ENCLOSURE 4 CONTAINS PROPRIETARY INFORMATION
WITHOLD FROM PUBLIC DISCLOSURE IN ACCORDANCE WITH 10 CFR 2.390

2807 West County Road 75
Monticello, MN 55362

800.895.4999
xcelenergy.com

November 12, 2018

L-MT-18-060
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: Revise the Safety Limit Minimum Critical Power Ratio in Reactor Core Safety Limit 2.1.1

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 revise the values for the Safety Limit Minimum Critical Power Ratio (SLMCPR) within Specification 2.1.1, "Reactor Core SLs [Safety Limits]", in the Monticello Nuclear Generating Plant (MNGP) Technical Specifications (TS). Specifically, it is proposed to revise the SLMCPRs for two recirculation loop and single recirculation loop operation. Also, non-technical changes to Specification 2.1.1 and Specification 5.6.3, "Core Operating Limits Report (COLR)", remove outdated and duplicate information.

Enclosure 1 provides a description and assessment of the proposed TS changes. The enclosure also provides the no significant hazards consideration evaluation in accordance with 10 CFR 50.92, "Issuance of Amendment", and environmental assessment. These provide the bases for the conclusion that the license amendment request involves no significant hazards consideration and meets the eligibility criterion for a categorical exclusion as 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).

Attachment 1 to Enclosure 1 provides the existing TS pages marked up to show the proposed changes. Attachment 2 to Enclosure 1 provides the revised (clean) TS pages. Attachment 3 to Enclosure 1 provides the TS Bases pages marked-up to show the associated TS bases changes and is provided for information only.

Enclosure 2 provides a non-proprietary copy of a Framatome Inc. (hereafter "Framatome") report entitled, "Monticello Cycle 30 SAFLIM3D MCPR Safety Limit Results", which summarizes the reload safety analyses results and technical bases for the proposed changes.

Enclosure 3 provides an affidavit executed to support withholding the proprietary version of this report provided in Enclosure 4 from public disclosure. Enclosure 4 contains proprietary information as defined by 10 CFR 2.390, "Public inspections, exemptions, requests for withholding". The affidavit sets forth the basis on which this information may be withheld from public disclosure by the NRC and addresses with specificity the considerations listed in 10 CFR 2.390(b)(4). Accordingly, NSPM requests that the Framatome proprietary information contained in Enclosure 4 be withheld from public disclosure in accordance with 10 CFR 2.390(a)4, as authorized by 10 CFR 9.17, "Agency records exempt from public disclosure", paragraph (a)(4). Correspondence with respect to the copyright or proprietary aspects of the Framatome information provided in Enclosure 4 or in the supporting Framatome affidavit in Enclosure 3 should be addressed to Mr. Alan Meginnis, Manager – Product Licensing, Framatome Inc., 2101 Horn Rapids Road, Richland, Washington 99354.

NSPM requests NRC approval of the proposed license amendment request by April 30, 2019, with an implementation period coinciding with startup from the spring 2019 refueling outage.

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 the non-proprietary enclosures and attachments, to the designated State Official.

If additional information is needed, please contact Mr. Richard Loeffler at (612) 342-8981.

Summary of Commitments

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

I declare under penalty of perjury that the foregoing is true and correct.
Executed on November 12, 2018.

Christopher R. Church
Site Vice President, Monticello Nuclear Generating Plant
Northern States Power Company – Minnesota

Enclosures (4)

cc: Administrator, Region III, US NRC (w/o Enclosure 4)
Project Manager, Monticello, US NRC
Resident Inspector, US NRC (w/o Enclosure 4)
State of Minnesota (w/o Enclosure 4)

ENCLOSURE 1

MONTICELLO NUCLEAR GENERATING PLANT

EVALUATION OF THE PROPOSED CHANGE

LICENSE AMENDMENT REQUEST: REVISE THE SAFETY LIMIT MINIMUM CRITICAL POWER RATIO IN REACTOR CORE SAFETY LIMIT 2.1.1

ATTACHMENTS:

1. Technical Specification Pages (Mark-up)
2. Technical Specification Pages (Retyped)
3. Technical Specification Bases Pages (Mark-up – for information only)

LICENSE AMENDMENT REQUEST: REVISE THE SAFETY LIMIT MINIMUM CRITICAL POWER RATIO IN REACTOR CORE SAFETY LIMIT 2.1.1

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 revise the Monticello Nuclear Generating Plant (MNGP) Technical Specifications (TS). This proposed change revises the Safety Limit Minimum Critical Power Ratio (SLMCPR) values within the Reactor Core Safety Limits specification, i.e., Specification 2.1.1, to reflect the results of the reload safety analyses for two recirculation loop and single recirculation loop operation. Also, non-technical changes to Specification 2.1.1 and Specification 5.6.3, "Core Operating Limits Report (COLR)", remove outdated and duplicate information.

2.0 DETAILED DESCRIPTION

2.1 Facility Description

MNGP is a single unit plant located on the south bank of the Mississippi River within the city limits of Monticello, Minnesota. The MNGP is a single cycle, forced circulation, low power density boiling water reactor, designed and supplied by the General Electric Corporation. The MNGP application for a Construction Permit and Operating License was submitted to the Atomic Energy Commission (AEC) on August 1, 1966. Amendment No. 1 to Provisional Operating License No. DPR-22 was issued on January 13, 1971, granting full power operation. MNGP began full power commercial operation on June 30, 1971.

The MNGP was designed and constructed to comply with NSPM's understanding of the intent of the AEC 70 proposed General Design Criteria (GDC) for Nuclear Power Plant Construction Permits, as published on July 11, 1967. MNGP was not licensed to NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants".

2.2 Reasons for the Proposed TS Changes

Results of cycle-specific reload safety analyses performed by Framatome, Inc., formerly "AREVA" (hereafter "Framatome"), demonstrate that required safety margins are maintained for both two and single recirculation loop reactor operation with a reduction in the SLMCPR values. These new cycle-specific SLMCPRs allow for more efficient operation.

