR 50 was published in 1971. The applicable MNGP principal design criteria predate these Appendix A criteria and are listed in the MNGP USAR, Section 1.2, “Principal Design Criteria.” In 1967, the Atomic Energy Commission (AEC) issued for public comment a revised set of proposed GDC. An evaluation comparing the MNGP design basis to the AEC-proposed GDCs of 1967 is presented in the MNGP USAR, Appendix E, “Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria.” Use of the methods in this LAR demonstrates that the intent of the GDCs will continue to be met.

4.2 Precedent

Duke Energy Progress, LLC, received amendments for the Brunswick Steam Electric Plant, Units 1 and 2, on March 6, 2020 (Reference 21), and Susquehanna Nuclear, LLC, received amendments for the Susquehanna Steam Electric Station, Units 1 and 2, on January 21, 2021 (Reference 22), to allow the application of advanced Framatome analysis methodologies to support a transition to ATRIUM 11 fuel.

4.3 No Significant Hazards Consideration Analysis

Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter “NSPM”), requests an amendment to the Technical Specifications (TS) for the Monticello Nuclear Generating Plant (MNGP). NSPM proposes to revise 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.

In conjunction, Technical Specification 3.3.1.1, “Reactor Protection System (RPS) Instrumentation,” will be revised to remove the Extended Flow Window (EFW) Stability – High function (Table 3.3.1.1-1, Function 2.g). Inclusion of this function is no longer necessary because the Framatome Best-estimate Enhanced Option-III stability methodology will become the long-term stability solution for the MNGP.

NSPM has evaluated if a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, “Issuance of amendment,” as discussed below:

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The proposed change revises the list of NRC-approved analytical methods used to establish core operating limits and removes the EFW Stability – High function because it is no longer needed with implementation of the Framatome Inc., Best-estimate Enhanced Option-III (BEO-III) stability methodology.

Since no individual precursors of an accident are affected, the proposed changes do not increase the probability of a previously analyzed event.

The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. The proposed changes revise the list of NRC-approved analytical methods used to establish core operating limits. The changes in methodology do not alter the assumptions of the accident analyses. Based on the above, the proposed changes do not increase the consequences of any previously analyzed accidents.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   Creation of the possibility of a new or different kind of accident requires creating one or more new accident precursors. New accident precursors may be created by modifications of plant configuration, including changes in allowable modes of operation. The proposed changes revise the list of NRC-approved analytical methods used to establish core operating limits and remove an un-needed TS instrument function (EFW Stability – High) and revise associated settings due to the adoption of the BEO-III methodology. The proposed changes do not involve any plant hardware modifications or changes to allowable modes of operation thereby ensuring no new accident precursors are created.

   Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

   Response: No

   The proposed changes revise the list of NRC-approved analytical methods used to establish core operating limits. The proposed changes will ensure that the current level of fuel protection is maintained by continuing to ensure that the fuel design safety criteria are met.

   Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, NSPM concludes that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of “no significant hazards consideration” is justified.

4.4 Conclusions

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

5.0 ENVIRONMENTAL CONSIDERATION

The proposed change would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, “Standards for Protection Against Radiation,” or would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22, “Criteria for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review,” specifically paragraph (c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the proposed change.

6.0 REFERENCES

1. ANP-10262PA, Revision 0, “Enhanced Option III Long Term Stability Solution,” Framatome Inc., May 2008

2. ANP-10344P-A, Revision 0, “Framatome Best-estimate Enhanced Option III Methodology,” Framatome Inc., March 2021

3. ANP-3932P, Revision 0, “Application of BEO-III Methodology with Period-Based Detection Algorithm at Monticello,” Framatome Inc., June 2021

4. Amendment No. 191, "Monticello Nuclear Generating Plant - Issuance of Amendment RE: Extended Flow Window (CAC No. MF5002)," February 23, 2017. (ADAMS Accession Nos. ML163428276, ML163428311, and ML17054C394)

5. ANP-3924P, Revision 0, “Applicability of Framatome BWR Methods to Monticello with ATRIUM 11 Fuel,” June 2021

6. ANP-2637P, Revision 9, “Boiling Water Reactor Licensing Methodology Compendium,” April 2021

7. ANP-3882P, Revision 0, “Mechanical Design Report for Monticello ATRIUM 11 Fuel Assemblies, Licensing Report,” March 2021

8. ANP-3893P, Revision 0, “Monticello Thermal-Hydraulic Design Report for ATRIUM 11 Fuel Assemblies,” March 2021

9. ANP-3903P, Revision 0, “ATRIUM 11 Fuel Rod Thermal-Mechanical Evaluation for Monticello LAR, Licensing Report,” March 2021

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

11. ANP-3877P, Revision 0, “Monticello ATRIUM 11 Equilibrium Fuel, Nuclear Fuel Design Report,” October 2020

12. ANP-3881P, Revision 0, “Monticello ATRIUM 11 Equilibrium Cycle, Fuel Cycle Design Report,” November 2020

13. ANP-10346P-A, Revision 0, “ATWS-I Analysis Methodology for BWRs Using RAMONA5-FA,” Framatome Inc., October 2019

14. ANP-3933P, Revision 0, “Monticello ATWS-I Evaluation for ATRIUM 11 Fuel,” June 2021

15. ANP-10333P-A, Revision 0, “AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA),” Framatome Inc., March 2018

16. ANP-3929P, Revision 0, “Monticello ATRIUM 11 Control Rod Drop Accident Analyses with the AURORA-B CRDA Methodology,” June 2021

17. U.S. Nuclear Regulatory Commission Regulatory Guide 1.236, “Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents,” June 2020 (ADAMS Accession No. ML20055F490)

18. ANP-3925P, Revision 0, “Monticello ATRIUM 11 Transient Demonstration Report,” July 2021

19. ANP-3934P, Revision 0, “Monticello LOCA Analysis for ATRIUM 11 Fuel,” July 2021

20. ANP-10332P-A, Revision 0, “AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios,” Framatome Inc., March 2019

21. U.S. NRC to Duke Energy Progress, LLC, “Brunswick Steam Electric Plant, Units 1 and 2 – Issuance of Amendments Nos. 299 and 327 to Revise Technical Specification 5.6.5b to Allow Application of Advanced Framatome ATRIUM 11 Fuel Methodologies (EPID-2018-LLA-0273),” March 6, 2020 (ADAMS Accession No. ML20073F186)

22. U.S. NRC to Susquehanna Nuclear, LLC, “Susquehanna Steam Electric Station, Units 1 and 2 – Issuance of Amendments Nos. 278 and 260 to Allow Application of Advanced Framatome ATRIUM 11 Fuel Methodologies (EPID-2019-LLA-0153),” January 21, 2021 (ADAMS Accession No. ML20164A181)

ENCLOSURE

ATTACHMENT 1

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST
APPLICATION TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

TECHNICAL SPECIFICATION PAGES (MARKED-UP)

(6 pages follow)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.2	--------------NOTE------------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f ~~or 2.g~~. ------------------------------------ Place associated trip system in trip.	12 hours OR In accordance with the Risk Informed Completion Time Program
B. -----------NOTE------------ Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f ~~or 2.g~~. ------------------------------ One or more Functions with one or more required channels inoperable in both trip systems.	B.1	Place channel in one trip system in trip.	6 hours OR In accordance with the Risk Informed Completion Time Program
	OR		
	B.2	Place one trip system in trip.	6 hours OR In accordance with the Risk Informed Completion Time Program
C. One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1 Initiate alternate method to detect and suppress thermal hydraulic instability oscillations. <u>AND</u> I.2 -------------NOTE---------------- LCO 3.0.4 is not applicable ------------------------------------- Restore required channels to OPERABLE.	12 hours 120 days
~~J. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.~~	~~J.1 Reduce THERMAL POWER below the MELLLA boundary defined in the COLR.~~	~~12 hours~~
J. ~~K.~~ Required Action and associated Completion Time of Condition I ~~or J~~ not met.	J.1 ~~K.1~~ Reduce THERMAL POWER to < 20% RTP.	4 hours

Table 3.3.1.1-1 (page 2 of 4)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
c. Neutron Flux – High	1	3(c)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.11(f)(g) SR 3.3.1.1.15	≤ 122% RTP
d. Inop.	1, 2	3(c)	G	SR 3.3.1.1.4 SR 3.3.1.1.15	NA
e. 2-Out-Of-4 Voter	1, 2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.12 SR 3.3.1.1.14 SR 3.3.1.1.15	NA
f. OPRM Upscale	≥ 20% RTP	3(c)	I	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.16	As specified in COLR
~~g. Extended Flow Window Stability – High~~	~~Within EFW boundary defined in COLR~~	~~3(c)~~	~~J~~	~~SR 3.3.1.1.1~~ ~~SR 3.3.1.1.2~~ ~~SR 3.3.1.1.4~~ ~~SR 3.3.1.1.6~~ ~~SR 3.3.1.1.11~~ ~~SR 3.3.1.1.15~~	~~As specified in COLR~~
3. Reactor Vessel Steam Dome Pressure – High	1, 2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.14	≤ 1075 psig

(c) Each APRM / OPRM channel provides inputs to both trip systems.

(f) If the as-found channel setpoint is not the Nominal Trip Setpoint but is conservative with respect to the Allowable Value, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(g) The instrument channel setpoint shall be reset to the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. The NTSP and the methodology used to determine the NTSP are specified in the Technical Requirements Manual.

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

4. Control Rod Block Instrumentation Allowable Value for the Table 3.3.2.1-1 Rod Block Monitor Functions 1.a, 1.b, and 1.c and associated Applicability RTP levels;

5. Reactor Protection System Instrumentation Delta W value for Table 3.3.1.1-1, Function 2.b, APRM Simulated Thermal Power – High, Note b; and

6. The Manual Backup Stability Protection (BSP) Scram Region (Region I), the Manual BSP Controlled Entry Region (Region II), the Reactor Protection System Instrumentation Period Based Detection Algorithm OPRM Upscale trip setpoints associated with Table 3.3.1.1-1 Function 2.f~~, and the EFWS – High setpoints associated with Table 3.3.1.1-1 Function 2.g~~.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel"

2. NEDO-31960-A, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology", with Supplement 1, dated November 1995

3. NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology and Reload Applications," August 1996

4. ~~Engineering Evaluation EC 25987, "Calculation Framework for the Extended Flow Window Stability (EFWS) Setpoints", as docketed in Xcel Energy letter to NRC L-MT-15-065, dated September 29, 2015~~ (Deleted)

5. XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," March 1984

6. EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," February 1998

7. ANF-89-98(P)(A) Revision 1 and Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," May 1995

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

8. XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," March 1983

9. XN-NF-80-19(P)(A) Volume 4 Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," June 1986

10. EMF-2158(P)(A) Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2," October 1999

11. XN-NF-80-19(P)(A) Volume 3 Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description," January 1987

12. ~~XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," February 1987~~

ANP-10333P-A, Revision 0, AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA), Framatome Inc., March 2018

13. ~~ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," August 1990~~

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

14. EMF-2209(P)(A) Revision 3, "SPCB Critical Power Correlation," September 2009

15. EMF-2245(P)(A) Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel," August 2000

16. EMF-2361(P)(A) Revision 0, "EXEM BWR-2000 ECCS Evaluation Model," May 2001

17. EMF-2292(P)(A) Revision 0, "ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients," September 2000

18. EMF-CC-074(P)(A) Volume 4 Revision 0, "BWR Stability Analysis: Assessment of STAIF with Input from MICROBURN-B2," August 2000

19. BAW-10247P-A Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," February 2008

20. ANP-10298P-A Revision 1, "ACE/ATRIUM 10XM Critical Power Correlation," March 2014

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

21. ANP-10307P-A Revision 0, “AREVA MCPR Safety Limit Methodology for Boiling Water Reactors,” June 2011

22. BAW-10255(P)(A) Revision 2, “Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code,” AREVA NP Inc., May 2008

23. ~~ANP-10262PA, Enhanced Option III Long Term Stability Solution, Revision 0, May 2008~~ ANP-10344P-A, Revision 0, "Framatome Best-estimate Enhanced Option III Methodology," Framatome Inc., March 2021

~~24. (Deleted)~~

24. 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

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

26. ANP-10335P-A, Revision 0, "ACE/ATRIUM 11 Critical Power Correlation," Framatome Inc., May 2018

27. ANP-10332P-A, Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios," Framatome Inc., June 2019

The COLR will contain the complete identification for each of the Technical Specification referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

ENCLOSURE

ATTACHMENT 2

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST
APPLICATION TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

TECHNICAL SPECIFICATION BASES PAGES

(MARKED-UP – FOR INFORMATION ONLY)

(11 pages follow)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.b. Intermediate Range Monitor – Inop

This trip signal provides assurance that a minimum number of IRMs are OPERABLE. Anytime an IRM mode switch is moved to any position other than "Operate," the detector voltage drops below a preset level, or when a module is not plugged in, an inoperative trip signal will be received by the RPS unless the IRM is bypassed. Since only one IRM in each trip system may be bypassed, only one IRM in each RPS trip system may be inoperable without resulting in an RPS trip signal.

This Function was not specifically credited in the accident analysis but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

Six channels of Intermediate Range Monitor – Inop with three channels in each trip system are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal.

Since this Function is not assumed in the safety analysis, there is no Allowable Value for this Function.

This Function is required to be OPERABLE when the Intermediate Range Monitor Neutron Flux – High High Function is required.

Average Power Range Monitor

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases. The APRM channels receive input signals from the local power range monitors (LPRMs) within the reactor core to provide an indication of the power distribution and local power changes. The APRM channels average these LPRM signals to provide a continuous indication of average reactor power from a few percent to greater than RTP. Each APRM channel also includes an Oscillation Power Range Monitor (OPRM) Upscale Function which monitors small groups of LPRM signals to detect thermal-hydraulic instabilities.

The APRM System is divided into four APRM channels and four 2-out-of-4 voter channels. Each APRM channel provides inputs to each of the four voter channels. The four voter channels are divided into two groups of two each; with each group of two providing inputs to one RPS trip system. The system is designed to allow one APRM channel, but no voter channels, to be bypassed. A trip from any one un-bypassed APRM will result in a "half-trip" in all four of the voter channels, but no trip inputs to either RPS trip system. Because APRM trip Functions 2.a, 2.b, 2.c, ~~2.f,~~ and 2.f

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

~~and 2.g~~ are implemented in the same hardware, these trip Functions are combined with APRM Inop trip Function 2.d. Any Function 2.a, 2.b, 2.c, or 2.d ~~2.d, or 2.g~~ trip from any two un-bypassed APRM channels will result in a full trip in each of the four voter channels, which in turn results in two trip inputs into each RPS trip system logic channel (A1, A2, B1 and B2). Similarly, any Function 2.d ~~, 2.f, or 2.g~~ or 2.f trip from any two un-bypassed APRM channels will result in a full trip from each of the four voter channels.

Three of the four APRM channels and all four of the voter channels are required to be OPERABLE to ensure that no single failure will preclude a scram on a valid signal. In addition, to provide adequate coverage of the entire core, consistent with the design bases for the APRM Functions 2.a, 2.b, ~~2.c, and 2.g,~~ and 2.c, at least 14 LPRM inputs, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located, must be operable for each APRM channel.

For the OPRM Upscale Function (Function 2.f), LPRMs are assigned to “cells". A minimum of 17 cells, each with a minimum of 2 LPRMs, must be OPERABLE for the OPRM Upscale Function to be OPERABLE (Ref. 25).

2.a. Average Power Range Monitor Neutron Flux – High (Setdown)

For operation at low power (i.e., Mode 2), the Average Power Range Monitor Neutron Flux – High (Setdown) Function is capable of generating a trip signal to prevent fuel damage resulting from abnormal operating transients in this power range. During most operation at low power levels, the Average Power Range Monitor Neutron Flux – High (Setdown) Function will provide a secondary scram to the Intermediate Range Monitor (IRM) Neutron Flux – High Function because of the relative setpoints. When the IRMs are on Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux – High (Setdown) Function will provide the primary trip signal for a core wide increase in power.

No specific safety analyses take credit for the Average Power Range Monitor Neutron Flux – High (Setdown) Function. However, this Function indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed 25% (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER ≤ 25% RTP.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is ≤ 25% RTP.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

There is no Allowable Value for this Function.

This Function is required to be OPERABLE in the MODES where the APRM Functions are required.

2.e. 2-Out-Of-4 Voter

The 2-Out-Of-4 Voter Function provides the interface between the APRM Functions, including the OPRM Upscale Function, and the final RPS trip system logic. As such, it is required to be OPERABLE in the MODES where the APRM Functions are required and is necessary to support the safety analysis applicable to each of those Functions. Therefore, the 2-Out-Of-4 Voter Function needs to be OPERABLE in MODES 1 and 2.

All four voter channels are required to be OPERABLE. Each voter channel includes self-diagnostic functions. If any voter channel detects a critical fault in its own processing, a trip is issued from that voter channel to the associated trip system.

The 2-Out-Of-4 Voter Function votes APRM Functions 2.a, 2.b, ~~2.c, and 2.g~~ and 2.c independently of Function 2.f. This voting is accomplished by the 2-Out-Of-4 Voter hardware in the Two-Out-Of-Four Logic Module. The voter also includes separate outputs to RPS for the two independently voted sets of Functions, each of which is redundant (four total outputs). The voter Function 2.e must be declared inoperable if any of its functionality is inoperable. However, due to the independent voting of APRM trips, and the redundancy of outputs, there may be conditions where the voter Function 2.e is inoperable, but trip capability for one or more of the other APRM Functions through that voter is still maintained. This may be considered when determining the condition of other APRM Functions resulting from partial inoperability of the 2-Out-Of-4 Voter Function 2.e.

There is no Allowable Value for this Function.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Three of the four channels are required to be operable. Each channel is capable of detecting thermal-hydraulic instabilities, by detecting the related neutron flux oscillations, and issuing a trip signal before the MCPR SL is exceeded. There is no allowable value for this function.

19, 20, and 22

The cycle-specific thermal-hydraulic detection algorithms trip settings are nominal settings determined applying the stability analysis licensing methodology (Refs. ~~18, 19 and 20~~) developed by the BWR Owners Group ~~and General Electric~~. There is no Allowable Value for this Function. The settings are not traditional instrumentation setpoints determined under an instrument setpoint methodology. Since the settings may vary cycle-to-cycle, the allowable value column in Table 3.3.1.1-1 indicates the OPRM Upscale Function trip settings, i.e., the period based detection algorithm, are “As specified in the COLR”. In accordance with the NRC Safety Evaluation for Amendment 159 (Ref. 24), the OPRM Upscale Function is not LSSS SL-related.

, General Electric, and Framatome. Utilization of the Framatome Best-estimate Enhanced Option III Methodology was approved for use at the MNGP in Reference 31.

