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Monticello Nuclear Generating Plant
2807 W County Road 75
Monticello, MN 55362

**WITHHOLD ENCLOSURE 6 FROM PUBLIC DISCLOSURE
UNDER 10 CFR 2.390**

September 17, 2010

L-MT-10-055
10 CFR 50.90

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

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

License Amendment Request: Revise the Minimum Critical Power Ratio Safety Limit in Reactor Core Safety Limit 2.1.1.2

Pursuant to 10 CFR 50.90, Northern States Power Company – Minnesota (NSPM), proposes to revise the values for the Minimum Critical Power Ratio Safety Limit (MCPR Safety Limit) in Reactor Core Safety Limit 2.1.1.2. This proposed change provides revised values for the MCPR Safety Limit for both single and two recirculation loop operation in the Monticello Nuclear Generating Plant (MNGP) Technical Specifications (TS).

Enclosure 1 provides a description of the proposed changes and includes the technical evaluation and associated no significant hazards determination and environmental evaluation. Enclosure 2 provides a marked-up copy of the TS page showing the proposed changes.

Enclosure 3 provides a non-proprietary summary of the technical bases for this change to the MCPR Safety Limit values provided by Global Nuclear Fuel – Americas, LLC. Enclosure 4 provides the Monticello Power / Flow Maps for Cycles 25 and 26. A proprietary version of the summary of the technical bases for this change is included in Enclosure 6. An affidavit attesting to the proprietary nature of this information is provided in Enclosure 5 in accordance with 10 CFR 2.390(b)(1) requesting that this information be withheld from public disclosure.

NSPM requests approval of this proposed license amendment request by March 2011, with an implementation period to coincide with startup from the MNGP spring 2011 refueling outage.

The MNGP Plant Operations Review Committee has reviewed this application. In accordance with 10 CFR 50.91, a copy of this application, with enclosures, is being provided to the designated Minnesota Official.

Should you have questions regarding this letter, please contact Mr. Richard Loeffler at (763) 295-1247.

Summary of Commitments

This letter proposes no new commitments and does not revise any existing commitments.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on September 17, 2010.

Timothy J. O'Connor
Site Vice President, Monticello Nuclear Generating Plant
Northern States Power Company – Minnesota

Enclosures (6)

cc: Administrator, Region III, USNRC
Project Manager, Monticello, USNRC
Resident Inspector, Monticello, USNRC (w/o Enclosure 6)
Minnesota Department of Commerce (w/o Enclosure 6)

ENCLOSURE 1

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST

REVISE THE MINIMUM CRITICAL POWER RATIO SAFETY LIMIT IN REACTOR CORE SAFETY LIMIT 2.1.1.2

DESCRIPTION OF CHANGES

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DESCRIPTION OF CHANGES

**LICENSE AMENDMENT REQUEST
REVISE THE MINIMUM CRITICAL POWER RATIO SAFETY LIMIT
IN REACTOR CORE SAFETY LIMIT 2.1.1.2**

1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, Northern States Power Company – Minnesota (NSPM), proposes to revise the values for the Minimum Critical Power Ratio Safety Limit (MCPR Safety Limit) in Reactor Core Safety Limit 2.1.1.2.

2.0 DETAILED DESCRIPTION

The proposed change involves revising the MCPR Safety Limits contained in Technical Specification (TS) Reactor Core Safety Limit Specification 2.1.1.2 for two recirculation loop operation and single recirculation loop operation. The MCPR Safety Limit for two recirculation loop operation is proposed to be revised from 1.10 to 1.15. The MCPR Safety Limit for single recirculation loop operation would be revised from 1.12 to 1.15. The changes to the MCPR Safety Limits are due to the results of cycle-specific analyses performed by Global Nuclear Fuel – Americas, LLC (GNF) for the Monticello Nuclear Generating Plant (MNGP) for the upcoming cycle.

The proposed changes to the MCPR Safety Limits are required for MNGP Cycle 26 operation scheduled to begin in the spring 2011.

Two sets of reload licensing analyses were performed; one for operation at the current licensed rated thermal power and one for Extended Power Uprate (EPU) / Maximum Extended Load Line Limit Analysis – Plus (MELLLA+) operation. Performance of two sets of reload licensing analyses was necessary due to the uncertainty in the timing of U. S. Nuclear Regulatory Commission (NRC) approval of the EPU and MELLLA+ license amendment requests (References 1 and 2, respectively) for the MNGP. This consideration required the current licensed rated thermal power reload analyses to be performed at the increased MCPR Safety Limit determined from the EPU / MELLLA+ reload licensing analyses.

3.0 PROPOSED CHANGES

It is proposed to revise the value of the MCPR Safety Limits for two and single recirculation loop operation in Reactor Core Safety Limit 2.1.1.2 as shown below.

2.1.1.2 With reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.10 for two recirculation loop operation or ≥ 1.12 for single recirculation loop operation.