As a separate, non-technical change, references to previous SLMCPR values determined under GEH [General Electric – Hitachi] safety analysis methodologies are proposed to be removed as they are no longer applicable.

2.3 Description of the Proposed TS Changes

The SLMCPR values determined by the cycle-specific reload safety analyses to replace the current SLMCPR values in Specification 2.1.1 are:

	Current TS Value	New TS Value
Two recirculation loop operation:		
Operation not in the EFW[1] domain or Operation in EFW domain and ratio of power to core flow < 42 MWt/Mlb/hr	1.15	1.08
Operation in EFW domain and ratio of power to core flow ≥ 42 MWt/Mlb/hr	1.19	1.14
Single recirculation loop operation:	1.20	1.13

(1) EFW stands for Extended Flow Window

The following non-technical changes are also proposed to Specification 2.1.1:

- Remove statements and values for GEH determined SLMCPRs.
- Remove references to "GEH" or "AREVA" methods as differentiation between the two methodologies is no longer necessary.
- Replace GEH SLMCPR statement in Specification 2.1.1.2 with "(Deleted)".

Attachment 1 to this enclosure provides the existing TS page marked-up to show the proposed TS changes. Attachment 2 provides a revised (clean) TS page. Specification 2.1.1 would then read after these changes are incorporated:

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 586 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 (Deleted)

2.1.1.3 With the reactor steam dome pressure ≥ 586 psig and core flow ≥ 10% rated core flow:

a. For operation not in the EFW domain, MCPR shall be ≥ 1.08 for two recirculation loop operation, or ≥ 1.13 for single recirculation loop operation, or

b. For operation in the EFW domain and the ratio of power to core flow < 42 MWt/Mlb/hr, MCPR shall be ≥ 1.08,

or

c. For operation in the EFW domain and the ratio of power to core flow ≥ 42 MWt/Mlb/hr, MCPR shall be ≥ 1.14.

2.1.1.4 Reactor vessel water level shall be greater than the top of active irradiated fuel.

Also, it is proposed to remove a duplicate reference, i.e., Item 24, in Specification 5.6.3 to topical report BAW-10255(P)(A), Revision 2, “Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code”, AREVA NP Inc., dated May 2008.

Attachment 3 to this enclosure provides the TS Bases pages marked-up to reflect these changes and is provided for information only. TS Bases changes are issued in accordance with MNGP Specification 5.5.9, “Technical Specification (TS) Bases Control Program”, following approval of the associated license amendment request.

2.4 Impact on Submittals Under NRC Review

There is no impact on any submittals currently under NRC review since the proposed TS changes do not involve any specifications that are proposed to be modified in the other submittals.

3.0 TECHNICAL EVALUATION

3.1 Safety Limit MCPR Analysis

Specification 2.1.1 provides the SLMCPRs pertaining to two and single recirculation loop operation. This license amendment request proposes to revise the SLMCPRs for two recirculation loop operation and single recirculation loop operation. The SLMCPRs are calculated by applying the NRC approved analysis methodologies listed in the COLR specification in the MNGP TS. The reactor core for the next cycle consists of a mixed core of ATRIUM™ 10XM and GE14 fuel types.

The SLMCPR is defined as the minimum value of the critical power ratio which ensures that at least 99.9% of the fuel rods in the core are expected to avoid boiling transition during normal operation or anticipated operational occurrences. The SLMCPR is determined using a statistical analysis employing a Monte Carlo process that perturbs the input parameters used in the calculation of the minimum critical power ratio (MCPR). The set of uncertainties used in the statistical analysis includes both fuel-related and plant-related uncertainties. The SLMCPR for two recirculation loop operation and for single recirculation loop operation were determined by applying the methodology presented in ANP-10307PA, “AREVA MCPR Safety Limit

Methodology for Boiling Water Reactors" (Reference 1). For the ATRIUM 10XM fuel, the updated version of the correlation presented in ANP-10298P-A, "ACE/ATRIUM 10XM Critical Power Correlation" (Reference 2), was applied. For the GE14 fuel, Revision 3 of the correlation presented in EMF-2209(P)(A), "SPCB Critical Power Correlation" (Reference 3), was applied.

The SLMCPR analysis is performed with a power distribution that conservatively represents expected reactor operating states that could both exist at the Operating Limit MCPR (OLMCPR) and produce a MCPR equal to the SLMCPR during anticipated operational occurrences. In the Framatome methodology, the effects of fuel channel bow on critical power performance are accounted for in the SLMCPR analysis.

Enclosures 2 and 4 provide a non-proprietary and a proprietary version of a Framatome report entitled, "Monticello Cycle 30 SAFLIM3D MCPR Safety Limit Results", (References 4 and 5, respectively). This report provides a summary of the methodology and results of the cycle-specific reload safety analysis results supporting the SLMCPRs determined for the next MNGP operating cycle.

Table 1 in this report identifies the uncertainties used within the MNGP cycle-specific SLMCPR analysis. Table 2 in this report provides the results of the SLMCPR analysis. Figure 1 in this report provides a copy of the Power / Flow Map for the MNGP.

When in two recirculation loop operation in the EFW domain above a ratio of core power to core flow greater than or equal to 42 MWt/Mlb/hr, an additional 0.03 CPR is added to the new cycle-specific SLMCPR determined by the reload safety analyses to account for lack of information on the power distribution uncertainties within this region, as presented in Reference 6.

The new cycle-specific SLMCR values determined for the next operating cycle are:

Two recirculation loop operation:

Operation not in the EFW domain or Operation in EFW domain and ratio of power to core flow < 42 MWt/Mlb/hr	1.08
Operation in EFW domain and ratio of power to core flow ≥ 42 MWt/Mlb/hr	1.14

Single recirculation loop operation: 1.13

No plant hardware or operational changes are required to be implemented with this proposed license amendment.