~~2.g. Extended Flow Window Stability – High~~

~~The Extended Flow Window Stability - High (EFWS-High) Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant, representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. The trip level is varied as a function of recirculation drive flow (i.e., at lower core flows, the setpoint is reduced proportional to the reduction in power experienced as core flow is reduced with a fixed control rod pattern).~~

~~The EFWS-High Function is an extension of the Simulated Thermal Power – High Function in that it adds additional line segments to the Simulated Thermal Power – High scram in order to ensure that plant operation cannot occur in the Channel Instability Exclusion Region (CIER) described by Ref. 22. Ensuring that plant operation within the CIER cannot occur preserves the validity of the analysis used to verify that the MCPRSL is not violated during instability events. Additionally, when the primary method of reactor instability detection and suppression is inoperable, the EFWS-High Function acts as a backup by initiating a scram upon operation at the natural circulation line.~~

~~Each APRM channel uses one total drive flow signal representative of total core flow. The total drive flow signal is generated by the flow processing logic, part of the APRM channel, by summing up the flow calculated from two flow transmitter signal inputs, one from each of the~~

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

~~two recirculation loop flows. The flow processing logic OPERABILITY is part of the APRM channel OPERABILITY requirements for this Function.~~

~~The THERMAL POWER time constant of < 7 seconds is based on the fuel heat transfer dynamics and provides a signal proportional to the THERMAL POWER.~~

~~The Extended Flow Window Stability-High Function is required to be OPERABLE when plant operation is above the Maximum Extended Load Line Limit Analysis line due to the possibility of very rapidly growing plant instability events following a significant reduction in core recirculation flow (Ref. 22).~~

~~The cycle specific EFWS-High trip settings are nominal settings determined applying the stability analysis licensing methodology described in Reference 22. The settings are not traditional instrumentation setpoints determined under an instrument setpoint methodology. (Refer to USAR Section 7 for description of the methodology that translates the analytical values describing the CIER to the nominal settings which enforce the CIER). Since the settings may vary cycle to cycle, the allowable value column in Table 3.3.1.1-1 indicates the EFWS-High trip settings are "As specified in the COLR". The EFWS Trip Function is not LSSS SL-related.~~

3. Reactor Vessel Steam Dome Pressure – High

An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes the neutron flux and THERMAL POWER transferred to the reactor coolant to increase, which could challenge the integrity of the fuel cladding and the RCPB. No specific safety analysis takes direct credit for this Function. However, the Reactor Vessel Steam Dome Pressure – High Function initiates a scram for transients that results in a pressure increase, counteracting the pressure increase by rapidly reducing core power. For the overpressurization protection analysis of Reference 9, reactor scram (the analyses conservatively assume scram on the Average Power Range Monitor Neutron Flux – High signal, not the Reactor Vessel Steam Dome Pressure – High signal), along with the S/RVs, limits the peak RPV pressure to less than the ASME Section III Code limits.

High reactor pressure signals are initiated from four pressure switches that sense reactor pressure. The Reactor Vessel Steam Dome Pressure – High Allowable Value is chosen to provide a sufficient margin to the ASME Section III Code limits during the event.

BASES

ACTIONS (continued)

Prior to expiration of the time allotted by the note, the absolute difference between the channel and calculated power is required to be restored to within the limit of SR 3.3.1.1.2 (≤ 2% RTP) or the applicable Condition entered and Required Actions taken. This note is based on the time required to perform APRM adjustments on multiple channels and the impact on safety; additional time is allowed when the APRM is indicating a higher power value than the calculated power, i.e., out of limits but conservative.

A.1 and A.2

Because of the diversity of sensors available to provide trip signals and the redundancy of the RPS design, an allowable out of service time of 12 hours has been shown to be acceptable (Ref. 16) to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided the associated Function's inoperable channel is in one trip system and the Function still maintains RPS trip capability (refer to Required Actions B.1, B.2 and C.1 Bases). Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel or the associated trip system must be placed in the tripped condition per Required Actions A.1 and A.2. Placing the inoperable channel in trip (or the associated trip system in trip) would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternatively, if it is not desired to place the channel (or trip system) in trip (e.g., as in the case where placing the inoperable channel in trip would result in a full scram), Condition D must be entered and its Required Action taken. The 12 hour allowance is not allowed for Reactor Mode Switch – Shutdown Position Function and Manual Scram Function channels since with one channel inoperable RPS trip capability is not maintained. In this case, Condition C must be entered and its Required Actions taken.

As noted, Action A.2 is not applicable for APRM Functions 2.a, 2.b, 2.c, 2.d, ~~2.f, or 2.g.~~ or 2.f. Inoperability of one required APRM channel affects both trip systems. For that condition, Required Action A.1 must be satisfied, and is the only action (other than restoring operability) that will restore capability to accommodate a single failure. Inoperability of more than one required APRM channel of the same trip function results in loss of trip capability and entry into Condition C, as well as entry into Condition A for each channel.

BASES

ACTIONS (continued)

As noted, Condition B is not applicable for APRM Functions 2.a, 2.b, 2.c, 2.d, ~~2.f, or 2.g.~~ or 2.f. Inoperability of an APRM channel affects both trip systems and is not associated with a specific trip system as are the APRM 2-out-of-4 voter and other non-APRM channels for which Condition B applies. For an inoperable APRM channel, Required Action A.1 must be satisfied, and is the only action (other than restoring OPERABILITY) that will restore capability to accommodate a single failure. Inoperability of a Function in more than one required APRM channel results in loss of trip capability for that Function and entry into Condition C, as well as entry into Condition A for each channel. Because Conditions A and C provide Required Actions that are appropriate for the inoperability of APRM Functions 2.a, 2.b, 2.c, 2.d or 2.f, and these functions are not associated with specific trip systems as are the APRM 2-out-of-4 voter and other non-APRM channels, Condition B does not apply.

<u>C.1</u>

Required Action C.1 is intended to ensure that appropriate actions are taken if one or more inoperable, untripped channels within the same trip system for the same Function result in the Function not maintaining RPS trip capability. A Function is considered to be maintaining RPS trip capability when sufficient channels are OPERABLE or in trip (or the associated trip system is in trip), such that both trip systems will generate a trip signal from the given Function on a valid signal. For the typical Function with one-out-of-two taken twice logic and the IRM Functions, this would require both trip systems to have one channel OPERABLE or in trip (or the associated trip system in trip). For Function 5 (Main Steam Isolation Valve – Closure), this would require both trip systems to have each channel associated with the MSIVs in three main steam lines (not necessarily the same main steam lines for both trip systems) OPERABLE or in trip (or the associated trip system in trip). For Function 8 (Turbine Stop Valve – Closure), this would require both trip systems to have three channels, each OPERABLE or in trip (or the associated trip system in trip). For Function 10 (Reactor Mode Switch – Shutdown Position) and Function 11 (Manual Scram), since each trip system only has one channel for each Function, with a channel inoperable, RPS trip capability is not maintained.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

BASES

ACTIONS (continued)

D.1

Required Action D.1 directs entry into the appropriate Condition referenced in Table 3.3.1.1-1. The applicable Condition specified in the Table is Function and MODE or other specified condition dependent and may change as the Required Action of a previous Condition is completed. Each time an inoperable channel has not met any Required Action of Condition A, B, or C and the associated Completion Time has expired, Condition D will be entered for that channel and provides for transfer to the appropriate subsequent Condition.

E.1, F.1, ~~G.1 and J.1~~ and G.1

If the channel(s) is not restored to OPERABLE status or placed in trip (or the associated trip system placed in trip) within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. The allowed Completion Times are reasonable, based on operating experience, to reach the specified condition from full power conditions in an orderly manner and without challenging plant systems. In addition, the Completion Time of Required Action E.1 is ~~Actions E.1 and J.1 are~~ consistent with the Completion Time provided in LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)."

H.1

If the channel(s) is not restored to OPERABLE status or placed in trip (or the associated trip system placed in trip) within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by immediately initiating action to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and are, therefore, not required to be inserted. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies are fully inserted.

BASES

ACTIONS (continued)

~~J.1~~

~~If EFWS High trip capability is not maintained, Condition J exists and action is required to ensure core stability from the prevailing Power/Flow operating condition. This is achieved by reducing THERMAL POWER below the MELLLA boundary, at which point the Option III long term stability protection offered by the OPRM and manual scrams are effective. The Completion Time of 12 hours is based on providing a reasonable time period to effect a power reduction and the low probability of an instability event occurring in that period.~~

J.1 ~~K.1~~

If the required channels are not restored to OPERABLE status and ~~the~~ Required Action I is ~~Actions of I or J are~~ not met within the associated Completion Time ~~Times~~, then the plant must be placed in an operating condition in which the LCO does not apply. To achieve this status, the THERMAL POWER must be reduced to less than 20% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the specified operating power level from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each RPS instrumentation Function are located in the SRs column of Table 3.3.1.1-1.

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the associated Function maintains RPS trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 16) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the RPS will trip when necessary.

SR 3.3.1.1.1

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a

BASES

REFERENCES (continued)

17. NEDC-32410P-A, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function", October 1995.

18. ~~NEDC-33075P-A, Revision 6, "General Electric Boiling Water Reactor Detect and Suppress Solution – Confirmation Density," January 2008.~~ (Deleted)

19. NEDO-31960-A, Supplement 1, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," November 1995.

20. NEDO-32465-A, "BWR Owners' Group Long-Term Stability Detect and Suppress Solutions Licensing Basis Methodology and Reload Applications," August 1996.

21. NEDC-32410P-A, Supplement 1, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function", November 1997.

22. ~~ANP-10262PA, Revision 0, "Enhanced Option III Long Term Stability Solution," May 2008.~~ ANP-10344P-A, Revision 0, "Framatome Best-estimate Enhanced Option III Methodology," Framatome Inc., March 2021

23. U.S. NRC Regulatory Issue Summary 2006-17, "NRC Staff Position on the Requirements of 10 CFR 50.36, "Technical Specifications," Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels," dated August 24, 2006.

24. Amendment No. 159, “Issuance of Amendment Re: Request to Install Power Range Neutron Monitoring System,” dated February 3, 2009. (ADAMS Accession No. ML083440681)

25. GHNE-0000-0073-4167-R2, “Reactor Long-Term Stability Solution Option III: Licensing Basis Hot Channel Oscillation Magnitude for Monticello Nuclear Generating Plant,” December 2007.

26. CA-10-135, “Instrument Setpoint Calculation – Intermediate Range Monitor (IRM) High Flux SCRAM and CR Block” (including Attachment 5, GEH document 0000-0121-5727, IRM Calibration Design Bases).

27. Calculation 09-239, “Turbine Bypass Valve Capacity for EPU.”

28. Amendment No. 171, “Issuance of Amendment Regarding the Restoration Period Before Declaring Average Power Range Monitors Inoperable,” dated January 25, 2013. (ADAMS Accession No. ML12339A035)

BASES

REFERENCES (continued)

29. MNGP EPU Task Report T0506, Revision 4, “Technical Specification Setpoints.”

30. Amendment No. 176, “Monticello Nuclear Generating Plant – Issuance of Amendment 176 to Renewed Facility Operating License Regarding Extended Power Uprate,” December 9, 2013. (ADAMS Accession No. ML13316B298)

31. ~~Amendment No. 191, “Monticello Nuclear Generating Plant Issuance of Amendment RE: Extended Flow Window (CAC No. MF5002),” February 23, 2017. (ADAMS Accession No. ML17054C394)~~

Amendment No. TBD, "Monticello Nuclear Generating Plant – Issuance of Amendment RE: Framatome Advanced Methods Adoption for ATRIUM 11 Fuel Transition (CAC No. XXXXXX)," DATE (ADAMS Accession No. TBD) [Specifics of reference to be added after amendment issuance.]

ENCLOSURE

ATTACHMENT 3a

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST
APPLICATION TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

ANP-3924NP REPORT, REVISION 0

APPLICABILITY OF FRAMATOME BWR METHODS TO MONTICELLO WITH ATRIUM 11 FUEL

JUNE 2021

(51 pages follow)

framatome

Applicability of Framatome BWR Methods to Monticello with ATRIUM 11 Fuel

ANP-3924NP
Revision 0

June 2021

Nature of Changes

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

Contents

List of Tables

List of Figures

Nomenclature

Acronym	Definition
3GFG	Third Generation FUELGUARD
ACE	Framatome's advanced critical power correlation **[ ]**
AFC	Advanced Fuel Channel
AOO	Anticipated Operational Occurrences
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
BOC	beginning of cycle
BWR	boiling water reactor
CDA	confirmation density algorithm
CFD	computational fluid dynamics
CFR	Code of Federal Regulations
CHF	critical heat flux
CPR	critical power ratio
CRDA	control rod drop accident
EFW	extended flow window
EOC	end of cycle
EPU	extended power uprate
FWCF	feedwater controller failure
HPCI	High Pressure Coolant Injection
IHPS	Inadvertant HPCI startup
KATHY	Karlstein thermal hydraulic test facility
LHGR	linear heat generation rate
LOFH	loss of feedwater heating
LOCA	loss of coolant accident
LTP	lower Tie Plate
LTR	licensing topical report
LUC	Lead use channel
LUT	look up table
MAPLHGR	maximum average planar linear heat generation rate

Acronym	Definition
MCPR	minimum critical power ratio
NRC	Nuclear Regulatory Commission, U. S.
OLMCPR	operating limit minimum critical power ratio
PCMI	pellet clad mechanical interaction
PHTF	Portable Hydraulic Test Facility
PLFR	part length fuel rod
RAI	request for additional information
RHR	residual heat removal
RSAR	reload safety analysis report
SLC	standby liquid control
SLMCPR	safety limit minimum critical power ratio
SER	safety evaluation report
UTP	Upper Tie Plate
Z4B	Zircaloy BWR material similar to Zircaloy-4
Zry-4	Zircaloy-4

1.0 INTRODUCTION

This document reviews the Framatome approved licensing methodologies to demonstrate that they are applicable to licensing and operation of the Monticello Nuclear Plant with ATRIUM 11 in the EPU/EFW operating domain with a representative power/flow operating map in Figure 1-1. Application of the new methods added for ATRIUM 11 (ACE ATRIUM 11, RODEX-4 for Chromia doped fuel, AURORA-B AOO, CRDA* and LOCA) for EPU/EFW applications are addressed in this document or in plant specific applications of the new methodologies. Application of Framatome methods to ATWS-I and the application of Framatome BEO-III stability methods are not addressed here and are discussed in separate reports, ANP-3933P and ANP-3932P respectively. These methodologies have all been approved for application to mixed core loadings as discussed in Appendix A including ATRIUM 10XM and ATRIUM 11.

The **[ ]** applied for CRDA startup range evaluation in AURORA-B CRDA and the use of a multiplier on approved Framatome hydrogen uptake model are the only plant specific applications addressed in this report.

For the introduction of ATRIUM 11 at EPU/EFW conditions a review of the RAI's received from previous license applications was used to identify anything that needed to be addressed. Most of the issues identified in previous license applications have been addressed by the NRC approved methodologies that are being used for the licensing of ATRIUM 11 fuel in Monticello.

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* For the Monticello ATRIUM 11 plant-specific application of CRDA, **[ ]** has been applied for the startup range evaluations.

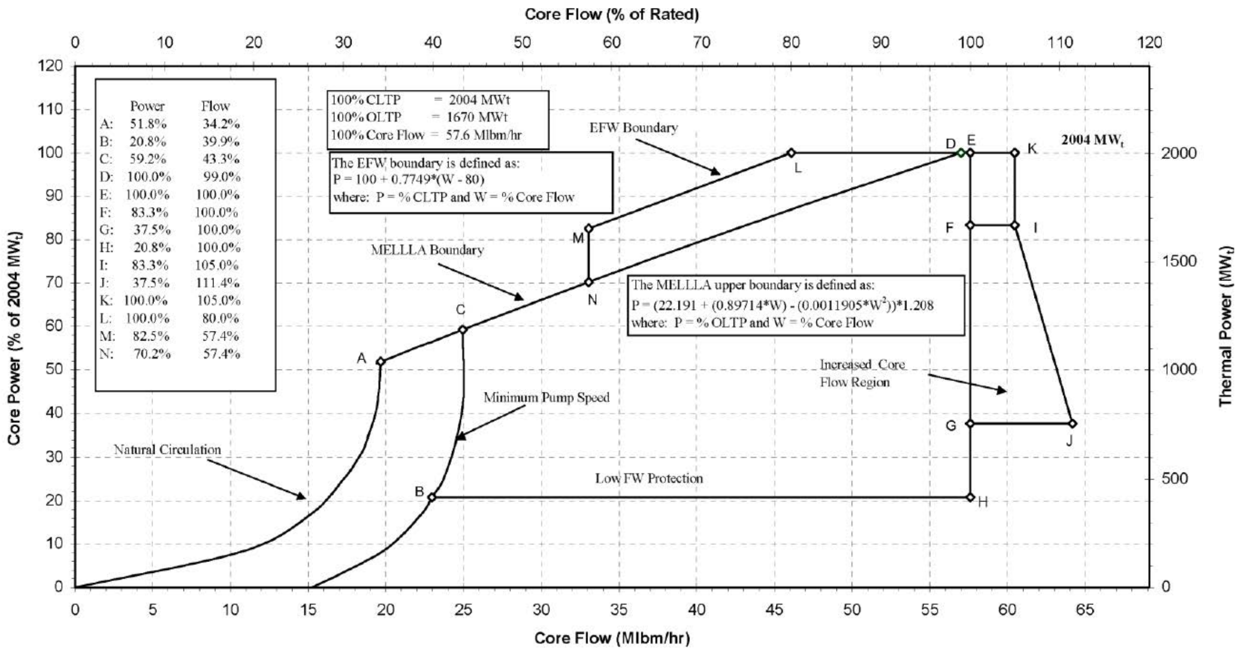

Figure 1-1
Monticello Power Flow Operating Map

2.0 OVERVIEW

The introduction of ATRIUM 11 coincides with the application of a new modern suite of methodologies (References 1 through 10) that also address a number of industry concerns. This is the fourth application of the entire suite of new and upgraded methodologies. Monticello currently operates with ATRIUM 10XM fuel and is transitioning to ATRIUM 11. The design characteristics of the ATRIUM 10XM and ATRIUM 11 are explicitly accounted for in all of the models for operation with the EPU/EFW domain. The differences in fuel design characteristics between the ATRIUM 10XM and ATRIUM 11 are discussed in Section 3.0.

The first step in determining the applicability of current licensing methods to Monticello operating conditions was a review of Framatome BWR topical reports listed in Table 2-1 and the Monticello facility operating license conditions to identify SER restrictions. This review identified that there are no SER restrictions on core power level or core flow for the Framatome topical reports up to and including EPU/EFW. The review also indicated that the **[**

]. This is discussed in the Thermal-Hydraulics section.

Based on the fundamental characteristics of the fuel designs, each of the major analysis domains: thermal-mechanics, thermal-hydraulics, mechanics, core neutronics, transient analysis, LOCA and stability are assessed to determine any challenges to application.