Reactor Core Safety Limit 2.1.1.2 will now read:

2.1.1.2 With reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.15 for two recirculation loop operation or ≥ 1.15 for single recirculation loop operation.

A mark-up of the proposed TS changes is provided in Enclosure 2. No changes are necessary to the TS Bases for this license amendment request.

4.0 TECHNICAL ANALYSIS

The proposed TS change will revise the MCPR Safety Limits contained in TS Reactor Core Safety Limit 2.1.1.2 for two recirculation loop operation and single recirculation loop operation to reflect changes due to the cycle-specific analysis performed by GNF for the MNGP for the next cycle beginning with startup from the spring 2011 refueling outage.

The revised MCPR Safety Limits are calculated using the NRC approved methodologies described and referenced through the GNF proprietary Licensing Topical Report, NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (GESTAR) (Reference 3).

GNF performed the MNGP Cycle 26 MCPR Safety Limits calculation in accordance with the NRC approved methodologies and uncertainties listed in Section 1.0, "Methodology" of the GNF report for the MNGP entitled, "GNF Additional Information Regarding the Requested Changes to the Technical Specification SLMCPR, Monticello Cycle 26" (Reference 4). This report summarizes the methodology, inputs, and results for the changes to the Monticello two recirculation loop and single recirculation loop MCPR Safety Limits. The MNGP Cycle 26 core consists only of the GE14 fuel type. A non-proprietary and proprietary version of this GNF report are provided in Enclosures 3 and 6, respectively to this letter.

Enclosure 4 provides the Monticello Power / Flow Map for Cycles 25 and 26 referred to in Section 2.9, "Power /Flow Map," of the GNF report.

The Studsvik Scandpower GARDEL core monitoring software is used as the core monitoring computer system at the MNGP, and will continue to be utilized as such for Cycle 26. GNF performed the MCPR Safety Limit calculations applying the General Electric GETAB power distribution methodology and uncertainties (Reference 5) which bound the associated GARDEL uncertainties.

A safety assessment and summary discussion of the GARDEL uncertainties versus the GETAB uncertainties is provided in the following sections.

4.1 Safety Assessment of Proposed Changes

The purpose of the MCPR Safety Limit is to ensure that specified acceptable fuel design limits are not exceeded during steady state operation and analyzed transients. The fuel cladding is one of the physical barriers that separate the radioactive materials from the environment. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Fuel cladding perforations can result from thermal stresses, which can occur from reactor operation significantly above design conditions.

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 cladding damage could occur. Although it is recognized that the onset of transition boiling (OTB) would not result in damage to the BWR fuel rod cladding, the critical power at which boiling transition is calculated to occur has been adopted as a convenient and conservative 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 safety limit (or MCPR Safety Limit) is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9 percent 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 Safety Limits analysis establishes values that will ensure that during normal operation and during abnormal operational transients, at least 99.9 percent of the fuel rods in the core do not experience OTB.

The revised MCPR Safety Limit for the MNGP was determined using cycle-specific fuel and core parameters, with NRC approved methodology, as discussed in Enclosures 3 and 6. Analysis of the limiting abnormal operational transients provides the allowed operating conditions in terms of MCPR, of the core during the fuel cycle such that if an event were to occur, the transient MCPR would not be less than the MCPR Safety Limit. The MCPR Safety Limit values for two recirculation loop operation and single recirculation loop operation are being increased in accordance with NRC approved methodologies, which includes additional uncertainties associated with EPU / MELLLA+ operation.

No plant hardware or operational changes are required with this proposed change.

4.2 GARDEL Core Monitoring Software and GETAB Uncertainties

The MCPR Safety Limit calculations were performed by GNF applying the NRC approved General Electric GETAB power distribution methodology and uncertainties. These uncertainties are documented in NEDO-10958-A, "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application" (Reference 5).

The GARDEL core monitoring software is used as the core monitoring system at the MNGP. The GARDEL bundle power uncertainty matches the GETAB value of 4.3 percent. The GARDEL nodal power uncertainty is 5.7 percent compared to the GETAB value of 8.7 percent. Both of these GARDEL uncertainty values validate the use of the GETAB values in the GNF MCPR Safety Limit calculations.

This demonstrates that the GETAB power distribution methodology uncertainties are applicable to represent the GARDEL core monitoring system uncertainties.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Determination

In accordance with the requirements of 10 CFR 50.90, Northern States Power Company – Minnesota (NSPM) requests an amendment to the facility Renewed Operating License DPR-22, for the Monticello Nuclear Generating Plant (MNGP). It is proposed to revise the Minimum Critical Power Ratio Safety Limits (MCPR Safety Limits) contained in Technical Specification (TS) Reactor Core Safety Limit Specification 2.1.1.2 for two recirculation loop operation and single recirculation loop operation to reflect cycle-specific limits determined by the reload safety analysis for the next operating cycle.