3.2 GARDEL Core Monitoring Software

The Studsvik Scandpower GARDEL core monitoring software is used as the core monitoring computer system at the MNGP. The GARDEL software uses Traversing Incore Probe (TIP) data and Local Power Range Monitors (LPRMs) readings to calculate “adaptive” thermal margins that take into account the power shapes as measured by nuclear instrumentation. The methodology for determining the radial and nodal power uncertainties is described in Studsvik Scandpower Topical Report SSP-09/444-C, “GARDEL BWR – Monticello NPP Power Distribution Uncertainties” (Reference 7). This report was provided in both proprietary and non-proprietary form in response to a request for additional information (Reference 8) in support of a previous SLMCPR change requested for the MNGP in 2010 (Reference 9). This prior SLMCPR change was approved by the NRC for the MNGP as License Amendment No. 165 (Reference 10). More recently, in August 2015, application of GARDEL was discussed in response to a request for additional information during the NRC review of the EFW license amendment request (Reference 11). Utilization of the EFW operating domain at the MNGP was approved by the NRC under License Amendment No. 191 (Reference 12).

3.3 Conclusions

The proposed changes are acceptable based on:

- Results of the MNGP cycle-specific reload safety analyses performed by Framatome indicate the required safety margin can be maintained while operating with the proposed revised SLMCPR values.
- The requested non-technical changes remove outdated references to SLMCPRs determined under the previous fuel vendor safety analysis methods. This enhances the readability of the specification by removing obsolete detail.
- Removal of Item 24 in Specification 5.6.3 removes a reference to the same topical report listed under Item 22.

4.0 REGULATORY ANALYSIS

4.1 Applicable Regulatory Requirements

10 CFR 50.36, “Technical specifications", details the content and information that must be included in a station's TS. In accordance with the requirements of 10 CFR 50.36, one of the items required to be included are safety limits. NUREG-0800, "Standard Review Plan for the Review of Safety Analyses Reports for Nuclear Power Plants", Section 4.4, "Thermal and Hydraulic Design", states that the critical power ratio is to be established such that at least 99.9 percent of fuel rods in the core would not be expected to experience departure from nucleate boiling or boiling transition during normal operation or anticipated operational occurrences.

The applicable 10 CFR 50, Appendix A, GDC criterion is presented first below followed by the corresponding criterion from the AEC 70 proposed GDCs provided for comparison. The applicable Principal Design Criteria (PDC) from the MNGP Updated Safety Analysis Report (USAR) Subsection 1.2.2, “Reactor Core”, are then provided for comparison.

10 CFR 50, Appendix A, GDC Criterion 10, “Reactor Design”, states:

> The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

The corresponding AEC proposed GDC Criterion 6, “Reactor Core Design (Category A)”, states:

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

Meeting the intent of the 10 CFR 50, Appendix A GDCs is supported by the design of the plant to the General Electric PDCs stated below from Subsection 1.2.2, “Reactor Core”, of the MNGP USAR:

> h. Thermal characteristics of the reactor core are adequate to prevent fuel clad surface heat flux or fuel material center temperatures which could cause sudden fuel cladding ruptures.

i. The reactor core and associated systems are designed to accommodate plant operational transients or maneuvers which might be expected without compromising safety and without fuel damage.

The 10 CFR 50, Appendix A GDCs, the AEC 70 proposed GDCs, and the MNGP PDCs, while worded somewhat differently, are equivalent in that the reactor core is designed to ensure that specified acceptable fuel design limits are not exceeded during both normal operation and during anticipated operational occurrences.

4.2 Precedents

Amendments to periodically revise the SLMCPRs are submitted and approved by the NRC for the various fuel vendors' analysis methodologies and the associated fuel types. The most recent amendments noted that involved a change to the SLMCPRs where Framatome, formerly "AREVA", safety analysis methodologies and AREVA fuel types were used, were for the Brunswick Steam Electric Plant, Units 1 and 2 (Reference 13), and for the Dresden Nuclear Power Station, Units 2 and 3, together with the Quad Cities Nuclear Power Station, Units 1 and 2 (Reference 14).

4.3 No Significant Hazards Consideration Analysis

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 revise the Monticello Nuclear Generating Plant (MNGP) Technical Specifications (TS). This proposed change revises the Safety Limit Minimum Critical Power Ratio (SLMCPR) values within the Reactor Core Safety Limits specification, i.e., Specification 2.1.1, to reflect the results of the reload safety analyses for two recirculation loop operation and single recirculation loop operation. Also, non-technical changes to Specification 2.1.1 and Specification 5.6.3, "Core Operating Limits Report (COLR)", remove outdated and duplicate information.

NSPM has evaluated the proposed change against the criteria of 10 CFR 50.92, "Issuance of amendment", to determine if the proposed change results in any significant hazards. The following is the evaluation of each of the 10 CFR 50.92(c) criteria:

1. Does the proposed change 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 amendment does not involve any plant modifications or operational changes that could affect system reliability or performance, or that could affect the probability of operator error. As such, the proposed changes do not affect any postulated accident precursors. Since no individual precursors of an accident are affected, the proposed amendment does not involve a significant increase in 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 basis for the SLMCPR calculation is to ensure that during normal operation and during anticipated operational occurrences, at least 99.9 percent of all fuel rods in the core do not experience transition boiling if the safety limit is not exceeded.

The revised SLMCPR values provide sufficient margin to transition boiling and the probability of fuel damage is not increased. The derivation of the cycle specific SLMCPR values have been performed applying the NRC approved applicable Framatome fuel licensing methodologies. As such, the proposed amendment involves no changes to the operation of any system or component during normal, accident, or transient operating conditions. The change does not affect the initiators of any accident.

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

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

   Response: No

   The revised SLMCPR are calculated applying NRC approved fuel analysis methodologies. 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 TS changes do not involve any new modes of operation or any changes to setpoints or any plant modifications. The revised SLMCPR have been shown to be acceptable by analysis for the next cycle of operation. The core operating limits will continue to be developed using NRC approved methods. The proposed SLMCPRs and the methods for establishing the core operating limits do not result in the creation of any new precursors to an accident.