Table 2-1 Framatome Licensing Topical Reports

Document Number	Document Title
XN-NF-79-56(P)(A) Revision 1 and Supplement 1(P)(A)	"Gadolinia Fuel Properties for LWR Fuel Safety Evaluation," Exxon Nuclear Company, November 1981
XN-NF-85-67(P)(A) Revision 1	"Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company, July 1986
XN-NF-85-92(P)(A)	"Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company, November 1986
ANF-89-98(P)(A) Revision 1 and Supplement 1	"Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995
ANF-90-82(P)(A) Revision 1	"Application of ANF Design Methodology for Fuel Assembly Reconstitution," Advanced Nuclear Fuels Corporation, May 1995
EMF-93-177(P)(A) Revision 1	"Mechanical Design for BWR Fuel Channels," Framatome ANP, August 2005
EMF-93-177P-A Revision 1 Supplement 1P-A Revision 0	"Mechanical Design for BWR Fuel Channels Supplement 1: Advanced Methods for New Channel Designs," AREVA Inc., September 2013
EMF-93-177 Revision 1 Supplement 2P-A Revision 1	"Mechanical Design for BWR Fuel Channels: Z4B Material," Framatome Inc., June 2019
BAW-10247PA Revision 0	"Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," AREVA NP, February 2008
BAW-10247PA Revision 0 Supplement 1P-A, Revision 0	"Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 1: Qualification of RODEX4 for Recrystallized Zircaloy-2 Cladding", April 2017
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
ANP-10340P-A Revision 0	"Incorporation of Chromium-Doped Fuel in AREVA Approved Methods", Framatome Inc., May 2018

Table 2-1 Framatome Licensing Topical Reports *(Continued)*

Document Number	Document Title
XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2	"Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," Exxon Nuclear Company, March 1983
XN-NF-80-19(P)(A) Volume 4 Revision 1	"Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, June 1986
EMF-2158(P)(A) Revision 0	"Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/ MICROBURN-B2," Siemens Power Corporation, October 1999
XN-NF-79-59(P)(A)	"Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies," Exxon Nuclear Company, November 1983
XN-NF-80-19(P)(A) Volume 3 Revision 2	"Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description," Exxon Nuclear Company, January 1987
ANP-10298P-A Revision 1	"ACE/ATRIUM 10XM Critical Power Correlation," AREVA, March 2014
ANP-10335P-A Revision 0	“ACE/ATRIUM 11 Critical Power Correlation”, Framatome Inc., May 2018
ANP-10307PA Revision 0	"AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," AREVA NP, June 2011
ANF-1358(P)(A) Revision 3	“The Loss of Feedwater Heating Transient in Boiling Water Reactors,” Framatome ANP, September 2005
ANP-10300P-A Revision 1	“AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios” Framatome Inc., January 2018
ANP-10332P-A Revision 0	“AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios” Framatome Inc., March 2019
ANP-10333P-A Revision 0	“AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)”, Framatome Inc., March 2018

3.0 ATRIUM 11 FUEL ASSEMBLY DESIGN

The ATRIUM 11 fuel assembly design consists of a LTP and UTP, 112 fuel rods, 9 spacer grids, a central water channel, and miscellaneous assembly hardware.

The fuel design utilizes a square internal water channel which occupies nine (3x3) lattice positions. The upper and lower ends of the water channel are attached to connecting hardware which provides a load chain between the upper and lower tie plates.

[

]

The 11x11 rod array is comprised of 92 full length fuel rods, 8 long PLFR and 12 short PLFRs. The PLFRs are captured in the LTP grid to prevent axial movement.

The fuel rod pitch is uniform in the upper section of the assembly relative to a non-uniform pitch in the lower section. The array of fuel rods remains orthogonal throughout the assembly.

The nine ULTRAFLOW™ spacers are **[** **]** and utilize **[**

]

[

]

Details of the fuel design characteristics are presented in Table 3-1 and Table 3-2 along with the equivalent values for the ATRIUM 10XM fuel design which is currently used and licensed in the Monticello unit.

Table 3-1 Fuel Assembly and Component Description

Table 3-2 Fuel Channel and Fastener Description

[ ]

4.0 THERMAL-MECHANICAL LIMITS METHODOLOGY

The LHGR limit is established to support plant operation while satisfying the fuel thermal-mechanical design criteria. The methodology for performing the fuel rod evaluation is described in Reference 3. The extension of these methods to fuel incorporating chromia is described in Reference 6. Fuel rod design criteria evaluated by the methodology are contained in References 3 and 11.

Fuel rod power histories are generated as part of the methodology for equilibrium cycle conditions as well as cycle-specific operation. These power histories include the effect of channel bow as described in Reference 3. The uncertainties of the important physical phenomenon are taken into account in the categories of operating power uncertainties, code model parameter uncertainties, and fuel manufacturing tolerances. In addition, adjustments are made to the power history inputs for possible differences in planned versus actual operation. Upper limits on the analysis results are obtained for comparison to the design limits for fuel melt, cladding strain, rod internal pressure and other characteristics as described by the design criteria.

Since the power history inputs, which include LHGR, fast neutron flux, reactor coolant pressure and reactor coolant temperature, are used as input to the analysis, the results explicitly account for conditions representative of the ATRIUM 11 operation. The resulting LHGR limit is used to monitor the fuel so it is maintained within the same maximum allowable steady-state power envelope as analyzed.

5.0 THERMAL-HYDRAULICS

5.1 *ATRIUM 11 Void Fraction*

The **[ ]** void-quality correlation has been qualified by Framatome against both the FRIGG void measurements, ATRIUM-10 and ATRIUM 10XM measurements. 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 **[ ]** respectively. **[**

] the use of the **[ ]** correlation for ATRIUM 11 is justified.

The ATRIUM 11 **[ ]** void fraction measurements. S-RELAP5 was assessed against previous measurements based upon fundamental hydraulic characteristics. The Marviken assembly of FRIGG had a 2-sigma error of **[ ]** in void prediction. The ATRIUM-10 has a 2-sigma error of **[ ]** for void. **[ ]**; therefore, the use of a 2-sigma error of **[ ]** is justified for the ATRIUM 11.

5.2 *ACE/ATRIUM 11 Critical Power Ratio Correlation*

The critical power ratio (CPR) correlation used in MICROBURN-B2, SAFLIM3D, S-RELAP5, RAMONA5-FA, and X-COBRA is based on the ACE/ATRIUM 11 critical power correlation described in Reference 7. As with all Framatome correlations, the range of applicability is enforced in Framatome methods through automated bounds checking and corrective actions. The ATRIUM 11 bounds checking process is similar to the ATRIUM 10XM as provided in Table 5-1. The ACE CPR correlation uses K-factor values to account for rod local peaking, rod location and bundle geometry effects.

The K-factor parameter is described in detail in Section 6.10 of Reference 7.

The ranges of applicability of the ACE/ATRIUM 11 and ACE/ATRIUM 10XM are compared in Table 5-2.

Table 5-1 ACE/ATRIUM 10XM Bounds Checking

[ ]

Table 5-2 Comparison of the Range of Applicability for the ACE/ATRIUM 11 and ACE/ATRIUM 10XM Correlations

[ ]

5.3 *Loss Coefficients*

Wall friction and component loss coefficients were determined for Monticello based on single-phase testing of a prototypic ATRIUM 11 fuel assembly in the PHTF. Prototypical fuel rods, spacer grids, flow channel, upper tie plate and lower tie plate were used in the testing. A description of the PHTF facility and an overview of the process for determining the component loss coefficients are described in Reference 12.

The ATRIUM 11 PHTF tests form the basis for the single phase loss coefficients currently used for design and licensing analyses supporting U.S. BWRs. The PHTF is used by Framatome to obtain single phase loss coefficients for the spacers. The friction factor correlation is a Reynolds dependent function based on the Moody friction model and the measured surface roughness. The pressure drops across the spacers are measured in the PHTF for each new design. **[**

]

The wall friction and component loss coefficients determined from the PHTF and utilized in the validation of the MICROBURN-B2 pressure drop model for the ATRIUM 11 fuel design are provided in Table 5-3.

PHTF data was reduced to determine single phase losses for the spacers in the **[**

] of the bundle. The values have been selected because they are representative of the hydraulic characteristics of the actual ATRIUM 11 fuel assemblies loaded into the reactor.

The modeling of the two-phase spacer pressure drop multiplier for the ATRIUM 11 fuel design has been confirmed with two-phase pressure drop measurements taken in the KATHY facility.

Figure 5-1 shows measured versus the MICROBURN-B2 predicted two phase pressure drop for a range of conditions. This figure confirms the applicability of the thermal-hydraulic models to predict pressure drop for the ATRIUM 11 design.

Table 5-3 Hydraulic Characteristics of ATRIUM 11 Fuel Assemblies

[

]

* Loss coefficients are referenced to the adjacent assembly bare rod flow area.

† [

]

Figure 5-1 Measured versus Predicted (MICROBURN-B2) Bundle Pressure Drop

6.0 TRANSIENTS AND ACCIDENTS

6.1 *Void Calculation Uncertainties*

The Framatome analyses methods and the correlations used are applicable for all Framatome designs in EFW conditions. The approach for addressing bias and uncertainties in the void calculation remains unchanged and is applicable for Monticello operation with the ATRIUM 11 fuel design.

The OLMCPR is determined based on the safety limit MCPR (SLMCPR) methodology and the transient analysis (ΔCPR) methodology. Void prediction uncertainty is not a direct input to either of these methodologies; however, the impact of void prediction uncertainty is inherently incorporated in both methodologies as discussed below.

The SLMCPR methodology explicitly considers important uncertainties in the Monte Carlo calculation performed to determine the number of rods in boiling transition. One of the uncertainties considered in the SLMCPR methodology is the bundle power uncertainty. This uncertainty is determined through comparison of calculated to measured core power distributions. Any miscalculation of void conditions will increase the error between the calculated and measured power distributions and be reflected in the bundle power uncertainty. Therefore, void prediction uncertainty is an inherent component of the bundle power uncertainty used in the SLMCPR methodology.

The transient analyses methodology utilizes a **[**

]

The transient analyses methodology results in predicted power increases that are bounding relative to benchmark tests. In addition, for licensing calculations [

] as defined in the transient analyses methodology. Therefore, uncertainty in the void prediction is inherently incorporated in the transient analysis methodology.

In addition to the impact of void prediction uncertainty being inherently incorporated in the analytical methods used to determine the OLMCPR, biasing of important input parameters in licensing calculations provides additional conservatism in establishing the OLMCPR. No additional adjustments to the OLMCPR are required to address void prediction uncertainty.

6.2 *Assessment of the Void-Quality Correlation*

As discussed in Section 5.1, the [ ] is equally applicable to the ATRIUM 11 applications at Monticello.

6.3 [ ]

[

]

Section 3.5.2.7 documented the NRC's review of this response as such:

> *However, the NRC staff does not agree with AREVA's third response.* **[**
>
> **]**

The result of this conclusion was Limitation and Condition 12 of AURORA-B AOO which requires plant-specific approval for any changes made to the transient coolant mixing. This section is intended to provide the description of the method used to determine **[**

].

6.3.1 Transient Mixing Determination

For Monticello, the mixing is evaluated using **[**

]

[

]

[ ]

Figure 6-1 [ ]

6.3.2 Implementation in AURORA-B AOO Licensing

Once the amount of mixing has been determined, the AURORA-B licensing model will be constructed. In order to ensure a conservative estimation of mixing is used, **[**

]

6.4 *Control Rod Drop Accident*

[

]

[ ]

Figure 6-2 Total Enthalpy Rise with CHF Multipliers

Figure 6-3 [ ]

[ ]

Figure 6-4 [ ]

6.5 *Loss of Coolant Accident*

The approved AURORA-B LOCA methodology, Reference 10, has been approved to be applicable to BWR/3 to BWR/6 with conditions extending up to EPU with extended flow windows. In addition, Limitation and Condition 27 of Reference 10 addresses the application of the methodology to **[**

].

6.6 *AURORA-B AOO Time Step Size*

Section 6.8.2 of ANP-10300P-A, Reference 1, provides a discussion of a time step size sensitivity study using the AURORA-B AOO methodology. The conclusion of this section states:

> **[**
>
> **]**

[

]

7.0 ATWS

7.1 *ATWS General*

The AURORA-B methodology is used for the ATWS overpressurization analysis. The ACE/ATRIUM 11 critical power correlation pressure limit is not a factor in the analysis.

Dryout might occur in the limiting (high power) channels of the core during the ATWS event. For the ATWS overpressurization analysis, ignoring dryout for the hot channels is conservative in that it maximizes the heat transferred to the coolant and results in a higher calculated pressure.

The ATWS event is not limiting relative to acceptance criteria identified in 10 CFR 50.46. The core remains covered and adequately cooled during the event. Following the initial power increase during the pressurization phase, the core returns to natural circulation conditions after the recirculation pumps trip and fuel cladding temperatures are maintained at acceptable low levels. The ATWS event is significantly less limiting than the loss of coolant accident relative to 10 CFR 50.46 acceptance criteria.

7.2 *Void Prediction*

Framatome performs cycle-specific ATWS analyses of the short-term reactor vessel peak pressure using the AURORA-B methodology. The ATWS peak pressure calculation is a core-wide pressurization event that is sensitive to similar phenomenon as other pressurization transients. Bundle design is included in the development of input for the coupled neutronic and thermal-hydraulic S-RELAP5 core model. Important inputs to the S-RELAP5 system model are biased in a conservative direction.

The Framatome transient analysis methodology utilizes a non-parametric uncertainty analysis which includes the uncertainties in individual phenomena. The methodology

[

]

[

] As demonstrated in Section 5.1 the void prediction is robust for past and present designs including ATRIUM 11.

The reference ATWS analysis evaluation presented in the topical report (Reference 1) of the core active density response, which is closely related to the void response, showed minimal changes in the peak vessel pressure. A study was also performed for the ASME overpressure event (FWCF) with similar results.

7.3 *ATWS Containment Heatup*

Fuel design differences may impact the power and pressure excursion experienced during the ATWS event. This in turn may impact the amount of steam discharged to the suppression pool and containment.

[

]

Table 7-1 [

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[

]

Table 7-2 [

]

[

]

* Boron worth is quoted as a positive value since it refers to the boron defect. The ppm boron used is 660 at 68 F. The calculation uses the equivalent boron at 319.2 F.

8.0 NEUTRONICS

From the neutronics perspective, the ATRIUM 11 fuel design differs from the ATRIUM 10XM fuel design primarily in the number and position of part length rods and the diameter, pitch and position of the fuel rods. The CASMO-4 code is designed to model a wide range of fuel rod diameters and pitches. The neutronic models have already been demonstrated to accurately model the vacant positions and this continues to be true for the ATRIUM 11 fuel design.

8.1 *Shutdown Margin*

The part length rod in the corner of the assembly improves the shutdown margin performance of the fuel design because of the flux trap that is created in the cold condition with the vacant rod position of all four assemblies in a control cell being in close proximity. The heterogeneous solution of CASMO-4 accurately models the vacant rod position and the associated reactivity. No change in predicted hot operating or cold critical eigenvalue is anticipated with the ATRIUM 11 fuel design.

8.2 *Bypass Modeling*

The bypass behavior of the ATRIUM 11 fuel design is identical to the ATRIUM 10XM fuel design, thus there is no difference in the modeling. Any differences in bypass heat deposition are treated explicitly.

Cycle-specific validation that the allowable bypass voiding at the LPRM D level has been met will continue to be performed. This validation will be documented in the corresponding Reload Safety Analysis Report for that cycle.

9.0 REFERENCES

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

2. ANP-10333P-A Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA) ," Framatome Inc., March 2018

3. BAW-10247PA Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," AREVA NP Inc., February 2008.

4. BAW-10247PA Revision 0 Supplement 1P-A Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 1: Qualification of RODEX4 for Recrystallized Zircaloy-2 Cladding," AREVA Inc., April 2017.

5. 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.

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

7. ANP-10335P-A Revision 0, "ACE/ATRIUM 11 Critical Power Correlation," Framatome Inc., May 2018.

8. EMF-93-177P-A Revision 1 Supplement 1P-A Revision 0, "Mechanical Design for BWR Fuel Channels Supplement 1: Advanced Methods for New Channel Designs," AREVA Inc., September 2013.

9. EMF-93-177 Revision 1 Supplement 2P-A Revision 1, “Mechanical Design for BWR Fuel Channels: Z4B Material,” Framatome Inc., June 2019

10. ANP-10332P-A Revision 0, “AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios,” March 2019.

11. ANF-89-98(P)(A) Revision 1 and Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995.

12. I.K. Madni, et al., "Development of Correlations for Pressure Loss/Drop Coefficients Obtained From Flow Testing of Fuel Assemblies In Framatome

ANP’S PHTF," Paper Number 22428, Proceedings of ICONE10, Arlington, VA, April 14-18,2002.

13. [ ]

14. [ ]

15. [ ]

16. [ ]

17. [ ]

18. [ ]

19. [ ]

20. Regulatory Guide 1.236, *Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents*, June 2020 (NRC ADAMS ML20055F490).

21. Draft Regulatory Guide 1327, *Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents*, November 2016 (NRC ADAMS ML16124A200).

APPENDIX A APPLICATION OF FRAMATOME METHODOLOGY FOR MIXED CORES

A.1. DISCUSSION

Framatome has considerable experience analyzing fuel design transition cycles and has methodology and procedures to analyze mixed cores composed of multiple fuel types. For each core design, analyses are performed to confirm that all design and licensing criteria are satisfied. The analyses performed explicitly include each fuel type in the core. The analyses consider the cycle-specific core loading and use input data appropriate for each fuel type in the core. The mixed core analyses are performed using generically approved methodology in a manner consistent with NRC approval of the methodology. Based on results from the analyses, operating limits are established for each fuel type present in the core. During operation, each fuel type is monitored against the appropriate operating limits.

Thermal-hydraulic characteristics are determined for each fuel type that will be present in the core. The thermal-hydraulic characteristics used in core design, safety analysis, and core monitoring are developed on a consistent basis for both Framatome fuel and other vendor co-resident fuel to minimize variability due to methods. For Monticello operation, the entire core will be composed of Framatome fuel designs.

For core design and nuclear safety analyses, the neutronic cross-section data is developed for each fuel type in the core using CASMO-4. MICROBURN-B2 is used to design the core and provide input to safety analyses (core neutronic characteristics, power distributions, etc.). Each fuel assembly is explicitly modeled in MICROBURN-B2 using cross-section data from CASMO-4 and geometric data appropriate for the fuel design.

Fuel assembly thermal-mechanical limits for all fuel are verified and monitored for each mixed core designed by Framatome. Framatome performs design and licensing analyses to demonstrate that Framatome fuel rod designs meet steady state limits and that transient limits are not exceeded during operational occurrences.

The CPR is evaluated for each fuel type in the core using calculated local fluid conditions and an appropriate critical power correlation. Fuel type specific correlation coefficients for Framatome fuel are based on data from the Framatome critical power test facility. The ACE/ATRIUM 10XM critical power correlation will be used for monitoring ATRIUM 10XM fuel present during the transition to operation with ATRIUM 11 at Monticello. The CPR correlation used for the ATRIUM 11 fuel is the ACE/ATRIUM 11 critical power correlation described in Reference 7. The ACE CPR correlation uses K-factor values to account for rod local peaking, rod location and bundle geometry effects.

In the safety limit MCPR analysis each fuel type present in the core is explicitly modeled using appropriate geometric data, thermal-hydraulic characteristics, and power distribution information (from CASMO-4 and MICROBURN-B2 analyses). CPR is evaluated for each assembly using fuel type specific correlation coefficients. Plant and fuel type specific uncertainties are considered in the statistical analysis performed to determine the safety limit MCPR. The safety limit MCPR analysis is performed each cycle and uses the cycle specific core configuration.

An operating limit MCPR is established for each fuel type in the core. For fast transients the AURORA-B code (Reference 1) is used to determine the overall system and hot channel response. The core nuclear characteristics used in AURORA-B are obtained from MICROBURN-B2 and reflect the actual core loading pattern. Critical power performance is evaluated using local fluid conditions and fuel type specific CPR correlation coefficients. The transient CPR response is used to establish an operating limit MCPR for each fuel type.

For transient events that are sufficiently slow such that the heat transfer remains in phase with changes in neutron flux during the transient, evaluations are performed with steady state codes such as MICROBURN-B2 in accordance with NRC approval. Such slow transients are modeled by performing a series of steady state solutions with appropriate boundary conditions using the cycle specific design core loading plan. Each fuel assembly type in the core is explicitly modeled. The change in CPR between the initial and final condition after the transient is determined, and if the CPR change is more severe than those determined from fast transient analyses, the slow transient result is used to determine the MCPR operating limit.

MAPLHGR operating limits are established and monitored for each fuel type in the core to ensure that 10 CFR 50.46 acceptance criteria are met during a postulated LOCA. The S-RELAP5 code is used to determine the overall system and hot channel response during a postulated LOCA. While system analyses are typically performed on an equilibrium core basis, the thermal-hydraulic characteristics of all fuel assemblies in the core are considered to ensure the LOCA analysis results are applicable to mixed core configurations.