NSPM has evaluated the proposed amendment in accordance with 10 CFR 50.91 against the standards in 10 CFR 50.92 and has determined that the operation of the MNGP in accordance with the proposed amendment presents no significant hazards. NSPM's evaluation against each of the criteria in 10 CFR 50.92 follows.

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

   Response: No.

   The basis of the MCPR Safety Limit is to ensure that during normal operation and during abnormal operational transients, at least 99.9 percent of all fuel rods in the core do not experience transition boiling if the limit is not violated. The revised MCPR Safety Limit values preserve the existing margin to transition boiling and probability of fuel damage is not increased. The derivation of the cycle specific MCPR Safety Limit values for incorporation into the TS, and their use to determine cycle-specific thermal limits, have been performed using the methodologies discussed in General Electric fuel licensing safety analysis report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (GESTAR).

   Licensing analyses are performed on the redesigned core with NRC approved methodologies to determine changes in the critical power ratio as a result of anticipated operational occurrences. These results are added to the MCPR Safety Limit values proposed herein to generate the MCPR Operating Limits in the MNGP Core Operating Limits Report (COLR). The COLR operating limits thus assure that the MCPR Safety Limit will not be exceeded during normal operation or anticipated operational occurrences. Postulated accidents are also analyzed to confirm NRC acceptance criteria are met.

   The proposed change to the MCPR Safety Limits does not directly or indirectly affect any plant system, equipment, component, or change the processes used to operate the plant. The revised MCPR Safety Limit values have no effect on the probability of an accident initiating event or transient.

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

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

   Response: No.

   The MCPR Safety Limits are numerical values, calculated to ensure that during normal operation and during abnormal operational transients, at least 99.9 percent of all fuel rods in the core do not experience transition boiling if the limit is not violated. The revised MCPR Safety Limits are calculated using NRC approved methodologies discussed in GESTAR. The proposed changes do not involve any new modes of operation, any changes to setpoints, or any plant modifications. This proposed change to the MCPR Safety Limit values does not directly or indirectly affect any plant system, equipment, or component and therefore does not affect the failure modes of any of these items.

   The revised MCPR Safety Limits have been shown to be acceptable for the next cycle of operation. The core operating limits will continue to be developed using NRC approved methods. The proposed MCPR Safety Limits and methods for establishing the core operating limits do not result in the creation of any new precursors to an accident. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

   Response: No.

   The revised MCPR Safety Limit values are calculated using the methodology discussed in GESTAR. The proposed changes do not alter any plant system, equipment, component, or the processes used to operate the plant, the proposed changes will not jeopardize or degrade the function or operation of any plant system or component governed by TSs. The proposed MCPR Safety Limit values do not involve a significant reduction in the margin of safety as currently defined in the TS Bases, because the MCPR Safety Limits calculated for the upcoming cycle preserve the required margin of safety.

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

Based on the above, NSPM has determined that operation of the facility in accordance with the proposed change does not involve a significant hazards consideration as defined in 10 CFR 50.92(c), in that it does not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

5.2 Applicable Regulatory Requirements

10 CFR 50.36, "Technical specifications," provides the regulatory requirements for the content required in the TSs which includes safety limits. The proposed changes revise the MCPR Safety Limits contained in Technical Specification (TS) Reactor Core Safety Limit Specification 2.1.1.2 for two recirculation loop operation and single recirculation loop operation to reflect cycle-specific limits determined by the reload safety analysis for the next operating cycle.

The MNGP was designed largely before the publishing of the 70 General Design Criteria for Nuclear Power Plant Construction Permits proposed by the Atomic Energy Commission (AEC) for public comment in July 1967, and constructed prior to the 1971 publication of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50. As such, the MNGP was not licensed to the Appendix A, General Design Criteria (GDC).

The MNGP Updated Safety Analysis Report (USAR), Section 1.2, lists the principal design criteria (PDC) for the design, construction and operation of the plant. USAR Appendix E provides a plant comparative evaluation to the 70 proposed AEC design criteria. It was concluded that the plant conforms to the intent of the GDC. The applicable GDC and PDC are discussed below.

- PDC 1.2.2 -- Reactor Core

  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 MNGP reload analyses are performed by GNF in accordance with the codes and methods discussed in the GESTAR licensing topical report and the following GDC are applicable under that basis.

- GDC 10 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.

Also, 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.

NSPM has evaluated the proposed changes against the applicable regulatory requirements and acceptance criteria. The technical analysis concludes that the proposed TS changes will continue to assure that the design requirements and acceptance criteria of MNGP reload safety analyses are met. Based on this, there is reasonable assurance that the health and safety of the public, following approval of this TS change, is unaffected.