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

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The SLMCPR provides a margin of safety by ensuring that at least 99.9 percent of the fuel rods do not experience transition boiling during normal operation and anticipated operational occurrences if the limit is not exceeded. Revision of the SLMCPR values using an NRC approved methodology, ensures that the required level of fuel protection is maintained by continuing to ensure that the fuel design safety criterion is met, i.e., that no more than 0.1 percent of the rods are expected to be in boiling transition if the SLMCPR is not exceeded.

   The margin of safety is established through the design of plant structures, systems, and components, and through the parameters for safe operation and setpoints of equipment relied upon to respond to transients and design basis accidents. The proposed change in SLMCPR does not change the requirements governing operation or availability of safety equipment assumed to operate to preserve the margin of safety. The change does not alter the behavior of the plant equipment.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NSPM concludes that the proposed amendment does not involve a 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 EVALUATION

A review has determined that the proposed amendment 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 amendment 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 amendment 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 be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. ANP-10307PA Revision 0, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors", AREVA NP, June 2011

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

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

4. Framatome Report FS1-0039793, Revision 1.0, "Monticello Cycle 30 SAFLIM3D MCPR Safety Limit Results (Non-Proprietary Version)", dated September 24, 2018

5. Framatome Report FS1-0039792, Revision 1.0, "Monticello Cycle 30 SAFLIM3D MCPR Safety Limit Results (Proprietary Version)", dated September 24, 2018

6. Letter from NSPM to NRC, "License Amendment Request for AREVA Extended Flow Window Supplement to Address Power Distribution Uncertainties (TAC No. MF5002)", (L-MT-16-041) dated September 14, 2016 (ADAMS Accession Number ML16258A480)

7. Studsvik Scandpower Report SSP-09/444-C, Rev.0, "GARDEL BWR – Monticello NPP Power Distribution Uncertainties", dated July 24, 2009 (ADAMS Accession Number ML110450403 – non-proprietary version)

8. Letter from NSPM to NRC, "Response to Requests for Additional Information (RAI) for the License Amendment Request to Revise the Minimum Critical Power Ratio Safety Limit in Reactor Core Safety Limit 2.1.1.2 (TAC No. ME4790)", (L-MT-11-009) dated February 8, 2011 (ADAMS Accession Number ML110450240)

9. Letter from NSPM to NRC, "License Amendment Request: Revise the Minimum Critical Power Ratio Safety Limit in Reactor Core Safety Limit 2.1.1.2", (L-MT-10-055) dated September 17, 2010 (ADAMS Accession Number ML102650399)

10. Letter from NRC to NSPM, "Monticello Nuclear Generating Plant – Issuance of Amendment Re: Minimum Critical Power Ratio Safety Limit (TAC No. ME4790)", dated May 4, 2011 (ADAMS Accession Number ML11101A111)

11. Letter from NSPM to NRC, "License Amendment Request for AREVA Extended Flow Window Supplement to Respond to NRC Staff Questions (TAC No. MF5002)", (L-MT-15-057) dated August 26, 2015 (ADAMS Accession No. ML15348A221)

12. Letter from NRC to NSPM, “Monticello Nuclear Generating Plant – Issuance of Amendment Re: Extended Flow Window (CAC No. MF5002)”, dated February 23, 2017 (ADAMS Accession Number ML16342B276)

13. Letter from NRC to Duke Energy Progress, LLC, “Brunswick Steam Electric Plant, Units 1 and 2 – Issuance of Amendments to Revise Technical Specification Section 2.1.1.2 Safety Limit Minimum Critical Power Ratio (CAC Nos. MF8470 and MF8471)”, dated March 10, 2017 (ADAMS Accession Number ML17059D146)

14. Letter from NRC to Exelon Generation Company, LLC, “Dresden Nuclear Power Station, Unit Nos. 2 and 3, and Quad Cities Nuclear Power Station, Unit Nos. 1 and 2 – Issuance of Amendments to Renewed Facility Operating License Nos. DPR-19, DPR-25, DPR-29, and DPR-30, to Revise Technical Specifications to Support Transitioning to AREVA Nuclear Fuel (CAC Nos. MF5736, MF5737, MF5738, and MF5739)”, dated October 20, 2016 (ADAMS Accession Number ML16218A498)

ENCLOSURE 1

ATTACHMENT 1

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST: REVISE THE SAFETY LIMIT MINIMUM CRITICAL POWER RATIO IN REACTOR CORE SAFETY LIMIT 2.1.1

TECHNICAL SPECIFICATION PAGES (MARK-UP)

2 pages follow

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 ~~With the reactor steam dome pressure < 686 psig or core flow < 10% rated core flow (GEH methods):~~

~~or~~

With the reactor steam dome pressure < 586 psig or core flow < 10% rated core flow ~~(AREVA methods)~~:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 (Deleted)~~With the reactor steam dome pressure ≥ 686 psig and core flow ≥ 10% rated core flow (GEH methods) MCPR shall be ≥ 1.15 for two recirculation loop operation or ≥ 1.15 for single recirculation loop operation.~~

2.1.1.3 With the reactor steam dome pressure ≥ 586 psig~~,~~ and core flow ≥ 10% rated core flow ~~(AREVA methods)~~:

a. For operation not in the EFW domain, MCPR shall be ≥ 1.08~~1.15~~ for two recirculation loop operation, or ≥ 1.13~~1.20~~ for single recirculation loop operation,

or

b. For operation in the EFW domain and the ratio of power to core flow < 42 MWt/Mlb/hr, MCPR shall be ≥ 1.08~~1.15~~,

or

c. For operation in the EFW domain and the ratio of power to core flow ≥ 42 MWt/Mlb/hr, MCPR shall be ≥ 1.14~~1.19~~.

2.1.1.4 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1332 psig.