The core monitoring system will monitor each fuel assembly in the core. Each assembly is modeled with geometric, thermal-hydraulic, neutronic, and CPR correlation input data appropriate for the specific fuel type. Each assembly in the core will be monitored relative to thermal limits that have been explicitly developed for each fuel type.

In summary, Framatome methodology is used consistent with NRC approval to perform design and licensing analyses for mixed cores. The cycle design and licensing analyses explicitly consider each fuel type in mixed core configurations. Limits are established for each fuel type and operation within these limits is verified by the monitoring system during operation.

ENCLOSURE

ATTACHMENT 3b

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST
APPLICATION TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

AFFIDAVIT FOR

ANP-3924P REPORT, REVISION 0

APPLICABILITY OF FRAMATOME BWR METHODS TO
MONTICELLO WITH ATRIUM 11 FUEL

JUNE 2021

(3 pages follow)

AFFIDAVIT

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-3924P Revision 0, "Applicability of Framatome BWR Methods to Monticello with ATRIUM 11 Fuel," dated June 2021 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: June 18, 2021

Alan B. Meginnis

ENCLOSURE

ATTACHMENT 4a

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST
APPLICATION TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

ANP-3882NP REPORT, REVISION 0

MECHANICAL DESIGN REPORT FOR MONTICELLO ATRIUM 11 FUEL ASSEMBLIES

LICENSING REPORT

MARCH 2021

(43 pages follow)

Mechanical Design Report for Monticello ATRIUM 11 Fuel Assemblies

ANP-3882NP
Revision 0

Licensing Report

March 2021

Nature of Changes

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

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Nomenclature

Acronym	**Definition**
AFC	Advanced fuel channel
AOO	Anticipated operational occurrences
ASME	American Society of Mechanical Engineers
B&PV	Boiler and pressure vessel
BQ	Beta-quench
BWR	Boiling water reactor
ELC	Enhanced load chain
EOL	End of life
FEA	Finite element analysis
HALC	Harmonized advanced load chain
KWUSTOSS	Framatome finite element analysis code
LTA	Lead test assembly
LOCA	Loss-of-coolant accident
LTP	Lower tie plate
MWd/kgU	Megawatt-days per kilogram of Uranium
NRC	U. S. Nuclear Regulatory Commission
PLFR	Part-length fuel rods
RSA	Response spectrum analysis
SRP	Standard review plan
UTP	Upper tie plate
Z4B	Proprietary Zircaloy BWR material similar to Zircaloy-4
Zry-4	Zircaloy-4
Zry-2	Zircaloy-2
3GFG	3rd Generation FUELGUARD

1.0 INTRODUCTION

This report documents the successful completion of all licensing analyses and related testing necessary to verify that the mechanical design criteria are met for the ATRIUM™ 11 Fuel Assemblies supplied by Framatome Inc. (Framatome) for insertion into Monticello Unit 1. This report also provides a description of the mechanical design and licensing methods. The scope of this report is limited to an evaluation of the mechanical design of the fuel assembly and fuel channel.

The ATRIUM 11 design is a Framatome advanced boiling water reactor (BWR) fuel design that builds on the history of proven ATRIUM family of fuel designs. The design uses an 11x11 fuel array, a [ ] fuel rod, a central water channel that displaces a 3x3 array of rods and is made from an advanced Zirconium alloy Z4B™ material, a modular lower tie plate with a 3rd generation FUELGUARD™ and nine ULTRAFLOW™ spacer grids [ ].

The fuel assembly structural design evaluation is not cycle-specific so this report is intended to be referenced for each cycle where the fuel design is in use. Minor changes to the fuel design and cycle-specific input parameters will be dispositioned for future reloads. Framatome will confirm the continued applicability of this report prior to delivery of each subsequent reload of ATRIUM 11 fuel at Monticello in a cycle specific compliance document.

The fuel assembly design was evaluated according to the Framatome BWR generic mechanical design criteria (Reference 1). The fuel channel design was evaluated to the criteria given in the fuel channel topical reports (References 2 and 3). The generic design criteria have been approved by the U.S. Nuclear Regulatory Commission (NRC) and the criteria are applicable to the subject fuel assembly and fuel channel design. Mechanical analyses have been performed using NRC-approved design analysis methodology (References 1, 2, 3 and 4). The methodology permits maximum licensed assembly and fuel channel exposures of [ ] (Reference 4, Section 1.0).

The fuel assembly and fuel channel meet all mechanical compatibility requirements for use in Monticello Unit 1. This includes compatibility with both co-resident fuel and the reactor core internals.

2.0 DESIGN DESCRIPTION

This section provides a design description of the ATRIUM 11 fuel assembly and fuel channel. Reload-specific design information is available in the design package provided by Framatome for each reload delivery.

2.1 *Overview*

Monticello has successfully operated for several cycles with reload quantities of ATRIUM 10XM fuel assemblies. Monticello will operate with ATRIUM 11 fuel assemblies in reload quantities starting with Monticello Unit 1 Cycle 32. The ATRIUM 11 bundle consists of an 11x11 fuel lattice with a square internal water channel that displaces a 3x3 array of rods.

The ATRIUM 11 incorporates key design features relative to previous ATRIUM designs as described in Reference 5.

Table 2-1 lists the key design parameters of the ATRIUM 11 fuel assembly.

2.1.1 Fuel Assembly

Figure 2-1 provides an illustration of the fuel assembly, and Table 2-1 lists the main fuel assembly attributes. The fuel assembly is accompanied by a fuel channel, as described later in this section.

[ ]

2.1.2 Upper Tie Plate and Connecting Hardware

Figure 2-2 provides an illustration of the UTP and connecting hardware.

2.1.3 Water Channel

Figure 2-1 and Figure 2-2 provides an illustration of the water channel, and Table 2-1 lists the main water channel attributes.

2.1.4 Spacer Grid

Figure 2-3 provides illustration of the spacer grid, and Table 2-1 lists the main spacer grid attributes.

[ ]

2.1.5 Lower Tie Plate

Figure 2-4 provides an illustration of the 3GFG FUELGUARD.

2.1.6 Fuel Rods

This mechanical design report documents the fuel structural analyses. The fuel rod thermal-mechanical report provides fuel rod design description detail. Figure 2-5 provides an illustration of the full-length and the two part-length fuel rods.

[

]

2.2 *Fuel Channel and Components*

Figure 2-6 provides an illustration of the fuel channel and components, and Table 2-2 lists the fuel channel component attributes.

Table 2-1
Fuel Assembly and Component Description

[ ]

Table 2-2
Fuel Channel and Channel Spacer Assembly Description

3.0 FUEL DESIGN EVALUATION

This section provides a summary of the mechanical methodology and results from the structural design evaluations. Results from the mechanical design evaluation demonstrate that the design satisfies the mechanical criteria to the analyzed exposure limit. Sections 3.1 through 3.4 correspond to the fuel assembly criteria sections within Section 3.0 of Reference 1. Section 3.5 and Table 3-2 corresponds to the advanced fuel channel criteria sections within Table 1.1 and 1.2 of Reference 2.

3.1 *Objectives*

The objectives of designing fuel assemblies (systems) to specific criteria are to provide assurance that:

- The fuel assembly (system) shall not fail as a result of normal operation and anticipated operational occurrences (AOOs). The fuel assembly (system) dimensions shall be designed to remain within operational tolerances, and the functional capabilities of the fuel shall be established to either meet or exceed those assumed in the safety analysis.
- Fuel assembly (system) damage shall never prevent control rod insertion when it is required.
- The number of fuel rod failures shall be conservatively estimated for postulated accidents.
- Fuel coolability shall always be maintained.
- The mechanical design of fuel assemblies shall be compatible with co-resident fuel and the reactor core internals.
- Fuel assemblies shall be designed to withstand the loads from handling and shipping.

The first four objectives are those cited in the Standard Review Plan (SRP). The latter two objectives are to assure the structural integrity of the fuel and the compatibility with the existing reload fuel. To satisfy these objectives, the criteria are applied to the fuel rod and the fuel assembly (system) designs. Specific component criteria are also necessary to assure compliance. The criteria established to meet these objectives include those given in Chapter 4.2 of the SRP.

3.2 *Fuel Rod Evaluation*

The mechanical design report documents the fuel structural analyses only. The fuel rod evaluation will be documented in the fuel rod thermal-mechanical report. However, the fuel rod mechanical fracturing (Reference 1, Section 3.2.7) is evaluated in Section 3.4.4 *Structural Deformations.*

3.3 *Fuel System Evaluation*

The detailed fuel system design evaluation is performed to ensure the structural integrity of the design under normal operation, AOO, faulted conditions, handling operations, and shipping. The analysis methods are based on fundamental mechanical engineering techniques, often employing finite element analysis, prototype testing, and correlations based on in-reactor performance data. Summaries of the major assessment topics and associated testing are described in the sections that follow.

Prototype testing is an essential element of Framatome methodology for demonstrating compliance with structural design requirements. Results from design verification testing may directly demonstrate compliance with criteria or may be used as input to design analyses.

Testing performed to qualify the mechanical design or evaluate assembly characteristics includes:

- Fuel assembly axial load structural strength
- Fuel assembly fretting
- Fuel assembly static lateral deflection
- Fuel assembly lateral vibration
- Fuel assembly impact
- Spacer grid lateral impact strength
- Tie plate lateral load strength

3.3.1 Stress, Strain, or Loading Limits on Assembly Components

The structural integrity of the fuel assemblies is assured by setting design limits on stresses and deformations due to various handling, operational, AOOs, and accident or faulted loads. Framatome uses Section III of the American Society of Mechanical Engineers (ASME) boiler and pressure vessel (B&PV) code as a guide to establish acceptable stress, deformation, and load limits for standard assembly components. These limits are applied to the design and evaluation of the UTP, LTP, spacer grids, springs, and load chain components, as applicable.

All significant loads experienced during normal operation, AOOs, and under faulted conditions are evaluated to confirm the structural integrity of the fuel assembly components. Outside of faulted conditions, most structural components are under the most limiting loading conditions during fuel handling. See Section 3.3.9 for a discussion of fuel handling loads and Section 3.4.4 for the structural evaluation of faulted conditions. Although normal operation and AOO loads are often not limiting for structural components, a stress evaluation may be performed to confirm the design margin and to establish a baseline for adding accident loads. The stress calculations use conventional, open-literature equations. A general-purpose, finite element stress analysis code, such as ANSYS, may be used to calculate component stresses.

3.3.2 Fatigue

Section addressed in the fuel rod thermal-mechanical report.

3.3.3 Fretting Wear

Fuel rod failures due to grid-to-rod fretting shall not occur. [

].

Fretting wear is evaluated by testing, as described in Section 3.3.3.1. The testing is conducted by [

]. The inspection measurements for wear are documented. The lack of significant wear demonstrates adequate rod restraint geometry at the contact locations. Also, the lack of significant wear at the spacer cell locations [ ] provides further assurance that no significant fretting will occur at higher exposure levels.

3.3.3.1 Fuel Assembly Fretting Test

A fretting test was conducted on a full-size test assembly to evaluate the ATRIUM 11 fuel rod support design. [ ]. After the test, the assembly was inspected for signs of fretting wear. No significant wear was found on fuel rods in contact with spacer springs [ ]. The results agree with past test results on BWR designs where no noticeable wear was found on the fuel rods or other interfacing components following exposure to coolant flow conditions.

3.3.4 Oxidation, Hydriding, and Crud Buildup

Section addressed in the fuel rod thermal-mechanical report.

3.3.5 Rod Bow

The predicted rod-to-rod gap closure due to bow is assessed by thermal hydraulics group for impact on thermal margins.

Differential expansion between the fuel rods and cage structure, and lateral thermal and flux gradients can lead to lateral creep bow of the rods in the spans between spacer grids. This lateral creep bow alters the pitch between the rods and may affect the peaking and local heat transfer. The Framatome design criterion for fuel rod bowing is [ ].

Visual exams on ATRIUM 11 have not revealed any unusual fuel rod bow behavior for exposures up to [ ] based on the latest experience from Lead Test Assembly post-irradiation exams. This exposure is beyond the threshold where increasing rod bow had been observed on other designs. Therefore, the ATRIUM 11 fuel design has been shown to have minimal rod bow. A rod gap closure ratio curve is provided in Reference 4.

3.3.6 Axial Irradiation Growth

Reference 4 requires [

] .

The fuel rod growth correlation is established from [

] .

Assembly growth is established from ATRIUM 10x10 and 11x11 arrayed fuel assemblies with water channels made of Z4B material. It is based on the ATRIUM fuel assembly growth data only and excludes designs with load bearing tie rods as well as the European bundle-in-basket designs. [

] .

The fuel rod and assembly growth approved correlations are described within Reference 4 along with the respective tolerance limits.

3.3.7 Rod Internal Pressure

Section addressed in the fuel rod thermal-mechanical report.

3.3.8 Assembly Lift-off

Fuel assembly lift-off is evaluated under both normal operating conditions (including AOOs) and under faulted conditions. 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.

For normal operating conditions, the net axial force acting on the fuel assembly is calculated by adding the loads from gravity, hydraulic resistance from coolant flow, difference in fluid flow entrance and exit momentum, and buoyancy. The calculated net force is confirmed to be in the downward direction, indicating no assembly lift-off. [

].

Mixed core conditions for assembly lift-off are considered on a cycle-specific basis, as determined by the plant operating conditions and other fuel types. Analyses to date indicate a large margin to assembly lift-off under normal operating conditions.

For faulted conditions, [

]. The fuel will not lift under normal or AOO conditions, it will not become disengaged from the fuel support under faulted conditions, nor block insertion of the control blade in all operating conditions.

3.3.9 Fuel Assembly Handling

The fuel assembly shall withstand, without permanent deformation, all normal axial loads from shipping and fuel handling operations. Analyses or testing shall demonstrate that the fuel is capable of [ ].

The fuel assembly structural components are assessed for axial fuel handling loads by analyses and testing. To demonstrate compliance with the criteria, the tests and analyses are performed by loading a test assembly or the individual components of the load chain to an axial tensile force greater than [ ]. An acceptable test and analysis demonstrates no yielding after loading.

Handling requirements for the fuel rod plenum spring are addressed in the fuel rod thermal-mechanical report.

3.3.9.1 Fuel Assembly Axial Load Tests

Each test is used in support of analytical or Finite Element Analysis to demonstrate that no significant permanent deformation occurs for loads [ ] .

Descriptions of tests:

[ ]

3.3.10 Miscellaneous Components

3.3.10.1 Compression Spring Forces

The compression spring force shall support the weight of the upper tie plate and fuel channel throughout the design life of the fuel. The ATRIUM 11 has a single large compression spring mounted on the central water channel. The compression spring serves the same function as previous ATRIUM family of fuel designs by providing support for the UTP and fuel channel. The spring force is calculated based on the installed deflection and specified spring force requirements to meet support criteria. Irradiation-induced relaxation is taken into account for EOL conditions. The minimum compression spring force at EOL is greater than the combined weight of the UTP assembly and fuel channel assembly. Since the compression spring design of the ATRIUM family of fuel assemblies load chain designs do not interact with the fuel rods, no consideration is required for fuel rod buckling loads.

3.3.10.2 LTP Seal Spring

The LTP seal spring shall limit the bypass coolant leakage rate between the LTP and fuel channel. The seal spring shall accommodate expected fuel channel deformation while remaining in contact with the fuel channel. Also, the seal spring shall have adequate corrosion resistance and be able to withstand the operating stresses without yielding.

Flow testing is used to confirm acceptable bypass flow characteristics. The seal spring is designed with adequate deflection to accommodate the maximum expected fuel channel bulge while maintaining acceptable bypass flow. [ ] is selected as the material because of its high strength at elevated temperature and its excellent corrosion resistance. Seal spring stresses are analyzed using a finite element method.

3.4 *Fuel Coolability*

For accidents in which severe fuel damage might occur, core coolability and the capability to insert control blades are essential. Chapter 4.2 of the SRP provides several specific areas important to fuel coolability, as discussed below.

3.4.1 Cladding Embrittlement

The loss-of-coolant accident (LOCA) evaluation is addressed in the Monticello LOCA maximum average planar linear heat generation rate (MAPLHGR) analysis for ATRIUM 11 fuel report.

3.4.2 Violent Expulsion of Fuel

Results for the control rod drop accident (CRDA) analysis are presented in the Monticello ATRIUM 11 fuel transition report and the subsequent cycle-specific Monticello reload licensing report.

3.4.3 Fuel Ballooning

The LOCA evaluation is addressed in the Monticello LOCA MAPLHGR analysis for ATRIUM 11 fuel report.

3.4.4 Structural Deformations

ATRIUM 11 structural component deformations or stresses from postulated accidents are limited according to requirements contained in the ASME B&PV Code, Section III, Division 1, Appendix F, and SRP Section 4.2, Appendix A.

The methodology for analyzing the fuel under the influence of accident loads is described in the Mechanical Designs for BWR Fuel Channels Topical Report (Reference 2) and is further discussed in Section 3.5.2. Evaluations performed for the fuel under accident conditions include [ ] .

The fuel channel is the most limiting component because it resists the majority of the bending moment. A limiting uniform horizontal [ ] was calculated for all AFC designs. An analysis was performed to ensure the structural integrity of the ATRIUM 11 components under the limiting fuel channel horizontal acceleration applied to an [ ] as a static load. The uniform horizontal [ ] was shown to bound the maximum fuel assembly acceleration at Monticello.

[ ] .

The reactor pressure vessel and internals analysis of record was assessed to evaluate the impact of the introduction of the ATRIUM 11 based on the methodology established for the ATRIUM 10XM fuel transition. Based on the similarity of fuel assembly designs, and an insignificant change in mass between the ATRIUM 11 and the analysis of record fuel assemblies, the analysis of record remains applicable with the introduction of the ATRIUM 11 fuel design.

The High Density Spent Fuel Storage Racks (HDSFSR) analysis accounts for the fuel as added mass in calculating the structural integrity under postulated seismic loads. The weights of legacy fuel assembly designs at Monticello encompass the weight of the ATRIUM 11 fuel design. Therefore, the HDSFSR remains applicable with the introduction of the ATRIUM 11 fuel design.

3.4.4.1 Test Verifications

Fuel assemblies are tested with, and without, a fuel channel as described in Appendix C of Reference 2. Testing is performed to obtain the dynamic characteristics of the fuel assembly and spacer grids. The stiffness, natural frequencies and damping values derived from the tests are used as inputs for analytical models of the fuel assembly and fuel channel. In general, the testing and analyses have shown the dynamic response of ATRIUM 11 to be similar to ATRIUM-10 and ATRIUM 10XM fuel assemblies.

3.4.4.1.1 Fuel Assembly Static Lateral Deflection Test

A lateral deflection test is performed to determine the fuel assembly stiffness, both with and without a fuel channel. The stiffness is obtained by supporting the fuel assembly at the two ends in a vertical position, applying a side displacement at the central spacer location, and measuring the corresponding force.

3.4.4.1.2 Fuel Assembly Lateral Vibration Tests

The lateral vibration testing consists of both a free vibration test and a forced vibration test [ ].

The test setup for the free vibration test [ ].

The forced vibration test [ ]

[

] .

3.4.4.1.3 Fuel Assembly Impact Tests

Impact testing was performed in a similar manner to the lateral deflection tests. The unchanneled assembly is supported in a vertical position with both ends fixed. The assembly is displaced a specified amount and then released. [

] .

3.4.4.1.4 Spacer Grid Lateral Impact Strength Test

Spacer grid impact strength is determined by a [

] .