6.0 ENVIRONMENTAL EVALUATION

NSPM has determined that the proposed amendment would not change a requirement with respect to installation or use of a facility or component located within the restricted area, as defined in 10 CFR 20, nor would it change an inspection or surveillance requirement in such a way that it does not meet the following criteria. The proposed amendment does not involve (i) a significant hazards consideration, or (ii) authorize a significant change in the types or a significant increase in the amounts of any effluent that may be released offsite, or (iii) result in a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for a categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, the NSPM concludes pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. NSPM to NRC, “License Amendment Request: Extended Power Uprate (TAC MD9990),” (L-MT-08-052) dated November 5, 2008.

2. NSPM to NRC, “License Amendment Request: Maximum Extended Load Line Limit Analysis Plus,” (L-MT-10-003) dated January 21, 2010.

3. Global Nuclear Fuel-Americas, LLC (GNF) Licensing Topical Report (LTR), NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (GESTAR II) and Supplement for the United States,” dated October 2007.

4. GNF Report, GNF-0000-0092-5692-R0-P and NP, “GNF Additional Information Regarding the Requested Changes to the Technical Specification SLMCPR, Monticello Cycle 26,” dated September 9, 2010.

5. NEDO-10958-A, "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," January 1977.

ENCLOSURE 2

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST

REVISE THE MINIMUM CRITICAL POWER RATIO SAFETY LIMIT
IN REACTOR CORE SAFETY LIMIT 2.1.1.2

MARKED-UP TECHNICAL SPECIFICATION PAGE

(1 page follows)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 <u>Reactor Core SLs</u>

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.15 for two recirculation loop operation or ≥ 1.15 for single recirculation loop operation.

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

2.1.2 <u>Reactor Coolant System Pressure SL</u>

Reactor steam dome pressure shall be ≤ 1332 psig.

2.2 SL VIOLATIONS

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

ENCLOSURE 3

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST

REVISE THE MINIMUM CRITICAL POWER RATIO SAFETY LIMIT IN REACTOR CORE SAFETY LIMIT 2.1.1.2

GNF ADDITIONAL INFORMATION REGARDING THE REQUESTED CHANGES TO THE TECHNICAL SPECIFICATION SLMCPR

MONTICELLO CYCLE 26

NON-PROPRIETARY INFORMATION

(23 pages follow)

9/7/2010

GNF-0000-0092-5692-R1-NP

eDRFSection: 0000-0092-5706-R1

GNF Additional Information Regarding the Requested Changes to the Technical Specification SLMCPR

Monticello Cycle 26

Proprietary Information Notice

This document is the GNF non-proprietary version of the GNF proprietary report. From the GNF proprietary version, the information denoted as GNF proprietary (enclosed in double brackets) was deleted to generate this version.

Important Notice Regarding Contents of this Report
Please Read Carefully

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1.0 Methodology

GNF performed the Safety Limit Minimum Critical Power Ratio (SLMCPR) calculation in accordance to NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel" (Revision 16) using the following NRC-approved methodologies and uncertainties:

- NEDC-32601P-A "Methodology and Uncertainties for Safety Limit MCPR Evaluations" (August 1999).
- NEDC-32694P-A "Power Distribution Uncertainties for Safety Limit MCPR Evaluations" (August 1999).
- NEDC-32505P-A "R-Factor Calculation Method for GE11, GE12 and GE13 Fuel" (Revision 1, July 1999).
- NEDO-10958-A "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application" (January 1977).

Table 2 identifies the actual methodologies used for the Monticello Cycle 25 and the Cycle 26 SLMCPR calculations.

2.0 Discussion

In this discussion, the TLO nomenclature is used for two recirculation loops in operation, and the SLO nomenclature is used for one recirculation loop in operation.

2.1. Major Contributors to SLMCPR Change

In general, the calculated safety limit is dominated by two key parameters: (1) flatness of the core bundle-by-bundle MCPR distribution, and (2) flatness of the bundle pin-by-pin power/R-Factor distribution. Greater flatness in either parameter yields more rods susceptible to boiling transition and thus a higher calculated SLMCPR. MIP (MCPR Importance Parameter) measures the core bundle-by-bundle MCPR distribution and RIP (R-Factor Importance Parameter) measures the bundle pin-by-pin power/R-Factor distribution. The impact of the fuel loading pattern on the calculated TLO SLMCPR using rated core power and rated core flow conditions has been correlated to the parameter MIPRIP, which combines the MIP and RIP values.

Table 3 presents the MIP and RIP parameters for Cycle 25 and Cycle 26 along with the TLO SLMCPR estimate using the MIPRIP correlation. If the minimum core flow case is applicable, the TLO SLMCPR estimate is also provided for that case although the MIPRIP correlation is only applicable to the rated core flow case. If the off-rated power case (82.5% rated core power and 57.4% rated core flow) is applicable, the TLO SLMCPR estimate is also provided for that case although the MIPRIP correlation is only applicable to the rated core power and rated core flow case. This is done only to provide some reasonable assessment basis of the minimum core flow case trend. In addition, Table 3 presents estimated impacts on the TLO SLMCPR due to methodology deviations, penalties, and/or uncertainty deviations from approved values. Based on the MIPRIP correlation and any impacts due to deviations from approved values, a final estimated TLO SLMCPR is determined. Table 3 also provides the actual calculated Monte Carlo SLMCPRs. Given the bias and uncertainty in the MIPRIP correlation [[ ]] and the inherent variation in the Monte Carlo results [[ ]], the change in the Monticello Cycle 26 calculated Monte Carlo TLO SLMCPR using rated core power and rated core flow conditions is consistent with the corresponding estimated TLO SLMCPR value.