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

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

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 1

ATTACHMENT 2

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST: REVISE THE SAFETY LIMIT MINIMUM CRITICAL POWER RATIO IN REACTOR CORE SAFETY LIMIT 2.1.1

TECHNICAL SPECIFICATION PAGES (RETYPED)

2 pages follow

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 586 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 (Deleted)

2.1.1.3 With the reactor steam dome pressure ≥ 586 psig and core flow ≥ 10% rated core flow:

a. For operation not in the EFW domain, MCPR shall be ≥ 1.08 for two recirculation loop operation, or ≥ 1.13 for single recirculation loop operation,

or

b. For operation in the EFW domain and the ratio of power to core flow < 42 MWt/Mlb/hr, MCPR shall be ≥ 1.08,

or

c. For operation in the EFW domain and the ratio of power to core flow ≥ 42 MWt/Mlb/hr, MCPR shall be ≥ 1.14.

2.1.1.4 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1332 psig.

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

24. (Deleted)

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 1

ATTACHMENT 3

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST: REVISE THE SAFETY LIMIT MINIMUM CRITICAL POWER RATIO IN REACTOR CORE SAFETY LIMIT 2.1.1

TECHNICAL SPECIFICATION BASES PAGES

(MARK-UP – FOR INFORMATION ONLY)

7 pages follow

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

USAR Section 1.2.2 (Ref. 1) requires the reactor core and associated systems to be designed to accommodate plant operational transients or maneuvers that might be expected without compromising safety and without fuel damage. Therefore, SLs ensure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs).

No changes.
For Info.

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish an SL, such that the MCPR is not less than the limit specified in Specification 2.1.1.2. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical barriers that separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses, which occur from reactor operation significantly above design conditions.

While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross, rather than incremental, cladding deterioration. Therefore, the fuel cladding SL is defined with a margin to the conditions that would produce onset of transition boiling (i.e., MCPR = 1.00). These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity SL ensures that during normal operation and during AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical

BASES

BACKGROUND (continued)

reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The reactor vessel water level SL ensures that adequate core cooling capability is maintained during all MODES of reactor operation. Establishment of Emergency Core Cooling System initiation setpoints higher than this SL provides margin such that the SL will not be reached or exceeded.

APPLICABLE SAFETY ANALYSES

~~Refer to Section 1.0 of the CORE OPERATING LIMIT REPORT (COLR) to determine whether General Electric – Hitachi (GEH) or AREVA methods were used for the current operating cycle.~~

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the fuel design criterion that a MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with the other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR Safety Limit.

~~Specifications 2.1.1.1 and 2.1.1.2 are each written to describe a Safety Limit that is appropriate for the respective safety analysis method being used to operate the reactor. This TS construction was approved to provide flexibility during the MNGP transition from operating under the GEH safety analysis methods to the AREVA safety analysis methods. Separate SLs were required because no single value of steam dome pressure would accurately cover the applicability range of GEH methodology as well as the applicability range of AREVA methodology during fuel transition operations. To accommodate this transition, TS 2.1.1.1 and 2.1.1.2 are structured to require the use of the GEH safety limit when operating under GEH safety analysis methods and use of the AREVA safety limit when operating under the AREVA safety analysis methods. The method used for safety analysis (whether GEH or AREVA) is established during the core reload safety analysis process that precedes any particular fuel cycle. In that same process, the appropriate operating limits for that analysis methodology are provided in the COLR.~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~The approved pressure range (700 to 1400 psia) of the GEXL 14 critical power correlation is applied to resolve a 10 CFR Part 21 condition concerning a potential to violate Reactor Core Safety Limit 2.1.1.1 during a Pressure Regulator Failure Maximum Demand (Open) transient (Reference 5). Application of this correlation, which applies to the GE14 fuel in the core, allows reduction of the reactor steam dome pressure from 785 to 686 psig, precluding violation of the safety limit for this event. This change in reactor steam dome pressure was approved in Amendment 185 (Reference 7).~~

~~AREVA~~ Framatome critical power correlations (ACE and SPCB) are applicable at reactor steam dome pressures > 586 psig. A Pressure Regulator Failure Maximum Demand (Open) transient applying ~~AREVA~~ Framatome safety analysis methods would not violate Reactor Core Safety Limit 2.1.1.1.

2.1.1.1 Fuel Cladding Integrity

~~AREVA~~ Framatome critical power correlation ACE is applicable at pressures ≥ 586 psig and core flows > 10% of rated flow. ~~AREVA~~ Framatome critical power correlation SPCB is applicable at pressures ≥ 586 psig and bundle inlet mass fluxes of ≥ 0.18 Mlb/hr/ft^2. ~~AREVA correlations (approved in Am. 188, Ref. 12) are used for cores analyzed with AREVA safety analysis methods, with the~~ The ACE correlation is used for ~~AREVA~~ Framatome fuel and the SPCB correlation is used for co-resident fuel.

~~The GEXL14 critical power correlation is applicable for all critical power calculations at pressures ≥ 686 psig and core flows ≥ 10% of rated flow (Reference 6).~~ For operation at low pressures or low flows, another basis is used, as follows:

> Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.56 psi. Analyses (Ref. 2) show that with a bundle flow of 28 x 10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be > 28 x 10^3 lb/hr. Full scale ATLAS test data taken at pressures from 0 psig to 785 psig indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 50 % RTP. Thus, a THERMAL POWER limit of 25% RTP for reactor pressure < 686 psig or < 10% core flow is conservative.

2.1.1.2 (Deleted)

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~2.1.1.2 and~~ 2.1.1.3 MCPR

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.

The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power.