The maximum force prior to the onset of buckling was determined from the tests. The results were adjusted to reactor operating temperature conditions to establish an allowable lateral load.

3.4.4.1.5 Tie Plate Strength Tests

In addition to the axial tensile tests described in Section 3.3.9.1, a lateral load test is performed on the UTP and LTP.

The UTP lateral load test was conducted on a test machine which applied [

]. This provides a limiting lateral load for accident conditions.

To determine a limiting lateral load for accident conditions for the 3GFG LTP, a lateral load test was conducted by attaching the grid of the tie plate to a rigid vertical plate [

].

The results were adjusted to reactor operating temperature conditions to establish an allowable lateral load per Reference 1, Section 3.3.1.

3.5 *Fuel Channel and Components*

The fuel channel assembly design criteria are summarized below, and evaluation results are summarized in Table 3-2. The analysis methods are described in detail in Reference 2.

3.5.1 Design Criteria for Normal Operation

Stress due to Pressure Differential. The stress limits during normal operation are obtained from the ASME B&PV Code, Section III, Division 1, Subsection NG for Service Level A. The calculated stress intensities are due to the differential pressure across the fuel channel wall. The pressure loading includes the normal operating pressure plus the increase during AOO. The unirradiated properties of the fuel channel material are used since the yield and ultimate tensile strength increase during irradiation (Reference 7). As an alternative to the elastic analysis stress intensity limits, a plastic analysis may be performed as permitted by paragraph NB-3228.3 of the ASME B&PV Code.

In the case of AOOs, the amount of bulging is limited to that value which will permit control blade movement. During normal operation, any significant permanent deformation due to yielding is precluded by restricting the maximum stresses at the inner and outer faces of the fuel channel to be less than the yield strength.

Fatigue. Cyclic changes in power and flow during operation impose a duty loading on the fuel channel. The cyclic duty from pressure fluctuations is limited to less than the fatigue lifetime of the fuel channel. The fatigue life is based on the O'Donnell and Langer curve (Reference 6), which includes a factor of 2 on stress amplitude or a factor of 20 on the number of cycles, whichever is more conservative.

Oxidation and Hydriding. Oxidation reduces the material thickness and results in less load-carrying capacity. The fuel channels have thicker walls than other components (e.g., fuel rods), and the normal amounts of oxidation and hydrogen pickup are not limiting provided: the alloy composition and impurity limits are carefully selected; the heat treatments are also carefully chosen; and the water chemistry is controlled. [

].

Long-Term Deformation. Changes to the geometry of the fuel channel occur due to creep deformation during the long term exposure in the reactor core environment. Overall deformation of the fuel channel occurs from a combination of bulging and bowing. Bulging of the side walls occurs because of the differential pressure across the wall. Lateral bowing of the fuel channel is caused primarily from the neutron flux and thermal gradients. Too much deflection may prevent normal control blade maneuvers and it may increase control blade insertion time above the Technical Specification limits. The total fuel channel deformation must not stop free movement of the control blade. [

].

3.5.2 Design Criteria for Accident Conditions

Fuel Channel Stresses, Load Limit, and Vertical Acceleration. The criteria are based on the ASME B&PV Code, Section III, Appendix F, for faulted conditions (Service Level D). Component support criteria for elastic system analysis are used as defined in paragraphs F-1332.1 and F-1332.2. The unirradiated properties of the fuel channel material are used since the yield and ultimate tensile strength increase during irradiation (Reference 7). [

].

Vertical acceleration produces a membrane stress in the axial direction due to a postulated impact of the channeled fuel assembly impacting the fuel support after liftoff.

The amount of bulging remains limited to that value which will permit control blade insertion.

Channel Bending from Combined Horizontal Excitations. [

].

Fuel Channel Gusset Strength. [

].

Table 3-1
Results for ATRIUM 11 Fuel Assembly Criteria

Criteria Section	Description	Criteria	Results
ANF-89-98(P)(A) (Reference 1) Associated Mechanical Design Criteria Sections			
3.3	**Fuel System Criteria**		
3.3.1	Stress, strain and loading limits on assembly components	The ASME B&PV Code Section III is used to establish acceptable stress levels or load limits for assembly structural components. The design limits for accident conditions are derived from Appendix F of Section III.	[ ].
3.3.3	Fretting wear	[ ].	[ ].
3.3.5	Rod bow	Protect thermal limits	[ ]
3.3.6	Axial irradiation growth		
	Upper end cap clearance	Clearance always exists	[ ]
3.3.8	Assembly lift-off		
	Normal operation (including AOOs)	No lift-off from fuel support	[ ]
	Postulated accident	No disengagement from fuel support	[ ].

Table 3-1
Results for ATRIUM 11 Fuel Assembly Criteria
(Continued)

Criteria Section	Description	Criteria	Results
3.3	**Fuel System Criteria** (Continued)		
3.3.9	Fuel assembly handling	[ ]	Verified by testing and Analyses to meet requirement
3.3.10	Miscellaneous components		
3.3.10.1	Compression spring forces	Support weight of UTP and fuel channel throughout design life	The design criteria are met
3.3.10.2	LTP seal spring	Accommodate fuel channel deformation, adequate corrosion, and withstand operating stresses	The design criteria are met
3.4	**Fuel Coolability**		
3.4.4	Structural deformations	Maintain coolable geometry and ability to insert control blades. SRP 4.2, App. A, and ASME Section III, App. F.	[ ].

Table 3-2
Results for ATRIUM 11 Advanced Fuel Channel Criteria

Criteria Section	Description	Criteria	Results
EMF-93-177(P)(A) (Reference 2) Associated Fuel Channel (FC) Criteria Sections			
FC 3.2	**ATRIUM 11 Advanced Fuel Channel – Normal Operation**		
FC 3.2.1	Stress due to pressure differential	The pressure load including AOO is limited to [ ] according to ASME B&PV Code, Section III. The pressure load is also limited such that [ ].	The deformation during AOO remains within functional limits for normal control blade operation and the [ ] is met. There is no significant plastic deformation.
FC 3.2.2	Fatigue	Cumulative cyclic loading to be less than the design cyclic fatigue life for Zircaloy.	Expected number of cycles is less than allowable
FC 3.2.3	Oxidation and hydriding	Oxidation shall be accounted for in the stress and fatigue analyses	The maximum expected oxidation is low in relation to the wall thickness. Oxidation was accounted for in the stress and fatigue analyses.
FC 7.0	Long-term deformation (bulge creep and bow)	Bulge and bow shall not interfere with free movement of the control blade	Margin to a stuck control blade remains positive

Table 3-2
Results for ATRIUM 11 Advanced Fuel Channel Criteria
(Continued)

Criteria Section	Description	Criteria	Results
FC 3.3	**ATRIUM 11 Advanced Fuel Channel – Accident Conditions**		
FC 3.3.1	Fuel channel stresses and load limit and vertical accelerations	The pressure load is limited to [ ]. The pressure load is also limited such that [ ].	The deformation during blowdown does not interfere with control blade insertion. This also satisfies the less restrictive [ ].
FC 3.3.1 (continued)	Fuel channel bending from combined horizontal excitations	Allowable bending moment based on ASME Code, Section III, Appendix F [ ].	[ ]
FC 3.3.2	Fuel channel gusset strength	Vertical load must be less than ASME allowable load rating based on testing.	[ ].

4.0 REFERENCES

1. ANF-89-98(P)(A) Revision 1 and Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995.

2. EMF-93-177(P)(A) Revision 1, "Mechanical Design for BWR Fuel Channels," Framatome ANP Inc., August 2005.

3. EMF-93-177P-A Revision 1, Supplement 1P-A, Revision 0, "Mechanical Design for BWR Fuel Channels Supplement 1: Advanced Methods for New Channel Designs," AREVA NP Inc., September 2013.

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. ANP-3653P, Revision 0, "Fuel Design Evaluation for ATRIUM 11 BWR Reload Fuel," Information Design Report, Framatome Inc., September 2018.

6. W. J. O'Donnell and B. F. Langer, "Fatigue Design Basis for Zircaloy Components," Nuclear Science and Engineering, Volume 20, January 1964.

7. Huang, P. Y., Mahmood, S. T., and Adamson, R. B. "Effects of Thermomechanical Processing on In-Reactor Corrosion and Post-Irradiation Properties of Zircaloy-2," *Zirconium in the Nuclear Industry: Eleventh International Symposium*, ASTM STP 1295, E. R. Bradley and G. P. Sabol, Eds., American Society for Testing and Materials, 1996, pp. 726-757.

ENCLOSURE

ATTACHMENT 4b

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST
APPLICATION TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

AFFIDAVIT FOR

ANP-3882P REPORT, REVISION 0

MECHANICAL DESIGN REPORT FOR MONTICELLO ATRIUM 11 FUEL ASSEMBLIES

LICENSING REPORT

MARCH 2021

(3 pages follow)

AFFIDAVIT

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-3882P, Revision 0 "Mechanical Design Report for Monticello ATRIUM 11 Fuel Assemblies," dated March 2021 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: March 19, 2021

Alan Meginnis

ENCLOSURE

ATTACHMENT 5a

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST
APPLICATION TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

ANP-3893NP REPORT, REVISION 0

MONTICELLO THERMAL-HYDRAULIC DESIGN REPORT
FOR ATRIUM 11 FUEL ASSEMBLIES

MAY 2021

(27 pages follow)

Monticello Thermal-Hydraulic Design Report for ATRIUM 11 Fuel Assemblies

ANP-3893NP
Revision 0

May 2021

Nature of Changes

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

Contents

Tables

Figures

Nomenclature

AOO	anticipated operational occurrence
ASME	American Society of Mechanical Engineers
BWR	boiling water reactor
CFR	code of federal regulations
CHF	critical heat flux
CPR	critical power ratio
CRDA	control rod drop accident
LOCA	loss-of-coolant accident
LTP	lower tie plate
MAPLHGR	maximum average planar linear heat generation rate
MCPR	minimum critical power ratio
NRC	Nuclear Regulatory Commission, U.S.
OD	outside diameter
PLFR	part-length fuel rod
RPF	radial peaking factor
UTP	upper tie plate

1.0 INTRODUCTION

The results of Monticello thermal-hydraulic analyses are presented to demonstrate that Framatome ATRIUM 11 fuel is hydraulically compatible with the previously loaded ATRIUM 10XM fuel design. This report also provides the hydraulic characterization of the ATRIUM 11 and the coresident ATRIUM 10XM design for Monticello.

The generic thermal-hydraulic design criteria applicable to the design have been reviewed and approved by the U.S. Nuclear Regulatory Commission (NRC) in the topical report ANF-89-98(P)(A) Revision 1 and Supplement 1 (Reference 1). In addition, thermal-hydraulic criteria applicable to the design have also been reviewed and approved by the NRC in the topical report XN-NF-80-19(P)(A) Volume 4 Revision 1 (Reference 2).

2.0 SUMMARY AND CONCLUSIONS

ATRIUM 11 fuel assemblies have been determined to be hydraulically compatible with the coresident ATRIUM 10XM fuel design in the Monticello reactor for the entire range of the licensed power-to-flow operating map. Detailed calculation results supporting this conclusion are provided in Section 3.2 and Table 3.4 - Table 3.8.

The ATRIUM 11 fuel design is geometrically different from the coresident ATRIUM 10XM design, but the designs are hydraulically compatible. **[**

]

Core bypass flow (defined as leakage flow through the LTP flow holes, channel seal, core support plate, and LTP-fuel support interface) is not adversely affected by the introduction of the ATRIUM 11 fuel design. Analyses at rated conditions show a core bypass flow of **[ ]** of rated core flow for a full core of ATRIUM 10 XM fuel and a first transition core configuration and **[ ]** for a second transition core configuration and a full core of ATRIUM 11 fuel.

Analyses demonstrate the thermal-hydraulic design and compatibility criteria discussed in Section 3.0 are satisfied for the Monticello core consisting of ATRIUM 10XM fuel with ATRIUM 11 fuel for the expected core power distributions and core power/flow conditions encountered during operation.

3.0 THERMAL-HYDRAULIC DESIGN EVALUATION

Thermal-hydraulic analyses are performed to verify that design criteria are satisfied and to help establish thermal operating limits with acceptable margins of safety during normal reactor operation and anticipated operational occurrences (AOOs). The design criteria that are applicable to the ATRIUM 11 fuel design are described in Reference 1. To the extent possible, these analyses are performed on a generic fuel design basis. However, due to reactor and cycle operating differences, many of the analyses supporting these thermal-hydraulic operating limits are performed on a plant- and cycle-specific basis and are documented in plant- and cycle-specific reports.

The thermal-hydraulic design criteria are summarized below:

- **Hydraulic compatibility (Reference 1, Section 4.1.1).** The hydraulic flow resistance of the reload fuel assemblies shall be sufficiently similar to the existing fuel in the reactor such that there is no significant impact on total core flow or the flow distribution among assemblies in the core. This criterion evaluation is addressed in Sections 3.1 and 3.2.
- **Thermal margin performance (Reference 1, Section 4.1.2).** Fuel assembly geometry, including spacer design and rod-to-rod local power peaking, should minimize the likelihood of boiling transition during normal reactor operation as well as during AOOs. The fuel design should fall within the bounds of the applicable empirically based boiling transition correlation approved for Framatome reload fuel. Within other applicable mechanical, nuclear, and fuel performance constraints, the fuel design should achieve good thermal margin performance. The thermal-hydraulic design impact on steady-state thermal margin performance is addressed in Section 3.3. Additional thermal margin performance evaluations dependent on the cycle-specific design are addressed in the reload licensing report.
- **Fuel centerline temperature (Reference 1, Section 4.1.3).** Fuel design and operation shall be such that fuel centerline melting is not projected for normal operation and AOOs. This criterion evaluation is addressed in the fuel rod thermal and mechanical design report.
- **Rod bow (Reference 1, Section 4.1.4).** The anticipated magnitude of fuel rod bowing under irradiation shall be accounted for in establishing thermal margin requirements. This criterion evaluation is addressed in Section 3.4.
- **Bypass flow (Reference 1, Section 4.1.5).** The bypass flow characteristics of the reload fuel assemblies shall not differ significantly from the existing fuel in order to provide adequate flow in the bypass region. This criterion evaluation is addressed in Section 3.5.
- **Stability (Reference 1, Section 4.1.6).** Reactors fueled with new fuel designs must be stable in the power and flow operating region. The stability performance of new fuel designs will be equivalent to, or better than, existing (approved) Framatome fuel

designs. This criterion evaluation is addressed in Section 3.6. Additional core stability evaluations dependent on the cycle-specific design are addressed in the reload licensing report.

- **Loss-of-coolant accident (LOCA) analysis (Reference 1, Section 4.2).** LOCAs are analyzed in accordance with Appendix K modeling requirements using NRC-approved models. The criteria are defined in 10 CFR 50.46. LOCA analysis results are presented in the break spectrum and MAPLHGR report.
- **Control rod drop accident (CRDA) analysis (Reference 1, Section 4.3).** The results from the CRDA analysis must meet the criteria of Regulatory Guide 1.236 (Reference 3). This criterion evaluation is addressed in the reload licensing report.
- **ASME overpressurization analysis (Reference 1, Section 4.4).** ASME pressure vessel code requirements must be satisfied. This criterion evaluation is addressed in the reload licensing report.
- **Seismic/LOCA liftoff (Reference 1, Section 4.5).** Under accident conditions, the assembly must remain engaged in the fuel support. This criterion evaluation is addressed in the mechanical design report.

A summary of the thermal-hydraulic design evaluations is given in Table 3.1.

3.1 *Hydraulic Characterization*

Basic geometric parameters for the ATRIUM 11 and ATRIUM 10XM fuel designs are summarized in Table 3.2. Component loss coefficients for the fuels mentioned are based on tests and are presented in Table 3.3. These loss coefficients include modifications to the test data reduction process **[**

]. The bare rod friction, ULTRAFLOW spacer, UTP and LTP losses for the ATRIUM 11 and ATRIUM 10XM fuel designs are based on tests performed at Framatome's Portable Hydraulic Test Facility. **[**

]

The primary resistance for the leakage flow through the LTP flow holes is **[**

]. The resistances for the leakage paths are shown in Table 3.3.

3.2 *Hydraulic Compatibility*

The thermal-hydraulic analyses were performed in accordance with the Framatome thermal-hydraulic methodology for BWRs. The methodology and constitutive relationships used by Framatome for the calculation of pressure drop in BWR fuel assemblies are presented in Reference 4 and are implemented in the XCOBRA code. The XCOBRA code predicts steady-state thermal-hydraulic performance of the fuel assemblies of BWR cores at various operating conditions and power distributions. XCOBRA received NRC approval in Reference 5. The NRC reviewed the information provided in Reference 6 regarding inclusion of water rod models in XCOBRA and accepted the inclusion in Reference 7.

Hydraulic compatibility, as it relates to the relative performance of the ATRIUM 11 and coresident ATRIUM 10XM fuel designs, has been evaluated. Detailed analyses were performed for full cores of each fuel design presented herein. Analyses for mixed cores with ATRIUM 11 and ATRIUM 10XM fuel were also performed to demonstrate the thermal-hydraulic design criteria are satisfied for transition core configurations.

The hydraulic compatibility analysis is based on **[**

]

Table 3.4 summarizes the input conditions for the analyses. These conditions reflect two of the state points considered in the analyses: 100% power/100% flow and 59.2% power/43.3% flow, which is the core flow at the minimum pump speed on the MELLLA line. Table 3.4 also defines the core loading for the transition core configurations. Input for other core configurations is similar in that core operating conditions remain the same and the same axial power distribution is used. Evaluations were made with the bottom-, middle-, and top-peaked axial power distributions presented in Figure 3.1. Results presented in this report are for the middle-peaked power distribution. Results for bottom- and top-peaked axial power distributions show similar trends.

Table 3.5 and Table 3.6 provide a summary of calculated thermal-hydraulic results using the first transition core configuration. Table 3.7 and Table 3.8 provide a summary of results for all core configurations evaluated. Core average results and the differences between the ATRIUM 11 and ATRIUM 10XM results at rated power are within the range which is considered compatible. Similar agreement occurs at lower power levels. As shown in Table 3.5, **[**

]. Table 3.6 shows that **[**

]. Differences in assembly flow between the ATRIUM 11 and ATRIUM 10XM fuel designs as a function of assembly power level are shown in Figure 3.2 and Figure 3.3.

Core pressure drop and core bypass flow fraction are also provided for the configurations evaluated. Based on the reported changes in pressure drop and assembly flow caused by the introduction of ATRIUM 11, the ATRIUM 11 design is considered hydraulically compatible with the coresident fuel design since the thermal-hydraulic design criteria are satisfied.

3.3 *Thermal Margin Performance*

Relative thermal margin analyses were performed in accordance with the thermal-hydraulic methodology for Framatome's XCOBRA code. The calculation of the fuel assembly critical power ratio (CPR) (thermal margin performance) is established by means of an empirical correlation based on results of boiling transition test programs. The CPR methodology is the approach used by Framatome to determine the margin to thermal limits for BWRs.

CPR values for ATRIUM 11 are calculated with the ACE/ATRIUM 11 critical power correlation (Reference 8) while the CPR values for ATRIUM 10XM are calculated with the ACE/ATRIUM 10XM critical power correlation (Reference 9). Assembly design features are incorporated in the CPR calculation through the K-factor term in the ACE correlations. The K-factors are based on the local power peaking for the nuclear design and on additive constants determined in accordance with approved procedures. The local peaking factors are a function of assembly void fraction and exposure.