2.2. Deviations in NRC-Approved Uncertainties

Tables 4 and 5 provide a list of NRC-approved uncertainties along with values actually used. A discussion of deviations from these NRC-approved values follows; all of which are conservative relative to NRC-approved values. Also, estimated impact on the SLMCPR is provided in Table 3 for each deviation.

2.2.1. R-Factor

At this time, GNF has generically increased the GEXL R-Factor uncertainty from [[ ]] to account for an increase in channel bow due to the emerging unforeseen phenomena called control blade shadow corrosion-induced channel bow, which is not accounted for in the channel bow uncertainty component of the approved R-Factor uncertainty. The step "σ RPEAK" in Figure 4.1 from NEDC-32601P-A, which has been provided for convenience in Figure 3 of this attachment, is affected by this deviation. Reference 4 technically justifies that a GEXL R-Factor uncertainty of [[ ]] accounts for a channel bow uncertainty of up to [[ ]].

Currently, Monticello has not experienced any control blade shadow corrosion-induced channel bow and is not expected to experience any in Cycle 26 to the extent that would invalidate the approved R-Factor uncertainty.

2.2.2. Core Flow Rate and Random Effective TIP Reading

At this time, GNF has not been able to show that the NRC-approved process to calculate the SLMCPR only at the rated core power and rated core flow condition is adequately bounding relative to the SLMCPR calculated at rated core power and minimum core flow, see Reference 5. The minimum core flow condition can be more limiting due to the control rod pattern used. GNF has modified the NRC-approved process for determining the SLMCPR to include analyses at the rated core power and minimum licensed core flow point in addition to analyses at the rated core power and rated core flow point. GNF believes this modification is conservative and may in the future provide justification that the original NRC-approved process is adequately bounding.

For the TLO calculations performed at 80.0% core flow, the approved uncertainty values for the core flow rate (2.5%) and the random effective TIP reading (1.2%) are conservatively adjusted by using the SLO uncertainty values of 6.0% and 2.85% for the core flow rate and random effective TIP reading respectively. The most limiting SLMCPR calculation is performed at 82.5% rated core power and 57.4% core flow; the approved uncertainty values for the core flow rate (2.5%) and the random effective TIP reading (1.2%) are conservatively adjusted by using the SLO uncertainty values of 6.0% and 2.85% for the core flow rate and random effective TIP reading respectively. The steps "σ CORE FLOW" and "σ TIP (INSTRUMENT)" in Figure 4.1 from NEDC-32601P-A, which has been provided for convenience in Figure 3 of this attachment, are affected by this deviation, respectively.

2.3. Departure from NRC-Approved Methodology

No departures from NRC-approved methodologies were used in the Monticello Cycle 26 SLMCPR calculations.

2.4. Fuel Axial Power Shape Penalty

At this time, GNF has determined that higher uncertainties and non-conservative biases in the GEXL correlations for the various types of axial power shapes (i.e., inlet, cosine, outlet and double hump) could potentially exist relative to the NRC-approved methodology values, see References 3, 6, 7 and 8. The following table identifies, by marking with an "X", this potential for each GNF product line currently being offered:

[[

]]

Axial bundle power shapes corresponding to the limiting SLMCPR control blade patterns are determined using the PANACEA 3D core simulator. These axial power shapes are classified in accordance to the following table:

[[

]]

If the limiting bundles in the SLMCPR calculation exhibit an axial power shape identified by this table, GNF penalizes the GEXL critical power uncertainties to conservatively account for the impact of the axial power shape. Table 6 provides a list of the GEXL critical power uncertainties determined in accordance to the NRC-approved methodology contained in NEDE-24011-P-A along with values actually used.

For the limiting bundles, the fuel axial power shapes in the SLMCPR analysis were examined to determine the presence of axial power shapes identified in the above table. These power shapes were not found; therefore, no power shape penalties were applied to the calculated Monticello Cycle 26 SLMCPR values.

2.5. Methodology Restrictions

The four restrictions identified on Page 3 of NRC's Safety Evaluation relating to the General Electric Licensing Topical Reports NEDC-32601P, NEDC-32694P, and Amendment 25 to NEDE-24011-P-A (March 11, 1999) are addressed in References 1, 2, 3, and 9.