The

~~For operating cycles using AREVA safety analysis methods, the~~ probability of the occurrence of boiling transition is determined using the approved ~~AREVA~~ correlations. ~~For such operating cycles,~~ References 8, 9, 10, and 11 describe the uncertainties and methodologies used in determining the MCPR SL. ~~However, based on reduced confidence in power distribution uncertainties in the extended operating domain (Extended Flow Window (EFW)), TS Safety Limit 2.1.1.3.c includes a penalty of 0.03 that must be added to the SLMCPR (i.e., the SLMCPR calculated with AREVA methods and uncertainties) when the ratio of core power to core flow equals or exceeds 42 MWt/Mlb/hr in the EFW domain.~~ This threshold is provided in Reference 13, and the basis for the 0.03 penalty is provided in Reference 14. This threshold is appropriate for MNGP because it represents a sufficiently high power-flow ratio that is outside the normal range of plant maneuvering. In this way, the SLMCPR adder (0.03) will not adversely affect full power operation. The adder (0.03) is not imposed on single-loop operation because single-loop operation is prohibited in the EFW region.

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TS Safety Limit 2.1.1.3.c applies when the ratio of core power to core flow equals or exceeds 42 MWt/Mlb/hr in the extended operating domain (Extended Flow Window (EFW)). Safety Limit 2.1.1.3.c consists of: 1) the cycle-specific SLMCPR calculated with Framatome methods and uncertainties each reload, and 2) a penalty of 0.03 added due to reduced confidence in power distribution uncertainties when operating at greater than 42 MWt/Mlb/hr in the EFW domain.

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~For operating cycles using GEH safety analysis methods, the probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations. Details of the fuel cladding integrity SL calculation are given in Reference 2. Reference 3 includes a tabulation of the uncertainties used in the determination of the MCPR SL and of the nominal values of the parameters used in the MCPR SL statistical analysis.~~

2.1.1.4 Reactor Vessel Water Level

During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active irradiated fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to prevent the release of radioactive materials to the environs. SL 2.1.1.1 and SL ~~2.1.1.2~~ 2.1.1.3 ensure that the core operates within the fuel design criteria. SL ~~2.1.1.3~~ 2.1.1.4 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, ~~2.1.1.2~~ 2.1.1.3, and ~~2.1.1.3~~ 2.1.1.4 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 50.67, “Accident source term,” limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

BASES

REFERENCES

1. USAR, Section 1.2.2.

2. ~~NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (revision specified in Specification 5.6.3).~~ (Deleted)

3. ~~NEDE-31152P, "General Electric Fuel Bundle Designs," Revision 8, April 2001.~~ (Deleted)

4. 10 CFR 50.67.

5. ~~GE Part 21 Notification SC05-03, “Potential to Exceed Low Pressure Technical Specification Safety Limit,” dated March 29, 2005.~~ (Deleted)

6. ~~NRC Letter to A. Lingenfelter (GNF), ‘Final Safety Evaluation for Global Nuclear Fuel (GNF) Topical Report (TR) NEDC-32851P, Revision 2, "GEXL14 Correlation for GE14 Fuel," (TAC No. MD5486)’ dated August 3, 2007.~~ (Deleted)

7. Amendment No. 185, “Issuance of Amendment to Reduce the Reactor Steam Dome Pressure Specified in the Reactor Core Safety Limits,” dated November 25, 2014. (ADAMS Accession No. ML14281A318).

8. EMF-2209(P)(A) Revision 3, “SPCB Critical Power Correlation”, AREVA NP, September 2009.

9. EMF-2245(P)(A) Revision 0, “Application of Siemens Power Corporation’s Critical Power Correlations to Co-Resident Fuel,” Siemens Power Corporation, August 2000.

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

11. ANP-10307PA, Revision 0, “AREVA MCPR Safety Limit Methodology for Boiling Water Reactors,” AREVA NP, June 2011.

12. Amendment No. 188, “Issuance of Amendment to Transition to AREVA ATRIUM 10XM Fuel and AREVA Safety Analysis Methods,” dated June 5, 2015. (ADAMS Accession Nos. ML15072A141, ML15154A477, and ML15072A135)

13. NRC letter to General Electric - Hitachi, Final Safety Evaluation for GE Hitachi Nuclear Energy Americas Topical Report NEDC-33173P, Revision 2 and Supplement 2, Parts 1-3, “Analysis of Gamma Scan Data and Removal of Safety Limit Minimum Critical Power Ratio (SLMCPR) Margin” (TAC No. ME1891). (ADAMS Accession No. ML113340215)

BASES

REFERENCES (continued)

14. GE-Hitachi, Final SE for NEDC-33173P, “Applicability of GE Methods to Expanded Operating Domains,” dated July 21, 2009. (ADAMS Accession No. ML083520464). This SE is an enclosure to NEDC-33173 Revision 4.

No changes.
For Info.

ENCLOSURE 2

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST: REVISE THE SAFETY LIMIT MINIMUM CRITICAL POWER RATIO IN REACTOR CORE SAFETY LIMIT 2.1.1

FRAMATOME REPORT FS1-0039793, REVISION 1.0, MONTICELLO CYCLE 30 SAFLIM3D MCPR SAFETY LIMIT RESULTS (NON-PROPRIETARY VERSION)

11 pages follow

IDENTIFICATION	REVISION
FS1-0039793	1.0

Framatome Fuel

framatome

TOTAL NUMBER OF PAGES: 11

Monticello Cycle 30 SAFLIM3D MCPR Safety Limit Results (Non-Proprietary Version)

ADDITIONAL INFORMATION:

This is the Non-Proprietary Version of FS1-0039792.

PROJECT	Monticello-1 (USA010)
HANDLING	None
CATEGORY	EIR - Engineering Information Report
STATUS	

DISTRIBUTION TO	PURPOSE OF DISTRIBUTION
Tony Will	

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Writer	HOWLETT Dirk	2018/09/21 08:42:01	Framatome Inc.
Reviewer	CHEN Wei-Ting	2018/09/24 18:15:43	Framatome Inc.
Approver	SCHNEPP Robert	2018/09/24 18:28:26	Framatome Inc.
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REVISIONS

REVISION	DATE	EXPLANATORY NOTES
1.0	See 1st page release date	New document. This is the Non-Proprietary Version of FS1-0039792.