For the compatibility evaluation, steady-state analyses evaluated ATRIUM 11 and ATRIUM 10XM assemblies with radial peaking factors (RPFs) between **[**

]. Table 3.5 and Table 3.6 show CPR results of the ATRIUM 11 and ATRIUM 10XM fuel. Table 3.7 and Table 3.8 show similar comparisons of CPR and assembly flow for the various core configurations evaluated. Analysis results indicate ATRIUM 11 fuel will not cause thermal margin problems for the coresident ATRIUM 10XM fuel design.

3.4 *Rod Bow*

The bases for rod bow are discussed in the mechanical design report. Rod bow magnitude is determined during the fuel-specific mechanical design analyses and confirmed on a cycle-specific basis.

[

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3.5 *Bypass Flow*

Total core bypass flow is defined as leakage flow through the LTP flow holes, channel seal, core support plate, and LTP-fuel support interface. Table 3.7 shows that total core bypass flow (excluding water rod flow) fraction at rated conditions is **[** **]** of rated core flow for the core configurations presented (middle-peaked power shape). In summary, adequate bypass flow will be available with the introduction of the ATRIUM 11 fuel design and applicable design criteria are met.

3.6 *Stability*

Each new fuel design is analyzed to demonstrate that the stability performance is equivalent to or better than an existing Framatome fuel design. The stability performance is a function of the core power, core flow, core power distribution, and to a lesser extent, the fuel design.

[

] A comparative stability

analysis was performed with the NRC-approved STAIF code (Reference 10). The study shows that the ATRIUM 11 fuel design has decay ratios equivalent to or better than other Framatome fuel designs.

As stated above, the stability performance of a core is strongly dependent on the core power, core flow, and power distribution in the core. Therefore, core stability is evaluated on a cycle-specific basis and addressed in the reload licensing report.

Table 3.1 Design Evaluation of Thermal and Hydraulic Criteria for the ATRIUM 11 Fuel Assembly

Report Section	Description	Criteria	Results or Disposition
	Thermal and Hydraulic Criteria		
3.1 / 3.2	Hydraulic compatibility	Hydraulic flow resistance shall be sufficiently similar to existing fuel such that there is no significant impact on total core flow or flow distribution among assemblies.	Verified on a plant-specific basis. ATRIUM 11 demonstrated to be compatible with ATRIUM 10XM fuel. [ ]
3.3	Thermal margin performance	Fuel design shall be within the limits of applicability of an approved CHF correlation.	ACE/ATRIUM 11 critical power correlation is applied to the ATRIUM 11 fuel. ACE/ATRIUM 10XM critical power correlation is applied to the ATRIUM 10XM fuel.
		≤0.1% of rods in boiling transition.	Verified on cycle-specific basis for Chapter 14 analyses.
	Fuel centerline temperature	No centerline melting.	Plant- and fuel-specific analyses are performed.
3.4	Rod bow	Rod bow must be accounted for in establishing thermal margins.	The lateral displacement of the fuel rods due to fuel rod bowing is not of sufficient magnitude to impact thermal margins. Verified on a cycle-specific basis.
3.5	Bypass flow	Bypass flow characteristics shall be similar among assemblies to provide adequate bypass flow.	Verified on a plant-specific basis. Analysis results demonstrate that adequate bypass flow is provided.

Table 3.1 Design Evaluation of Thermal and Hydraulic Criteria for the ATRIUM 11 Fuel Assembly *(Continued)*

Report Section	Description	Criteria	Results or Disposition
	Thermal and Hydraulic Criteria *(Continued)*		
3.6	Stability	New fuel designs are stable in the approved power and flow operating region, and stability performance will be equivalent to (or better than) existing (approved) Framatome fuel designs.	ATRIUM 11 channel and core decay ratios have been demonstrated to be equivalent to or better than other approved Framatome fuel designs. Core stability behavior is evaluated on a cycle-specific basis.
	LOCA analysis	LOCA analyzed in accordance with Appendix K modeling requirements. Criteria defined in 10 CFR 50.46.	Plant- and fuel-specific analysis is performed with Appendix K LOCA models and verified with cycle-specific calculations.
	CRDA analysis	Criteria defined in Regulatory Guide 1.236.	Cycle-specific analysis is performed.
	ASME over-pressurization analysis	ASME pressure vessel code requirements shall be satisfied.	Cycle-specific analysis is performed.
	Seismic/LOCA liftoff	Assembly remains engaged in fuel support.	Plant- and fuel-specific analyses are performed.

Table 3.2 Comparative Description for Monticello ATRIUM 11 and ATRIUM 10XM Fuel Types

Fuel Parameter	ATRIUM 10XM	ATRIUM 11
Number of fuel rods		
Full-length fuel rods	79	92
PLFRs	12	
Short PLFRs		12
Long PLFRs		8
Fuel clad OD, in	0.4047	0.3701
Number of spacers	9	9
Active fuel length, in		
Full-length fuel rods	145.24	145.24
PLFRs	75.0	
Short PLFRs		55.87
Long PLFRs		88.03
Hydraulic resistance characteristics	Table 3.3	Table 3.3
Number of water rods	1	1
Water rod OD, in	1.378*	1.299*

* Square water channel outer width.

Table 3.3 Hydraulic Characterization Comparison for Monticello ATRIUM 11 and ATRIUM 10XM Fuel Types

[

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[

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Table 3.4 Monticello
Thermal-Hydraulic Design Conditions

Reactor Conditions	100%P / 100%F	59.2%P / 43.3%F
Core power level, MWt	2004.0	1186.4
Core exit pressure, psia	1033.0	963.0
Core inlet enthalpy, Btu/lbm	522.6	498.1
Total core coolant flow, Mlbm/hr	57.6	24.9
Axial power shape	Middle-peaked (Figure 3.1)	Middle-peaked (Figure 3.1)

	Number of Assemblies	
	Central Region	Peripheral Region
First Transition Core Loading		
[		]
[		]
Second Transition Core Loading		
[		]
[		]

Table 3.5 Monticello
First Transition Core Thermal-Hydraulic Results at
Rated Conditions (100%P / 100%F)

[

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[

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Table 3.6 Monticello
First Transition Core Thermal-Hydraulic Results at
Off-Rated Conditions (59.2%P / 43.3%F)

[

]

[

]

Table 3.7 Monticello Thermal-Hydraulic Results at Rated Conditions (100%P / 100%F) for Transition to ATRIUM 11 Fuel

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Table 3.8 Monticello Thermal-Hydraulic Results at Off-Rated Conditions (59.2%P / 43.3%F) for Transition to ATRIUM 11 Fuel

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Figure 3.1 Axial Power Shapes

[

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Figure 3.2 First Transition Core:
Hydraulic Demand Curves
100%P / 100%F

[

]

Figure 3.3 First Transition Core:
Hydraulic Demand Curves
59.2%P / 43.3%F

4.0 REFERENCES

1. ANF-89-98(P)(A) Revision 1 and Supplement 1*, Generic Mechanical Design Criteria for BWR Fuel Designs*, Advanced Nuclear Fuels Corporation, May 1995.

2. XN-NF-80-19(P)(A) Volume 4 Revision 1, *Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads*, Exxon Nuclear Company, June 1986.

3. RG 1.236, *Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents*, June 2020. (NRC ADAMS ML20055F490).

4. XN-NF-79-59(P)(A), *Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies*, Exxon Nuclear Company, November 1983.

5. XN-NF-80-19(P)(A) Volume 3 Revision 2, *Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description*, Exxon Nuclear Company, January 1987.

6. Letter, R.A. Copeland (ANF) to R.C. Jones (USNRC), “Explicit Modeling of BWR Water Rod in XCOBRA,” RAC:002:90, January 9, 1990.

7. Letter, R.C. Jones (USNRC) to R.A. Copeland (ANF), no subject (regarding XCOBRA water rod model), February 1, 1990.

8. ANP-10335P-A Revision 0, *ACE/ATRIUM 11 Critical Power Correlation*, Framatome, May 2018.

9. ANP-10298P-A Revision 1, *ACE/ATRIUM 10XM Critical Power Correlation*, AREVA Inc., March 2014.

10. EMF-CC-074(P)(A) Volume 1, *STAIF – A Computer Program for BWR Stability Analysis in the Frequency Domain;* and Volume 2, *STAIF – A Computer Program for BWR Stability Analysis in the Frequency Domain – Code Qualification Report*, Siemens Power Corporation, July 1994.

ENCLOSURE

ATTACHMENT 5b

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST
APPLICATION TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

AFFIDAVIT FOR

ANP-3893P REPORT, REVISION 0

MONTICELLO THERMAL-HYDRAULIC DESIGN REPORT
FOR ATRIUM 11 FUEL ASSEMBLIES

MAY 2021

(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-3893P Revision 0, "Monticello Thermal-Hydraulic Design Report for ATRIUM 11 Fuel Assemblies," dated May 2021 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 20, 2021

Alan B. Meginnis

ENCLOSURE

ATTACHMENT 6a

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST
APPLICATION TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

ANP-3903NP REPORT, REVISION 0

ATRIUM 11 FUEL ROD THERMAL-MECHANICAL EVALUATION FOR MONTICELLO LAR

LICENSING REPORT

MARCH 2021

(24 pages follow)

ATRIUM 11 Fuel Rod Thermal-Mechanical Evaluation for Monticello LAR

ANP-3903NP
Revision 0

Licensing Report

March 2021

Nature of Changes

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

Contents

List of Tables

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Nomenclature

Acronym	Definition
3GFG	3rd generation FUELGUARD
AOO	anticipated operational occurrences
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
BOL	beginning of life
BWR	boiling water reactor
CRWE	control rod withdrawal error
CUF	cumulative usage factor
EOL	end of life
EFW	extended flow window
FDL	fuel design limit
ID	inside diameter
LAR	License Amendment Request
LHGR	linear heat generation rate
LTP	lower tie plate
MWd/kgU	megawatt days per kilogram of initial uranium
NRC	Nuclear Regulatory Commission, U. S.
OD	outside diameter
PCI	pellet-to-cladding-interaction
PLFR	part length fuel rod
ppm	parts per million
SRA	stress relieved annealed
S-N	stress amplitude versus number of cycles
UTL	upper tolerance limit

1.0 INTRODUCTION

This document reports the results of thermal-mechanical analyses for the performance of ATRIUM 11 fuel assemblies inserted into to an equilibrium cycle for the Monticello unit and demonstrates that the design criteria relevant to thermal-mechanical performance are satisfied. This report is intended to support a License Amendment Request (LAR) for the approval to use the Framatome advanced analysis methods that will be deployed coincident with the implementation of the ATRIUM 11 fuel assembly design. These analyses include the use of chromia additive in the urania portions of the fuel and operation in the Extended Flow Window (EFW) operation domain. Both the design criteria and the analysis methodology have been approved by the U. S. NRC (NRC).

The analysis results are evaluated according to the generic fuel rod thermal and mechanical design criteria contained in ANF-89-98(P)(A) Revision 1 and Supplement 1 (Reference 1) along with design criteria provided in the RODEX4 fuel rod thermal-mechanical topical report (Reference 2)*. Approved methodology for the inclusion of chromia additive in the fuel pellets is also used (Reference 3).

The RODEX4 fuel rod thermal-mechanical analysis code is used to analyze the fuel rod for fuel centerline temperature, cladding strain, rod internal pressure, cladding collapse, cladding fatigue and external oxidation. The code and application methodology are described in the RODEX4 topical report (Reference 2). The cladding steady-state stress and plenum spring design methodology are summarized in Reference 1.

The following sections describe the fuel rod design, design criteria and methodology with reference to the source topical reports. Results from the analyses are summarized for comparison to the design criteria.

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* (N.B., the cladding external oxidation limit from that topical report of **[ ]** was reduced to **[ ]** when the RODEX4 methodology was approved for application to the Monticello unit (Reference 4)).

2.0 SUMMARY AND CONCLUSIONS

Key results are compared against each design criterion in Table 2-1. Results are presented for the limiting cases. Additional RODEX4 results are given in Section 3.0.

The analyses support a maximum fuel rod discharge exposure of 62 MWd/kgU.

Fuel rod criteria applicable to the design are summarized in Section 3.0. Analyses show the criteria are satisfied when the fuel is operated at or below the LHGR (linear heat generation rate) limit (Fuel Design Limit – FDL) presented in Figure 2-1.

Table 2-1 Summary of Fuel Rod Design Evaluation Results

Criteria Section*	Description	Criteria	Result, Margin† or Comment
3.2	**Fuel Rod Criteria**		
3.2.1	Internal hydriding	[	]
(3.1.1)	Cladding collapse	[	]
(3.1.2)	Overheating of fuel pellets	No fuel melting margin to fuel melt > 0. °C	[ ]
3.2.5	Stress and strain limits		
(3.1.1) (3.1.2)	Pellet-cladding interaction	[	]
3.2.5.2	Cladding steady-state stresses	[	]
3.3	**Fuel System Criteria**		
(3.1.1)	Fatigue	[	]
(3.1.1)‡	Oxidation, hydriding, and crud buildup	[	]
(3.1.1) (3.1.2)	Rod internal pressure	[	]
3.3.9	Fuel rod plenum spring (fuel handling)	Plenum spring to [	]

* Numbers in the column refer to paragraph sections in the generic design criteria document, ANF-89-98(P)(A) Revision 1 and Supplement 1 (Reference 1). A number in parentheses is the paragraph section in the RODEX4 fuel rod topical report (Reference 2).

† Margin is defined as (limit – result).

‡ The cladding external oxidation limit is restricted to [ ] by Reference 4.

[

]

Figure 2-1 LHGR Limit (Normal Operation)

3.0 FUEL ROD DESIGN EVALUATION

Summaries of the design criteria and methodology are provided in this section along with analysis results in comparison to criteria. Both the fuel rod criteria and fuel system criteria as directly related to the fuel rod analyses are covered.

The fuel rod analyses cover normal operating conditions and AOOs (anticipated operational occurrences). For the slow transients at rated operating conditions, the fuel centerline temperature analysis (overheating of fuel) and cladding strain analysis are assessed.

Other fuel rod-related topics on overheating of cladding, cladding rupture, fuel rod mechanical fracturing, rod bow, axial irradiation growth, cladding embrittlement, violent expulsion of fuel and fuel ballooning are evaluated as part of the respective fuel assembly structural analysis, thermal hydraulic analyses, or LOCA analyses, and are reported elsewhere. The evaluation of fast transients and transients at off-rated conditions also are reported separately from this report.

3.1 *Fuel Rod Design*

The ATRIUM 11 fuel rod is conventional in design configuration and very similar to past designs such as the ATRIUM 10XM fuel rods.

[

] plenum spring on the upper end of the fuel column assists in maintaining a compact fuel column during shipment and initial reactor operation.

There are two Part-length Fuel Rod (PLFR) designs incorporated in the fuel assembly. The longer is **[** **]** long, while the shorter is **[** **]** long. **[**

]

[

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As on previous ATRIUM fuel designs that incorporated the 3rd generation FUELGUARD (3GFG) Lower Tie Plate (LTP), the PLFR's have a **[**

]

Table 3-1 lists the main parameters for the fuel rod and components.

3.2 *RODEX4 and Statistical Methodology Summary*

RODEX4 evaluates the thermal-mechanical response of the fuel rod surrounded by coolant. The fuel rod model considers the fuel column, gap region, cladding, gas plena and the fill gas and released fission gases. The fuel rod is divided into axial and radial regions with conditions computed for each region. The operational conditions are controlled by the **[**

]

The heat conduction in the fuel and clad is **[**

]

Mechanical processes include **[**

]

As part of the methodology, fuel rod power histories are generated **[**

]

Since RODEX4 is a best-estimate code, uncertainties are taken into account by a **[**

] Uncertainties taken into account in the analysis are summarized as:

- Power measurement and operational uncertainties – **[**

  ].
- Manufacturing uncertainties – **[**
  ]
- Model uncertainties – **[**

  ]

[

]

3.3 *Summary of Fuel Rod Design Evaluation*

Results from the analyses are listed in Table 3-2. Summaries of the methods and codes used in the evaluation are provided in the following paragraphs. The design criteria also are listed along with references to the sections of the design criteria topical reports (References 1 and 2).

The fuel rod thermal and mechanical design criteria are summarized as follows.

- **Internal Hydriding**. The fabrication limit [ ] to preclude cladding failure caused by internal sources of hydrogen (Section 3.2.1 of Reference 1).

- **Cladding Collapse**. Clad creep collapse shall be prevented. [ ] (Section 3.1.1 of Reference 2).

- **Overheating of Fuel Pellets**. The fuel pellet centerline temperature during anticipated transients shall remain below the melting temperature (Section 3.1.2 of Reference 2).

- **Stress and Strain Limits**. [ ] during normal operation and during anticipated transients (Sections 3.1.1 and 3.1.2 of Reference 2).

  Fuel rod cladding steady-state stresses are restricted to satisfy limits derived from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code (Section 3.2.5.2 of Reference 1).

- **Cladding Fatigue**. The fatigue cumulative usage factor for clad stresses during normal operation and design cyclic maneuvers shall be below [ ] (Section 3.1.1 of Reference 2).

- **Cladding Oxidation, Hydriding and Crud Buildup**. Section 3.1.1 of Reference 2 limits the maximum cladding oxidation to less than [ ] to prevent clad corrosion failure. The oxidation limit is further reduced to [ ] (Reference 4).

- **Rod Internal Pressure**. The rod internal pressure is limited [ ] to ensure that significant outward clad creep does not occur and unfavorable hydride reorientation on cooldown does not occur (Section 3.1.1 of Reference 2).

- **Plenum Spring Design (Fuel Handling)**. The rod plenum spring must maintain a force against the fuel column stack [ ] (Section 3.3.9 of Reference 1).

Cladding collapse, overheating of fuel, cladding transient strain, cladding cyclic fatigue, cladding oxidation, and rod pressure are evaluated [ ]. Cladding stress and the plenum spring are evaluated [ ]

3.3.1 Internal Hydriding

The absorption of hydrogen by the cladding can result in cladding failure due to reduced ductility and formation of hydride platelets. Careful moisture control during fuel fabrication reduces the potential for hydrogen absorption on the inside of the cladding. The fabrication limit **[** **]** is verified by quality control inspection during fuel manufacturing.

3.3.2 Cladding Collapse

Creep collapse of the cladding and the subsequent potential for fuel failure is avoided in the design by limiting the axial gap formation due to fuel densification subsequent to pellet-clad contact. The size of the axial gaps which may form due to densification following first pellet-clad contact shall be less than **[** **]**

The evaluation is performed using the RODEX4 code and methodology. RODEX4 takes into account the **[** **]**

Table 3-2 lists the results for an equilibrium cycle.

3.3.3 Overheating of Fuel Pellets

Fuel failure from the overheating of the fuel pellets is not allowed. The centerline temperature of the fuel pellets must remain below melting during normal operation and AOOs. The melting point of the fuel includes adjustments for **[** **]** Framatome establishes an LHGR limit to protect against fuel centerline melting during steady-state operation and during AOOs.

Fuel centerline temperature is evaluated using the RODEX4 code and methodology for both normal operating conditions and AOOs.

Table 3-2 lists the results for an equilibrium cycle.

3.3.4 Stress and Strain Limits

3.3.4.1 Pellet/Cladding Interaction

Cladding strain caused by transient-induced deformations of the cladding is calculated using the RODEX4 code and methodology. [

] The strain limit is 1%.

Table 3-2 lists the results for an equilibrium cycle.