No new GNF fuel designs are being introduced in Monticello Cycle 26; therefore, the NEDC-32505P-A statement "...if new fuel is introduced, GENE must confirm that the revised R-Factor method is still valid based on new test data" is not applicable.

2.6. Minimum Core Flow Condition

For Monticello Cycle 26, the minimum core flow SLMCPR calculation performed at 80.0% core flow and rated core power condition was limiting as compared to the rated core flow and rated core power condition. The most limiting SLMCPR calculation was performed at the 82.5% rated core power and 57.4% core flow. At low core flows, the search spaces for the limiting rod pattern and the nominal rod pattern are essentially the same. Additionally, the condition that MIP [[ ]] establishes a reasonably bounding limiting rod pattern. Hence, the rod pattern used to calculate the SLMCPR at 82.5% rated power/57.4% rated flow reasonably assures that at least 99.9% of the fuel rods in the core would not be expected to experience boiling transition during normal operation or anticipated operational occurrences during the operation of Monticello Cycle 26. Consequently, the SLMCPR value calculated from the 57.4% core flow and 82.5% rated core power condition limiting MCPR distribution reasonably bounds this mode of operation for Monticello Cycle 26.

2.7. Limiting Control Rod Patterns

The limiting control rod patterns used to calculate the SLMCPR reasonably assures that at least 99.9% of the fuel rods in the core would not be expected to experience boiling transition during normal operation or anticipated operational occurrences during the operation of Monticello Cycle 26.

2.8. Core Monitoring System

The utility has requested GNF to perform the SLMCPR calculation applying the GETAB power distribution methodology and uncertainties. The utility has provided documents to GNF stating that the GETAB power distribution uncertainties bound those of the GARDEL core monitoring system used at Monticello.

2.9. Power/Flow Map

The utility has provided the current and previous cycle power/flow map in a separate attachment.

2.10. Core Loading Diagram

Figures 1 and 2 provide the core-loading diagram for the current and previous cycle respectively, which are the Reference Loading Pattern as defined by NEDE-24011-P-A. Table 1 provides a description of the core.

2.11. Figure References

Figure 3 is Figure 4.1 from NEDC-32601P-A. Figure 4 is Figure III.5-1 from NEDC-32601P-A. Figure 5 is Figure III.5-2 from NEDC-32601P-A.

2.12. Additional SLMCPR Licensing Conditions

For Monticello Cycle 26, the additional SLMCPR licensing condition that the SLMCPR shall be established by adding 0.03 (per reference 10) to the cycle-specific SLMCPR value calculated using the NRC-approved methodologies documented in NEDE-24011-P-A has been applied (see Table 3).

2.13. Summary

The requested changes to the Technical Specification SLMCPR values are 1.15 for TLO and 1.15 for SLO for Monticello Cycle 26.

3.0 References

1. Letter, Glen A. Watford (GNF-A) to U.S. Nuclear Regulatory Commission Document Control Desk with attention to R. Pulsifer (NRC), "Confirmation of 10x10 Fuel Design Applicability to Improved SLMCPR, Power Distribution and R-Factor Methodologies", FLN-2001-016, September 24, 2001.