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LIST OF FIGURES

1. SUMMARY OF RESULTS / CONCLUSIONS

The purpose of this analysis is to provide the safety limit minimum critical power ratio (SLMCPR) results for Monticello Nuclear Generating Plant (MNGP) in support of Cycle 30.

1.1. METHODOLOGY

The two-loop operation (TLO) and single-loop operation (SLO) SLMCPRs were determined using the methodology presented in Reference 1. The SLMCPR is defined as the minimum value of the critical power ratio which ensures at least 99.9% of the fuel rods in the core are expected to avoid boiling transition (BT) during normal operation or an anticipated operational occurrence (AOO). The SLMCPR is determined using a statistical analysis employing a Monte Carlo process that perturbs the input parameters used in the calculation of minimum critical power ratio (MCPR). The set of uncertainties used in the statistical analysis includes both fuel-related and plant-related uncertainties.

The SLMCPR analysis is performed with a power distribution that conservatively represents expected reactor operating states that could both exist at the operating limit MCPR (OLMCPR) and produce a MCPR equal to the SLMCPR during an AOO. [

]

In the Framatome methodology, the effects of channel bow on the critical power performance are accounted for in the SLMCPR analysis. [

]

[

] This adjustment is a plant specific extension of the Reference 1 approved methodology and is discussed in Reference 2 Section 2-25. [

]

During their review of the extended flow window (EFW) submittal, the NRC suggested a core power / core flow ratio of 42 MWt/Mlb/hr serve as the threshold above which a 0.03 penalty must be added to the SLMCPR otherwise supported by Framatome analysis methods. A 0.03 penalty has been applied when the ratio of core power to core flow is ≥ 42 MWt / Mlbm/hr in the EFW region, refer to Figure 1. The 0.03 penalty is not applied when MNGP is operating in the Maximum Extended Load Line Limit (MELLLA) region or operating in the EFW region where the ratio of core power to core flow is < 42 MWt / Mlbm/hr. Single-loop operation is not allowed in the EFW; therefore, the 0.03 penalty implemented when the ratio of core power to core flow is ≥ 42 MWt / Mlbm/hr in the EFW region is not applicable to the SLO SLMCPR.

1.2. ANALYSIS

The core loading and cycle depletion from the Monticello Cycle 30 fuel cycle design was used as the basis for the SLMCPR analysis. The cycle is made up of GE14 and ATRIUM™ 10XM* fuel. TLO analyses were performed [

] for the Monticello Cycle 30 power flow map. Analyses were also performed [

] The SLO analyses used a [

]

---

* ATRIUM is a trademark of Framatome Inc.

The ACE/ATRIUM 10XM critical power correlation (Reference 4) is used for the ATRIUM 10XM fuel while the SPCB critical power correlation (Reference 5) is used for the GE14 fuel. The application of the SPCB critical power correlation to GE14 fuel [ ]

The uncertainties used in the SLMCPR analysis are presented in Table 1. Appropriate core monitoring uncertainties have been developed for GE14 and ATRIUM 10XM fuel use at Monticello. The radial and nodal power uncertainties used in the analysis include the combined effects of a 60 day TIP interval (48 days + 25% grace), one TIP machine out-of-service, and an LPRM calibration interval of 1000 MWd/ST.

Based on the channel bow statistics provided in Table 1 of Reference 7, [

]

[

]

1.3. RESULTS

Analyses performed with the Cycle 30 core design support the following MCPR safety limits:

- Two-loop operation in the MELLLA region or when the ratio of core power to core flow is < 42 MWt / Mlbm/hr in the EFW: a MCPR safety limit of 1.08 [
  ]

- Two-loop operation when the ratio of core power to core flow is ≥ 42 MWt / Mlbm/hr in the EFW: a MCPR safety limit of 1.14 [ ] (this SLMCPR includes a 0.03 penalty to account for power distribution uncertainties beyond those inherent in the Framatome safety analysis methodology). Prior to the 0.03 penalty, the SLMCPR for ≥ 42 MWt / Mlbm/hr in the EFW is 0.03 higher than the SLMCPR for < 42 MWt / Mlbm/hr in the EFW.
- Single-loop operation: a MCPR safety limit of 1.12 [ ] and a MCPR safety limit of 1.13 [ ] Single-loop operation is not allowed within the EFW, therefore the 0.03 penalty does not apply.

Table 1 Uncertainties Used in SLMCPR Analysis

Parameter	Standard Deviation	Reference
Reactor System Related Uncertainties		
Feedwater flow rate	1.8%	3, Item 7.1.1 and Comment 7-1
Feedwater temperature	0.8%	3, Item 7.1.2 and Comment 7-1
Core pressure	0.8%	3, Item 7.1.3 and Comment 7-1
Total core flow rate		
Two-loop	2.5%	3, Item 7.1.4
Single-loop	6.0%	3, Item 7.1.5
Core Monitoring Uncertainties		
[		
		]

* These uncertainties are based on a 60 day TIP interval (48 days + 25% grace), one traversing incore probe (TIP) out-of-service (OSS), and an LPRM calibration interval of 1000 MWd/ST, Reference 9.

Table 2 Results Summary of SLMCPR Analysis

	Operating Condition	SLMCPR	Number of Rods in Boiling Transition*	Percentage of Rods in Boiling Transition*
TLO	ratio of core power to core flow < 42 MWt / Mlbm/hr in the EFW	1.08	33	0.0746
TLO	ratio of core power to core flow ≥ 42 MWt / Mlbm/hr in the EFW	1.14†	36	0.0814
SLO		1.13	34	0.0769
		1.12	41	0.0927

* The acceptance criterion is ≤0.10% rods in BT. The total number of heated rods in the core = 44212. Therefore, the maximum allowable number of BT rods is 44.

† The SLMCPR includes a 0.03 penalty to account for power distribution uncertainties beyond those inherent in the Framatome safety analysis methodology.