3.3.4.2 Cladding Stress

Cladding stresses are calculated using solid mechanics elasticity solutions and finite element methods. The stresses are conservatively calculated for the individual loadings and are categorized as follows:

Category	Membrane	Bending
Primary	[	]
Secondary	[	]

Stresses are calculated at the cladding outer and inner diameter in the three principal directions for both beginning of life (BOL) and end of life (EOL) conditions. At EOL, the stresses due to mechanical bow and contact stress are decreased due to irradiation relaxation. The separate stress components are then combined, and the stress intensities for each category are compared to their respective limits.

The cladding-to-end cap weld stresses are evaluated for loadings from differential pressure, differential thermal expansion, rod weight, and plenum spring force.

The design limits are derived from the ASME (American Society of Mechanical Engineers) Boiler and Pressure Vessel (B&PV) Code Section III (Reference 5) and the minimum specified material properties.

Table 3-3 lists the results in comparison to the limits for Beginning-of-Life (BOL) Hot conditions and End-of-Life (EOL) at both Hot and Cold conditions.

3.3.5 Fuel Densification and Swelling

Fuel densification and swelling are limited by the design criteria for fuel temperature, cladding strain, cladding collapse, and rod internal pressure criteria. Although there are no explicit criteria for fuel densification and swelling, the effect of these phenomena are included in the RODEX4 code and methodology.

3.3.6 Fatigue

Fuel rod cladding fatigue is calculated using the RODEX4 code and methodology. **[**

]. The CUF (cumulative usage factor) is summed for each of the axial regions of the fuel rod using Miner's rule. The axial region with the highest CUF is used in the subsequent **[**

] The maximum CUF for the cladding must remain below **[ ]** to satisfy the design criterion. Table 3-2 lists the results for an equilibrium cycle.

3.3.7 Oxidation, Hydriding, and Crud Buildup

Cladding external oxidation is calculated using the RODEX4 code and methodology. The corrosion model includes an enhancement factor that is derived from poolside measurement data to obtain a fit of the expected oxide thickness. An uncertainty value for the model enhancement factor also is determined from the data. The model uncertainty is included as part of the **[ ]**

[

]

In the event abnormal crud is observed at a plant, a specific analysis is required to address the higher crud level. An abnormal level of crud is defined by a formation that increases the calculated fuel average temperature by 25°C above the design basis calculation. The formation of crud is not calculated within RODEX4. Instead, an upper bound of expected crud based on plant observations is input by the use of the crud heat transfer coefficient. The corrosion model also takes into consideration the effect of the higher thermal resistance from the crud on the corrosion rate. A higher corrosion rate is therefore included as part of the abnormal crud evaluation. A similar specific analysis is required if an abnormal corrosion layer is observed instead of crud.

In the case of the Monticello unit, no additional crud is taken into account in the calculations because an abnormal crud or corrosion layer (beyond the design basis) has not been observed at the Monticello unit.

[

]

Currently, [

]

The oxide limit is evaluated such that greater than [

]

Table 3-2 lists the results for an equilibrium cycle.

3.3.8 Rod Internal Pressure

Fuel rod internal pressure is calculated using the RODEX4 code and methodology. The maximum rod pressure is calculated under steady-state conditions and also takes into account slow transients. Rod internal pressure is limited to **[** **]** The expected upper bound of rod pressure **[** **]** is calculated for comparison to the limit.

Table 3-2 lists the results for an equilibrium cycle.

3.3.9 Plenum Spring Design (Fuel Assembly Handling)

The plenum spring must maintain a force against the fuel column to prevent **[** **]** This is accomplished by designing and verifying the spring force in relation to the fuel column weight. The plenum spring is designed such that the **[** **]**

Table 3-1 Key Fuel Rod Design Parameters, ATRIUM 11 for Monticello LAR

[

]

* This length does not include the extension at the top of the full-length fuel rod that ensures engagement with the upper tie plate.

† The theoretical density of enriched UO_2-Cr is 10.94 g/cm^3, while that for UO_2-Gd_2O_3 is 10.96 g/cm^3.

Table 3-1 Key Fuel Rod Design Parameters, ATRIUM 11 for Monticello LAR (cont'd)

[

]

Table 3-2 RODEX4 Fuel Rod Results Equilibrium Cycle*

[

]

* Note that the results are provided up to fuel assembly discharge.
† Margin is defined as (limit – result).

Table 3-3 Cladding and Cladding-End Cap Steady-State Stresses

Description, Stress Category	Criteria	Result		
		BOL Cold	BOL Hot	EOL Hot
Cladding stress				
P_m (primary membrane stress)	**[**			**]**
$P_m + P_b$ (primary membrane + bending)	**[**			**]**
P + Q (primary + secondary)	**[**			**]**
Cladding-End Cap stress				
$P_m + P_b$	**[**			**]**

4.0 REFERENCES

1. ANF-89-98(P)(A) Revision 1 and Supplement 1, *Generic Mechanical Design Criteria for BWR Fuel Designs*, Advanced Nuclear Fuels Corporation, May 1995.

2. BAW-10247PA Revision 0, *Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors*, AREVA NP Inc., February 2008.

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

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ENCLOSURE

ATTACHMENT 6b

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST
APPLICATION TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

AFFIDAVIT FOR

ANP-3903P REPORT, REVISION 0

ATRIUM 11 FUEL ROD THERMAL-MECHANICAL EVALUATION FOR MONTICELLO LAR

LICENSING REPORT

MARCH 2021

(3 pages follow)

AFFIDAVIT

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-3903P Revision 0, "ATRIUM 11 Fuel Rod Thermal-Mechanical Evaluation for Monticello LAR," dated March 2021 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: March 16, 2021

Alan B. Meginnis

ENCLOSURE

ATTACHMENT 7a

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST
APPLICATION TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

ANP-3877NP REPORT, REVISION 0

MONTICELLO ATRIUM 11 EQUILIBRIUM FUEL

NUCLEAR FUEL DESIGN REPORT

OCTOBER 2020

(240 pages follow)

framatome

Monticello ATRIUM 11 Equilibrium Fuel

ANP-3877NP
Revision 0

Nuclear Fuel Design Report

October 2020

Nature of Changes

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

Contents

List of Tables

List of Figures

Nomenclature

Acronym	**Definition**
BOL	beginning of life
BWR	boiling water reactor
EVC	plenum region in a fuel pin modeled as an evacuated section
kg/MTU	kilograms per metric ton of initial uranium
LHGR	linear heat generation rate
LPF	local peaking factor
MCPR	minimum critical power ratio
MWd/MTU	megawatt days per metric ton of initial uranium
NRC	(United States) Nuclear Regulatory Commission

1.0 INTRODUCTION

This report provides results from the neutronic design analyses performed by Framatome Inc. for the Monticello ATRIUM 11 equilibrium design. The methodology, design criteria, and general assumptions used in the fuel design are also provided.

Applicable neutronic design criteria are provided in the approved topical report ANF-89-98(P)(A) Revision 1 and Supplement 1 (Reference 2). Neutronic design analysis methodology used to determine conformance to design criteria has been reviewed and approved by the NRC in the topical report EMF-2158(P)(A) (Reference 3).

The fuel design general assumptions include **[**

] . The neutronic component of this fuel design includes axially-varying enrichment and Gadolinia and natural UO_2 blankets at the top and bottom of the assembly. Mechanical design parameters for the fuel design are from Reference 1 and are shown in Table 2.1. Other pertinent fuel and reactor core design information is given in Section 2.0 and in Appendices A through D.

2.0 NEUTRONIC DESIGN

The results of the neutronic design analyses are presented in this section. The fuel was designed to meet applicable design criteria, as well as reactivity and control requirements. Applicable neutronic design criteria outlined in Reference 2 are summarized below:

- **Power Distribution.** The local power distribution in the fuel assembly combined with the core power distribution shall result in Linear Heat Generation Rate (LHGR) and Minimum Critical Power Ratio (MCPR) values that are within the limits established for each fuel design.
- **Kinetics Parameters.** The moderator void reactivity coefficient due to boiling in the active channels and the Doppler fuel temperature reactivity coefficient shall be negative. The negative void and Doppler reactivity coefficients ensure a negative power coefficient during reactor operation. (Calculation results show that the assembly average Doppler and void reactivity coefficients remain negative for the life of the assembly. These results demonstrate that the Reference 2 Section 5 kinetics criteria are met on a bundle average basis.)
- **Control Blade Reactivity.** The design of the fuel assembly and the reactor core loading shall be such that the technical specification shutdown margin requirement is met for all reactor conditions.

2.1 *Neutronic Design Description*

The neutronic design parameters for fabrication batch MON-EQ are presented in Table 2.1. The key nuclear design characteristics are summarized below:

- The fuel assembly contains **[ ]**.
- Each fuel assembly has top and bottom natural uranium blankets.
- The enrichments are designed to yield a local power distribution which results in a balanced design relative to MCPR, LHGR, and other reactor operating requirements, e.g., power peaking.
- Gadolinia (Gd_2O_3 blended with UO_2) rods are designed to control assembly reactivity in order to meet reactivity control requirements in the reactor, e.g. cold shutdown margin.
- Fuel assembly designs utilize axially varying enrichment and/or gadolinia. The axial distributions of the lattices in the assemblies are shown in Figures 2.1, 2.2, and 2.3. The fuel rod distribution and axial descriptions are presented in Figures 2.4 through 2.8. The enrichment and gadolinia distribution maps for each of the assembly lattices are displayed in Appendix D.
- The fuel assembly incorporates an advanced fuel channel which improves uranium utilization. For D-lattice plants, the fuel assembly is offset 40 mils toward the control blade.

2.2 *Lattice Control Blade Worths and Kinetics Parameters*

Beginning of life (BOL) lattice reactivities (k_∞) have been calculated for moderator and fuel conditions ranging from cold to hot operating conditions. From these reactivities, BOL control blade worths and kinetics parameters have been determined based on Original Equipment Blade (OEB), Marathon Upper (MMZ), and Marathon Lower (MLZ) control blades (Reference 4).

Kinetics parameters are calculated for fuel temperature (Doppler), moderator void, and moderator temperature. **[**

] The results of these calculations are presented in Tables 2.2 through 2.89.

2.3 *Enriched Lattice Uncontrolled Reactivities and Isotopic Data*

The enriched lattice exposure-dependent uncontrolled reactivities **[** **]** are presented graphically in Appendix A, and in tabular format in Appendix B. The enriched lattice exposure-dependent isotopic data **[** **]** are presented in Appendix C.

2.4 *Criticality Compliance*

The spent fuel storage and new fuel vault criticality compliance is not addressed in this report because the fuel design herein is meant for demonstration of methods, but the criticality compliance will be explicitly addressed in the Monticello ATRIUM 11 transition.

Table 2.1 Neutronic Design Parameters

Table 2.1 Neutronic Design Parameters *(Continued)*

Table 2.1 Neutronic Design Parameters *(Continued)*

Parameter	Design Value
Control Blade Data for OEB	
Total span, inch	General Electric Proprietary
Total support span, inch	“
Total thickness, inch	“
Total face-to-face internal dimension, inch	“
B_4C rod absorber Number of rods Diameter of rod, inch Diameter of sheath, inch Theoretical density B_4C, % B_4C zone span, inch	“
Control Blade Data for MMZ	
Total span, inch	“
Total support span, inch	“
Total thickness, inch	“
Total face-to-face internal dimension, inch	“
Modeled empty air rods (wing absorber zone 1) Number of rods Diameter of air, inch Diameter of sheath, inch Air zone span, inch	“
B_4C rod absorber (wing absorber zone 2) Number of rods Diameter of rod, inch Diameter of sheath, inch Theoretical density B_4C, % B_4C zone span, inch	“
Hafnium rod absorber (wing absorber zone 3) Number of rods Diameter of rod, inch Diameter of sheath, inch Hafnium rod zone span, inch	“

Table 2.1 Neutronic Design Parameters *(Continued)*

Parameter	Design Value
Control Blade Data for MLZ	
Total span, inch	General Electric Proprietary
Total support span, inch	“
Total thickness, inch	“
Total face-to-face internal dimension, inch	“
Modeled empty air rods (wing absorber zone 1) Number of rods Diameter of air, inch Diameter of sheath, inch Air zone span, inch	“
B_4C rod absorber (wing absorber zone 2) Number of rods Number of smeared empty locations Diameter of rod, inch Diameter of sheath, inch Theoretical density B_4C, % B_4C zone span, inch	“
Hafnium rod absorber (wing absorber zone 3) Number of rods Diameter of rod, inch Diameter of sheath, inch Hafnium rod zone span, inch	“

Table 2.1 Neutronic Design Parameters *(Continued)*

Parameter	**Design Value**
Core Data*	
Number of fuel assemblies in the core	484
Rated thermal power level, MWt	2004.0
Rated core flow, Mlbm/hr	57.6
Inlet subcooling, Btu/lbm	23.4
Dome pressure, psia	1025
[ ]	**[ ]**
[ ]	**[ ]**
[ ]	**[ ]**

* Some values are representative of rated conditions and may vary depending on the core statepoint.

Table 2.2 Lattice [ ] Control Blade Worths at BOL for Control Blade Type OEB

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Table 2.3 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MMZ

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Table 2.4 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MLZ

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Table 2.5 Lattice [ ] Kinetics Parameters at BOL

[ ]

Table 2.6 Lattice [ ] Control Blade Worths at BOL for Control Blade Type OEB

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Table 2.7 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MMZ

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Table 2.8 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MLZ

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Table 2.9 Lattice [ ] Kinetics Parameters at BOL

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Table 2.10 Lattice [ ] Control Blade Worths at BOL for Control Blade Type OEB

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Table 2.11 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MMZ

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Table 2.12 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MLZ

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Table 2.13 Lattice [ ] Kinetics Parameters at BOL

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Table 2.14 Lattice [ ] Control Blade Worths at BOL for Control Blade Type OEB

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Table 2.15 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MMZ

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Table 2.16 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MLZ

[ ]

Table 2.17 Lattice [ ] Kinetics Parameters at BOL

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Table 2.18 Lattice [ ] Control Blade Worths at BOL for Control Blade Type OEB

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Table 2.19 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MMZ

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Table 2.20 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MLZ

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Table 2.21 Lattice [ ] Kinetics Parameters at BOL

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Table 2.22 Lattice [ ] Control Blade Worths at BOL for Control Blade Type OEB

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Table 2.23 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MMZ

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Table 2.24 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MLZ

[ ]

Table 2.25 Lattice [ ] Kinetics Parameters at BOL

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Table 2.26 Lattice [ ] Control Blade Worths at BOL for Control Blade Type OEB

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Table 2.27 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MMZ

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Table 2.28 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MLZ

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Table 2.29 Lattice [ ] Kinetics Parameters at BOL

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Table 2.30 Lattice [ ] Control Blade Worths at BOL for Control Blade Type OEB

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Table 2.31 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MMZ

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Table 2.32 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MLZ

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Table 2.33 Lattice [ ] Kinetics Parameters at BOL

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Table 2.34 Lattice [ ] Control Blade Worths at BOL for Control Blade Type OEB

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Table 2.35 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MMZ

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Table 2.36 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MLZ

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Table 2.37 Lattice [ ] Kinetics Parameters at BOL

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Table 2.38 Lattice [ ] Control Blade Worths at BOL for Control Blade Type OEB

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Table 2.39 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MMZ

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Table 2.40 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MLZ

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Table 2.41 Lattice [ ] Kinetics Parameters at BOL

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Table 2.42 Lattice [ ] Control Blade Worths at BOL for Control Blade Type OEB

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Table 2.43 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MMZ

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Table 2.44 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MLZ

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Table 2.45 Lattice [ ] Kinetics Parameters at BOL

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Table 2.46 Lattice [ ] Control Blade Worths at BOL for Control Blade Type OEB

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Table 2.47 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MMZ

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Table 2.48 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MLZ

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Table 2.49 Lattice [ ] Kinetics Parameters at BOL

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Table 2.50 Lattice [ ] Control Blade Worths at BOL for Control Blade Type OEB

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Table 2.51 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MMZ

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Table 2.52 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MLZ

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Table 2.53 Lattice [ ] Kinetics Parameters at BOL

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Table 2.54 Lattice [ ] Control Blade Worths at BOL for Control Blade Type OEB

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Table 2.55 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MMZ

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Table 2.56 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MLZ

Table 2.57 Lattice [ ] Kinetics Parameters at BOL

Table 2.58 Lattice [ ] Control Blade Worths at BOL for Control Blade Type OEB

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Table 2.59 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MMZ

Table 2.60 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MLZ

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Table 2.61 Lattice [ ] Kinetics Parameters at BOL

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Table 2.62 Lattice [ ] Control Blade Worths at BOL for Control Blade Type OEB

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Table 2.63 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MMZ

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Table 2.64 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MLZ

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Table 2.65 Lattice [ ] Kinetics Parameters at BOL

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Table 2.66 Lattice [ ] Control Blade Worths at BOL for Control Blade Type OEB

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Table 2.67 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MMZ

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Table 2.68 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MLZ

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Table 2.69 Lattice [ ] Kinetics Parameters at BOL

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Table 2.70 Lattice [ ] Control Blade Worths at BOL for Control Blade Type OEB

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Table 2.71 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MMZ

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Table 2.72 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MLZ

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Table 2.73 Lattice [ ] Kinetics Parameters at BOL

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Table 2.74 Lattice [ ] Control Blade Worths at BOL for Control Blade Type OEB

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Table 2.75 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MMZ

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Table 2.76 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MLZ

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Table 2.77 Lattice [ ] Kinetics Parameters at BOL

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Table 2.78 Lattice [ ] Control Blade Worths at BOL for Control Blade Type OEB

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Table 2.79 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MMZ

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Table 2.80 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MLZ

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Table 2.81 Lattice [ ] Kinetics Parameters at BOL

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Table 2.82 Lattice [ ] Control Blade Worths at BOL for Control Blade Type OEB

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Table 2.83 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MMZ

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Table 2.84 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MLZ

Table 2.85 Lattice [ ] Kinetics Parameters at BOL

Table 2.86 Lattice [ ] Control Blade Worths at BOL for Control Blade Type OEB

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Table 2.87 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MMZ

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Table 2.88 Lattice [ ] Control Blade Worths at BOL for Control Blade Type MLZ

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Table 2.89 Lattice [ ] Kinetics Parameters at BOL

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Figure 2.1 Assembly Type [ ]

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Figure 2.2 Assembly Type [ ]

Figure 2.3 Assembly Type [ ]

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Figure 2.4 [ ] Fuel Rod Distribution

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Figure 2.5 [ ] Fuel Rod Distribution

Figure 2.6 [ ] Fuel Rod Distribution

Figure 2.7 Fuel Rod Axial Description

Figure 2.8 Fuel Rod Axial Description

3.0 REFERENCES

1. FS1-0044655, Revision 2.0, Monticello Unit 1 Equilibrium Cycle Specific ATRIUM 11 LAR Fuel Assembly Mechanical Data for Core Engineering, September 2020.

2. ANF-89-98(P)(A) Revision 1 and Supplement 1, *Generic Mechanical Design Criteria for BWR Fuel Designs*, Advanced Nuclear Fuels Corporation, May 1995.

3. EMF-2158(P)(A), Revision 0, *Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2*, Siemens Power Corporation, October 1999.

4. FS1-0051706, Revision 1.0, Monticello ATRIUM 11 Equilibrium Cycle Cross-Section Library Generation and FUELRQ Uranium Requirements, September 2020.