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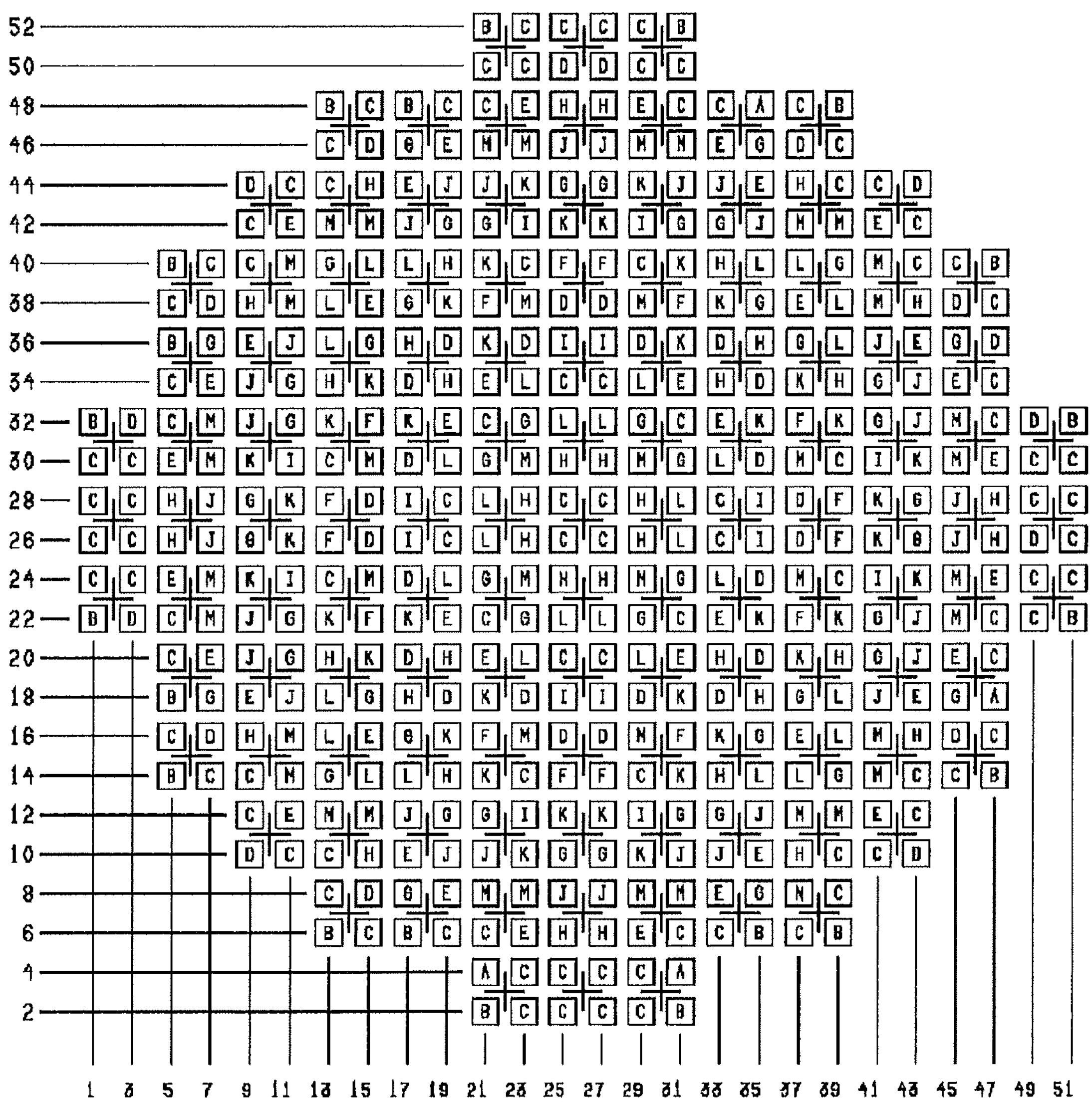
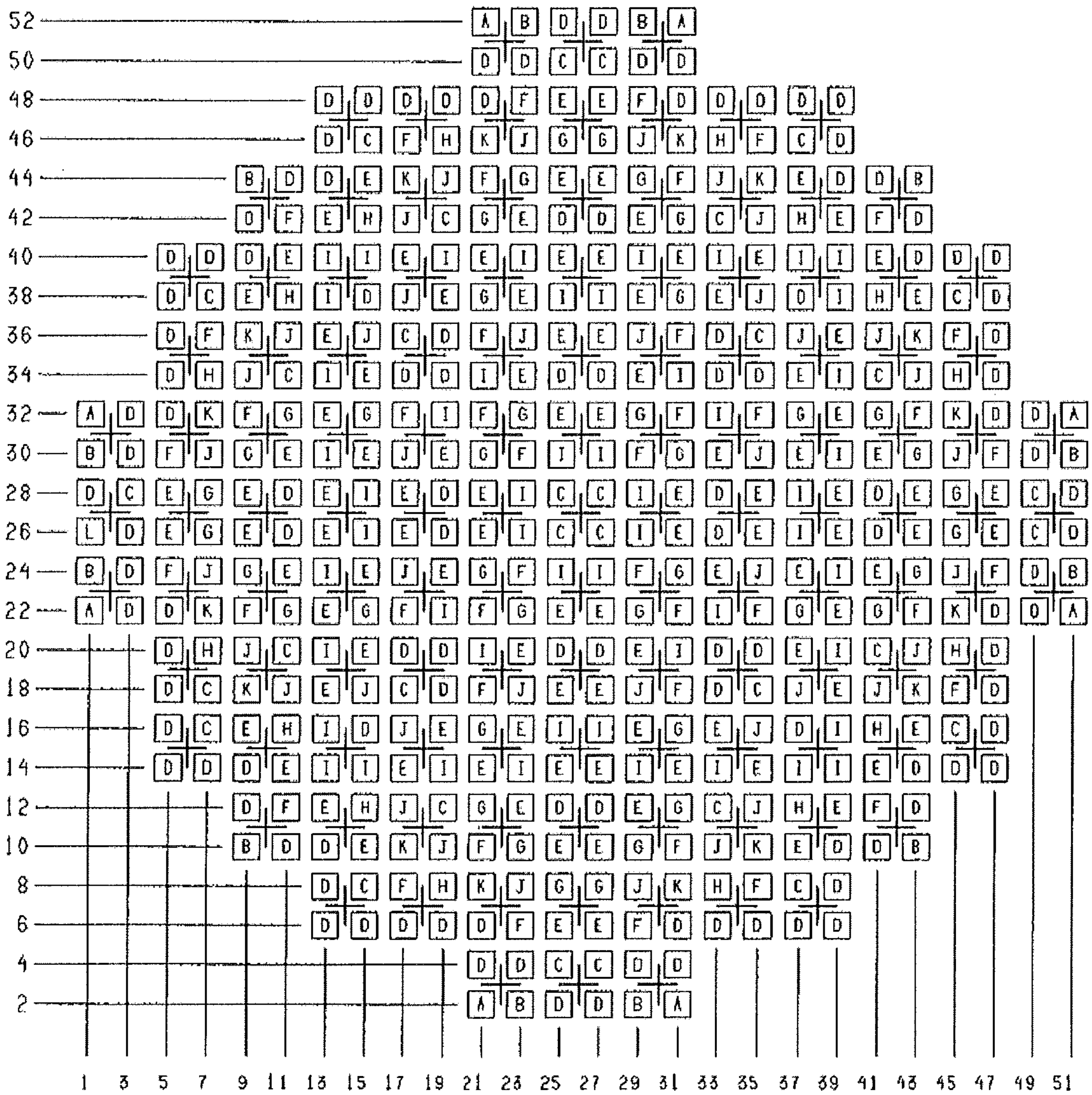