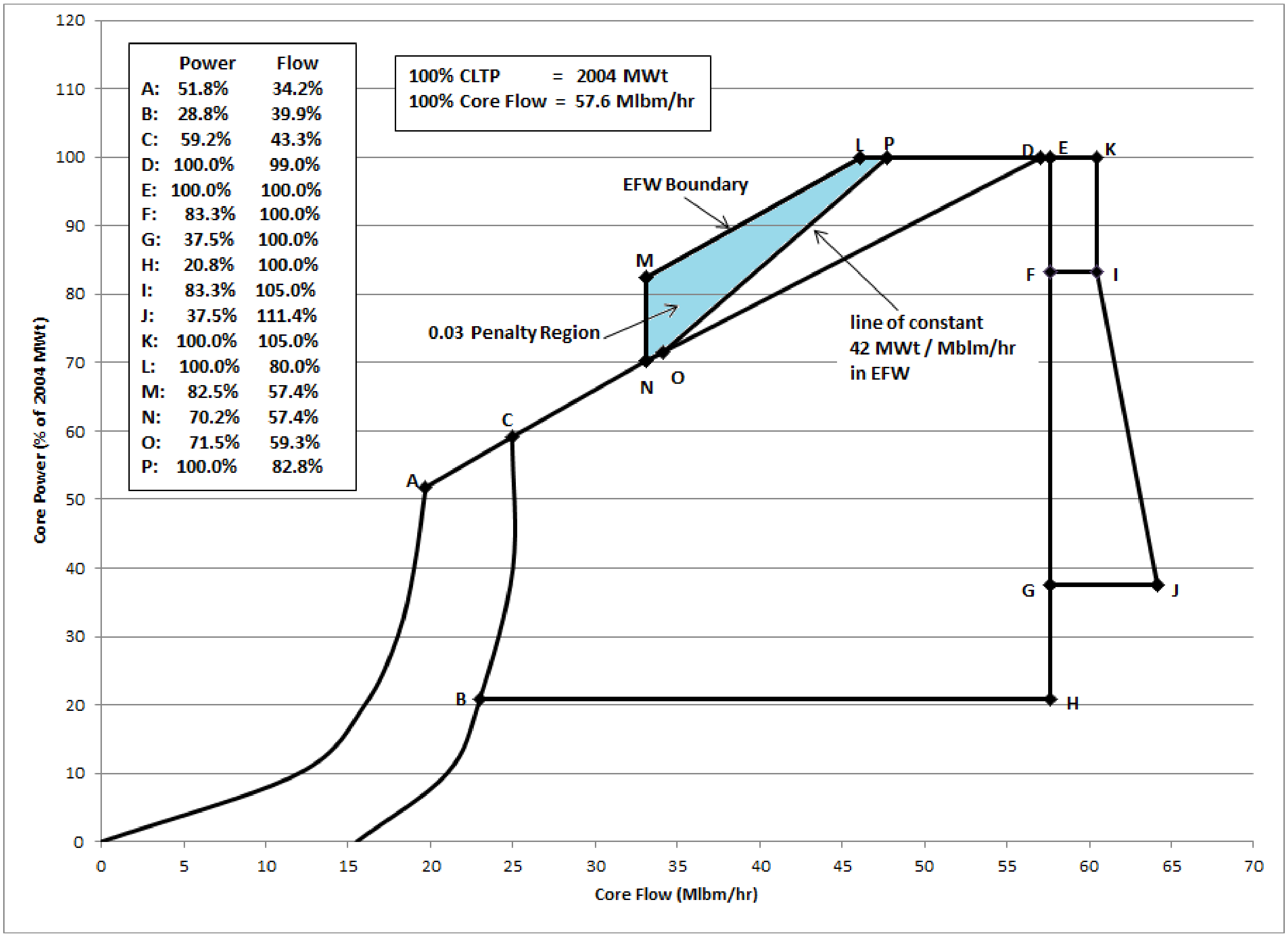

Figure 1 Monticello Power / Flow Map

2. REFERENCES

1. ANP-10307PA Revision 0, *AREVA MCPR Safety Limit Methodology for Boiling Water Reactors*, AREVA NP, June 2011.

2. ANP-3224P Revision 2, *Applicability of AREVA NP BWR Methods to Monticello*, AREVA NP, June 2013.

3. ANP-3666P Revision 0, *Monticello Cycle 30 Plant Parameters Document*, Framatome, June 2018.

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

5. EMF-2209(P)(A) Revision 3, *SPCB Critical Power Correlation*, AREVA NP, September 2009.

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

7. Letter, D Lutz (GNF) to D Mienke (Xcel), "Monticello's Transition to Areva Fuel, Item 7, Fuel Channel Bow Data," eDRF 0000-0143-5145, January 18, 2012 (38-2200963-000).

8. FS1-0027113 Revision 1.0, *Assembly Flow Uncertainty in Monticello When the Core is Monitored with GARDEL*, AREVA, September 2016.

9. FS1-0037377 Revision 1.0, *Monticello Calculation Plan for Cycle 30*, Framatome, June 2018.

ENCLOSURE 3

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST: REVISE THE SAFETY LIMIT MINIMUM CRITICAL POWER RATIO IN REACTOR CORE SAFETY LIMIT 2.1.1

AFFIDAVIT FOR FRAMATOME REPORT FS1-0039792, REVISION 1.0, MONTICELLO CYCLE 30 SAFLIM3D MCPR SAFETY LIMIT RESULTS (PROPRIETARY VERSION)

3 pages follow

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

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 FS1-0039792, Revision 1, "Monticello Cycle 30 SAFLIM3D MCPR Safety Limit Results (Proprietary Version)," dated September 2018 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."

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(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.

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(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.

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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.

SUBSCRIBED before me this 21st day of September, 2018.

Hailey M Siekawitch
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 9/28/2020

HAILEY M SIEKAWITCH
Commission Expires
NOTARY
PUBLIC
Sept. 28, 2020
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