Appendix A Enriched Lattice Hot Uncontrolled Reactivity and LPF Plots

[ ]

Figure A.1 [ ] Hot Uncontrolled k_∞

Figure A.2 [ ] Hot Uncontrolled LPF

Figure A.3 [ ] Hot Uncontrolled k$_\infty$

[ ]

Figure A.4 [ ] Hot Uncontrolled LPF

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Figure A.5 [ ] Hot Uncontrolled k_∞

Figure A.6 [ ] Hot Uncontrolled LPF

Figure A.7 [ ] Hot Uncontrolled k_∞

Figure A.8 [ ] Hot Uncontrolled LPF

Figure A.9 [ ] Hot Uncontrolled k_∞

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Figure A.10 [ ] Hot Uncontrolled LPF

[ ]

Figure A.11 [ ] Hot Uncontrolled k_∞

[ ]

Figure A.12 [ ] Hot Uncontrolled LPF

[ ]

Figure A.13 [ ] Hot Uncontrolled k_∞

Figure A.14 [ ] Hot Uncontrolled LPF

Figure A.15 [ ] Hot Uncontrolled k_{∞}

Figure A.16 [ ] Hot Uncontrolled LPF

Figure A.17 [ ] Hot Uncontrolled k_∞

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Figure A.18 [ ] Hot Uncontrolled LPF

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Figure A.19 [ ] Hot Uncontrolled k_∞

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Figure A.20 [ ] Hot Uncontrolled LPF

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Figure A.21 [ ] Hot Uncontrolled k$_\infty$

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Figure A.22 [ ] Hot Uncontrolled LPF

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Figure A.23 [ ] Hot Uncontrolled k_∞

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Figure A.24 [ ] Hot Uncontrolled LPF

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Figure A.25 [ ] Hot Uncontrolled k_∞

Figure A.26 [ ] Hot Uncontrolled LPF

Figure A.27 [ ] Hot Uncontrolled k_∞

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Figure A.28 [ ] Hot Uncontrolled LPF

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Figure A.29 [ ] Hot Uncontrolled k_∞

Figure A.30 [ ] Hot Uncontrolled LPF

Figure A.31 [ ] Hot Uncontrolled k_∞

Figure A.32 [ ] Hot Uncontrolled LPF

Figure A.33 [ ] Hot Uncontrolled k_∞

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Figure A.34 [ ] Hot Uncontrolled LPF

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Figure A.35 [ ] Hot Uncontrolled k$_\infty$

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Figure A.36 [ ] Hot Uncontrolled LPF

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Figure A.37 [ ] Hot Uncontrolled k_∞

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Figure A.38 [ ] Hot Uncontrolled LPF

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Appendix B Enriched Lattice Hot Uncontrolled Reactivity and LPF Tables

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Table B.1 [ ] Hot Uncontrolled k_∞

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Table B.2 [ ] Hot Uncontrolled LPF

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Table B.3 [ ] Hot Uncontrolled k_∞

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Table B.4 [ ] Hot Uncontrolled LPF

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Table B.5 [ ] Hot Uncontrolled k_∞

Table B.6 [ ] Hot Uncontrolled LPF

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Table B.7 [ ] Hot Uncontrolled k_∞

Table B.8 [ ] Hot Uncontrolled LPF

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Table B.9 [ ] Hot Uncontrolled k_∞

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Table B.10 [ ] Hot Uncontrolled LPF

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Table B.11 [ ] Hot Uncontrolled k_∞

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Table B.12 [ ] Hot Uncontrolled LPF

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Table B.13 [ ] Hot Uncontrolled k_∞

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Table B.14 [ ] Hot Uncontrolled LPF

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Table B.15 [ ] Hot Uncontrolled k_∞

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Table B.16 [ ] Hot Uncontrolled LPF

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Table B.17 [ ] Hot Uncontrolled k_∞

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Table B.18 [ ] Hot Uncontrolled LPF

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Table B.19 [ ] Hot Uncontrolled k_∞

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Table B.20 [ ] Hot Uncontrolled LPF

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Table B.21 [ ] Hot Uncontrolled k_∞

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Table B.22 [ ] Hot Uncontrolled LPF

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Table B.23 [ ] Hot Uncontrolled k_∞

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Table B.24 [ ] Hot Uncontrolled LPF

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Table B.25 [ ] Hot Uncontrolled k_∞

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Table B.26 [ ] Hot Uncontrolled LPF

Table B.27 [ ] Hot Uncontrolled k_{∞}

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Table B.28 [ ] Hot Uncontrolled LPF

Table B.29 [ ] Hot Uncontrolled k_∞

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Table B.30 [ ] Hot Uncontrolled LPF

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Table B.31 [ ] Hot Uncontrolled k_∞

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Table B.32 [ ] Hot Uncontrolled LPF

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Table B.33 [ ] Hot Uncontrolled k_∞

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Table B.34 [ ] Hot Uncontrolled LPF

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Table B.35 [ ] Hot Uncontrolled k_∞

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Table B.36 [ ] Hot Uncontrolled LPF

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Table B.37 [ ] Hot Uncontrolled k_{∞}

Table B.38 [ ] Hot Uncontrolled LPF

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Appendix C Enriched Lattice Isotopic Data Tables

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Table C.1 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

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Table C.2 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

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Table C.3 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

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Table C.4 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

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Table C.5 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

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Table C.6 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

Table C.7 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

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Table C.8 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

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Table C.9 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

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Table C.10 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

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Table C.11 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

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Table C.12 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

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Table C.13 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

Table C.14 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

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Table C.15 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

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Table C.16 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

Table C.17 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

Table C.18 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

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Table C.19 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

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Table C.20 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

Table C.21 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

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Table C.22 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

Table C.23 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

Table C.24 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

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Table C.25 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

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Table C.26 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

Table C.27 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

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Table C.28 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

Table C.29 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

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Table C.30 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

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Table C.31 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

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Table C.32 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

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Table C.33 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

Table C.34 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

[ ]

Table C.35 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

Table C.36 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

[ ]

Table C.37 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

[ ]

Table C.38 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

Table C.39 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

[ ]

Table C.40 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

[ ]

Table C.41 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

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Table C.42 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

[

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Table C.43 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

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Table C.44 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

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Table C.45 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

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Table C.46 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

[ ]

Table C.47 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

[ ]

Table C.48 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

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Table C.49 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

Table C.50 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

[ ]

Table C.51 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

Table C.52 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

[ ]

Table C.53 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

Table C.54 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

[ ]

Table C.55 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

[ ]

Table C.56 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

[ ]

Table C.57 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

Appendix D Lattice Enrichment Distribution Maps

Figure D.1 [ ] Enrichment Distribution

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Figure D.2 [ ] Enrichment Distribution

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Figure D.3 [ ] Enrichment Distribution

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Figure D.4 [ ] Enrichment Distribution

Figure D.5 [ ] Enrichment Distribution

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Figure D.6 [ ] Enrichment Distribution

Figure D.7 [ ] Enrichment Distribution

Figure D.8 [ ] Enrichment Distribution

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Figure D.9 [ ] Enrichment Distribution

Figure D.10 [ ] Enrichment Distribution

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Figure D.11 [ ] Enrichment Distribution

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Figure D.12 [ ] Enrichment Distribution

Figure D.13 [ ] Enrichment Distribution

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Figure D.14 [ ] Enrichment Distribution

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Figure D.15 [ ] Enrichment Distribution

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Figure D.16 [ ] Enrichment Distribution

Figure D.17 [ ] Enrichment Distribution

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Figure D.18 [ ] Enrichment Distribution

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Figure D.19 [ ] Enrichment Distribution

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Figure D.20 [ ] Enrichment Distribution

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Figure D.21 [ ] Enrichment Distribution

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Figure D.22 [ ] Enrichment Distribution

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Figure D.23 [ ] Enrichment Distribution

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Figure D.24 [ ] Enrichment Distribution

Figure D.25 [ ] Enrichment Distribution

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Figure D.26 [ ] Enrichment Distribution

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Figure D.27 [ ] Enrichment Distribution

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Figure D.28 [ ] Enrichment Distribution

[ ]

ENCLOSURE

ATTACHMENT 7b

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST
APPLICATION TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

AFFIDAVIT FOR

ANP-3877P REPORT, REVISION 0

MONTICELLO ATRIUM 11 EQUILIBRIUM FUEL

NUCLEAR FUEL DESIGN REPORT

OCTOBER 2020

(3 pages follow)

AFFIDAVIT

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-3877P, Revision 0 "Monticello ATRIUM 11 Equilibrium Fuel Nuclear Fuel Design Report," dated October 2020 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: October 22, 2020

Alan G Meg

Alan Meginnis

ENCLOSURE

ATTACHMENT 8a

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST
APPLICATION TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

ANP-3881NP REPORT, REVISION 0

MONTICELLO ATRIUM 11 EQUILIBRIUM CYCLE

FUEL CYCLE DESIGN REPORT

NOVEMBER 2020

(83 pages follow)

Monticello ATRIUM 11 Equilibrium Cycle

ANP-3881NP
Revision 0

Fuel Cycle Design Report

November 2020

Nature of Changes

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

Contents

List of Tables

List of Figures

Nomenclature

Acronym	**Definition**
ACE	Framatome critical power correlation
APLHGR	average planar linear heat generation rate
BOC	beginning of cycle
BWR	boiling water reactor
CSDM	cold shutdown margin
EOC	end of cycle
EOFP	end of full power capability
GWd/MTU	gigawatt days per metric ton of initial uranium
HEXR	hot excess reactivity
LHGR	linear heat generation rate
MCPR	minimum critical power ratio
MICROBURN-B2	Framatome Inc. advanced BWR core simulator methodology with PPR capability
MWd/MTU	megawatt days per metric ton of initial uranium
NRC	(United States) Nuclear Regulatory Commission
PPR	Pin Power Reconstruction. The PPR methodology accounts for variation in local rod power distributions due to neighboring assemblies and control state. The local rod power distributions are reconstructed based on the actual flux solution for each statepoint.
R Value	the larger of zero or the shutdown margin at BOC minus the minimum calculated shutdown margin in the cycle
SLC	standby liquid control

1.0 INTRODUCTION

This report documents the Framatome Inc. equilibrium cycle design and the results from a representative Cycle N for the Monticello BWR. This design analysis utilizes the ATRIUM 11 fuel design and has been performed with the approved Framatome Inc. neutronics methodology (References 1 and 3).

The CASMO-4 lattice depletion code was used to generate nuclear data including cross sections and local power peaking factors. The MICROBURN-B2 version 2 three dimensional core simulator code, combined with the application of the applicable critical power correlation (Reference 3), was used to model the core. The following MICROBURN-B2 version 2 modeling features were also used in the analyses supporting this report:

- pin power reconstruction (PPR) to determine the thermal margins
- **[ ]**
- **[ ]**
- **[ ]**
- **[ ]**

Design results including projected control rod patterns and evaluations of thermal and reactivity margins for the representative equilibrium Cycle N, hereafter identified as Cycle 15, are presented in this report.

2.0 SUMMARY

The equilibrium fresh fuel batch size [ ] and batch average enrichment [ ] were determined to meet the energy requirements provided by Xcel Energy in Reference 2. The loading of the Cycle 15 fuel as described in this report results in a projected full power energy capability of [ ]. Beyond the nominal full power capability, Cycle 15 has been designed to achieve [ ] of additional energy via power coastdown operation.

In order to obtain optimum operating flexibility, the projected control rod patterns were developed with acceptable margin to thermal limits. The equilibrium cycle design calculations also demonstrate adequate hot excess reactivity and cold shutdown margin throughout the cycle. Key results from the Cycle 15 analysis are summarized in Table 2.1. Table 2.2 summarizes the assembly identification range for Cycle 15 by nuclear fuel type batch. Tables 2.3, 2.4, and 2.5 contain the assumed thermal limits for the equilibrium design. Figures 2.1 and 2.2 provide a summary of the Cycle 15 design step-through projection.

Table 2.1 Cycle 15 Energy and Key Results Summary

Table 2.2 Cycle 15 Assembly ID Range by Nuclear Fuel Type

Table 2.3 Monticello ATRIUM 11 Equilibrium Cycle Design Assumed MCPR Operating Limit

[ ]

Table 2.4 Monticello ATRIUM 11 Equilibrium Cycle Design Assumed LHGR Limit

[ ]

Table 2.5 Monticello ATRIUM 11 Equilibrium Cycle Design Assumed MAPLHGR Limit

[ ]

Figure 2.1 Monticello Unit 1 Cycle 15 Design Step-through k-eff versus Cycle Exposure

Figure 2.2 Monticello Unit 1 Cycle 15 Design Margin to Thermal Limits versus Cycle Exposure

3.0 CYCLE 15 FUEL CYCLE DESIGN

3.1 *General Description*

Elevation views of the equilibrium fuel design axial enrichment and gadolinia distributions are shown in Appendix B, Figures B.1 through B.3 and originate from Reference 4. The loading pattern maintains quarter symmetry within a scatter load fuel management scheme. This loading, in conjunction with the control rod patterns presented in Appendix A, shows acceptable power peaking and associated margins to limits. The analyses supporting this equilibrium cycle design were based on the core parameters shown in Table 3.1. Figures 3.1 and 3.2, along with Table 3.1, define the reference loading pattern used in the equilibrium Cycle 15.

3.2 *Control Rod Patterns and Thermal Limits*

Projected control rod patterns and resultant key operating parameters including thermal margins from Cycle 15 are shown in Appendix A. The thermal margins presented in this report were determined using the MICROBURN-B2 3D core simulator PPR model to provide adequate margin to thermal limits from Reference 2. A detailed summary of the core parameters resulting from the step-through projection analysis is provided in Tables A.1 and A.2. Limiting results from the Cycle 15 step-through are summarized in Table 2.1 and in Figure 2.2. The hot operating target k-eff versus cycle exposure which was determined to be appropriate for this evaluation is shown in Table 3.2. The k-eff and margin to limits results from the Cycle 15 depletion are presented graphically in Figures 2.1 and 2.2. The k-eff values presented in Figure 2.1 and in Appendix A are not bias corrected. Selected exposure and radial power distributions from the Cycle 15 step-through are presented in Appendix C.

3.3 *Hot Excess Reactivity and Cold Shutdown Margin*

The Cycle 15 calculations demonstrate adequate hot excess reactivity, SLC shutdown margin, and cold shutdown margin throughout the cycle. Key shutdown margin and R-Value results are presented in Table 2.1. The shutdown margin is in conformance with the Technical Specification limit of R + 0.38 %Δk/k at BOC. The cold target k-eff versus exposure determined to be appropriate for calculation of cold shutdown margin is shown in Table 3.3. The core hot excess reactivity was calculated [ ]. Table 3.4 summarizes the reactivity margins versus cycle exposure, including the SLC shutdown margin for Cycle 15.

Table 3.1 Cycle 15 Core Composition and Design Parameters

[ ]

Table 3.2 Monticello ATRIUM 11 Equilibrium Cycle Design Hot Operating Target k-eff Versus Cycle Exposure

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Table 3.3 Monticello ATRIUM 11 Equilibrium Cycle Design Cold Critical Target k-eff Versus Cycle Exposure

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Table 3.4 Cycle 15 Reactivity Margin Summary

Table 3.4 Cycle 15 Reactivity Margin Summary *(Continued)*

Figure 3.1 Monticello Unit 1 Cycle 15 Reference Loading Pattern

Figure 3.2 Cycle 15 Upper Left Quarter Core Layout by Fuel Type

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Figure 3.2 Cycle 15 Upper Right Quarter Core Layout by Fuel Type *(Continued)*

Figure 3.2 Cycle 15 Lower Left Quarter Core Layout by Fuel Type *(Continued)*

Figure 3.2 Cycle 15 Lower Right Quarter Core Layout by Fuel Type *(Continued)*

4.0 REFERENCES

1. EMF-2158(P)(A) Revision 0, Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2, Siemens Power Corporation, October 1999.

2. FS1-0048226, Revision 2.0, Monticello ATRIUM 11 Equilibrium Neutronics Design Parameters, February 2020.

3. ANP-10335P-A, Revision 0, ACE/ATRIUM 11 Critical Power Correlation, May 2018.

4. ANP-3877P, Revision 0, Monticello ATRIUM 11 Equilibrium Fuel Nuclear Fuel Design Report, October 2020.

Appendix A Monticello Representative Equilibrium Cycle 15 Step-through Depletion Summary, Control Rod Patterns and Core Average Axial Power and Exposure Distributions

Table A.1 Cycle 15 Design Depletion Summary

Table A.2 Cycle 15 Design Depletion Thermal Margin Summary

Figure A.1 Cycle 15 Control Rod Pattern and Axial Distributions at [ ]

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Figure A.2 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.3 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.4 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.5 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.6 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.7 Cycle 15 Control Rod Pattern and Axial Distributions at [ ]

Figure A.8 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.9 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.10 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.11 Cycle 15 Control Rod Pattern and Axial Distributions at [ ]

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Figure A.12 Cycle 15 Control Rod Pattern and Axial Distributions at [ ]

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Figure A.13 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.14 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.15 Cycle 15 Control Rod Pattern and Axial Distributions at [ ]

Figure A.16 Cycle 15 Control Rod Pattern and Axial Distributions at [ ]

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Figure A.17 Cycle 15 Control Rod Pattern and Axial Distributions at [ ]

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Figure A.18 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.19 Cycle 15 Control Rod Pattern and Axial Distributions at [ ]

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Figure A.20 Cycle 15 Control Rod Pattern and Axial Distributions at [ ]

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Figure A.21 Cycle 15 Control Rod Pattern and Axial Distributions at [ ]

Figure A.22 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.23 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.24 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.25 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.26 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.27 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.28 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.29 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.30 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.31 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.32 Cycle 15 Control Rod Pattern and Axial Distributions at [ ]

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Figure A.33 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.34 Cycle 15 Control Rod Pattern and Axial Distributions at [ ]

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Figure A.35 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.36 Cycle 15 Control Rod Pattern and Axial Distributions at [ ]

Figure A.37 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.38 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.39 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.40 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.41 Cycle 15 Control Rod Pattern and Axial Distributions at
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Figure A.42 Cycle 15 Control Rod Pattern and Axial Distributions at [ ]

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Figure A.43 Cycle 15 Control Rod Pattern and Axial Distributions at
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Appendix B Elevation Views of the Monticello Equilibrium Cycle Design Fuel Assemblies

Figure B.1 Elevation View for the Monticello Equilibrium Cycle [ ] Fuel Assembly Design

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Figure B.2 Elevation View for the Monticello Equilibrium Cycle
[ ] Fuel Assembly Design

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Figure B.3 Elevation View for the Monticello Equilibrium Cycle
[ ] Fuel Assembly Design

Appendix C Monticello Representative Equilibrium Cycle 15 Radial Exposure and Power Distributions

Figure C.1 Cycle 15 BOC Exposure Distribution ([ ])

Figure C.1 Cycle 15 BOC Exposure Distribution ([ ]) *(Continued)*

Figure C.2 Cycle 15 EOC Exposure Distribution ([ ])

Figure C.2 Cycle 15 EOC Exposure Distribution ([ ]) *(Continued)*

[ ]

Figure C.3 Cycle 15 Radial Power Distribution at [ ]

Figure C.3 Cycle 15 Radial Power Distribution at [ ] *(Continued)*

Figure C.4 Cycle 15 Radial Power Distribution at [ ] (EOFP)

[ ]

Figure C.4 Cycle 15 Radial Power Distribution at [ ] (EOFP) *(Continued)*

ENCLOSURE

ATTACHMENT 8b

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST
APPLICATION TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

AFFIDAVIT FOR

ANP-3881P REPORT, REVISION 0

MONTICELLO ATRIUM 11 EQUILIBRIUM CYCLE

FUEL CYCLE DESIGN REPORT

NOVEMBER 2020

(3 pages follow)

AFFIDAVIT

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-3881P, Revision 0 "Monticello ATRIUM 11 Equilibrium Cycle Fuel Cycle Design Report," dated November 2020 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:

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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: November 10, 2020

Alan Meginnis