FUEL TYPE	
A = GE14-P10DNAB393-17GZ-100T-145-T6-2599	H = GE14-P10DNAB392-16GZ-100T-145-T6-3102
B = GE14-P10DNAB392-16GZ-100T-145-T6-2824	I = GE14-P10DNAB391-12GZ-100T-145-T6-3103
C = GE14-P10DNAB392-16GZ-100T-145-T6-2931	J = GE14-P10DNAB373-16GZ-100T-145-T6-3375
D = GE14-P10DNAB392-17GZ-100T-145-T6-2932	K = GE14-P10DNAB391-16GZ-100T-145-T6-3376
E = GE14-P10DNAB392-16GZ-100T-145-T6-2931	L = GE14-P10DNAB391-15GZ-100T-145-T6-3377
F = GE14-P10DNAB424-14GZ-100T-145-T6-3100	M = GE14-P10DNAB391-12GZ-100T-145-T6-3378
G = GE14-P10DNAB375-16GZ-100T-145-T6-3101	N = GE14-P10DNAB392-17GZ-100T-145-T6-2932

Figure 1. Current Cycle Core Loading Diagram



FUEL TYPE	
A = GE14-P10DNAB393-17GZ-100T-145-T6-2598	G = GE14-P10DNAB392-16GZ-100T-145-T6-2931
B = GE14-P10DNAB393-17GZ-100T-145-T6-2599	H = GE14-P10DNAB424-14GZ-100T-145-T6-3100
C = GE14-P10DNAB393-17GZ-100T-145-T6-2599	I = GE14-P10DNAB375-16GZ-100T-145-T6-3101
D = GE14-P10DNAB392-16GZ-100T-145-T6-2824	J = GE14-P10DNAB392-16GZ-100T-145-T6-3102
E = GE14-P10DNAB392-16GZ-100T-145-T6-2931	K = GE14-P10DNAB391-12GZ-100T-145-T6-3103
F = GE14-P10DNAB392-17GZ-100T-145-T6-2932	L = GE14-P10DNAB391-14GZ-100T-145-T6-2480

Figure 2. Previous Cycle Core Loading Diagram

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Figure 3. Figure 4.1 from NEDC-32601P-A

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Figure 4. Figure III.5-1 from NEDC-32601P-A

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Figure 5. Figure III.5-2 from NEDC-32601P-A

Table 1. Description of Core

Description	Cycle 25 1775 MWt 47.46 Mlb/hr	Cycle 25 1775 MWt 57.60 Mlb/hr	Cycle 26 1653 MWt 33.06 Mlb/hr	Cycle 26 2004 MWt 46.08 Mlb/hr	Cycle 26 2004 MWt 57.60 Mlb/hr	Cycle 26 2004 MWt 60.48 Mlb/hr
Number of Bundles in the Core	484		484			
Limiting Cycle Exposure Point (i.e. BOC/MOC/EOC)	BOC	EOC	EOC	BOC	BOC	BOC
Cycle Exposure at Limiting Point (MWd/STU)	0	11500	10500	200	200	200
% Rated Core Power	100.0	100.0	82.5	100.0	100.0	100.0
% Rated Core Flow	82.4	100.0	57.4	80.0	100.0	105.0
Reload Fuel Type	GE14		GE14			
Latest Reload Batch Fraction, %	33.9		30.6			
Latest Reload Average Batch Weight % Enrichment	3.90		3.87			
Core Fuel Fraction: GE14	100.0		100.0			
Core Average Weight % Enrichment	3.91		3.90			

Table 2. SLMCPR Calculation Methodologies

Description	Cycle 25	Cycle 26
Non-power Distribution Uncertainty	NEDC-32601P-A	NEDC-32601P-A
Power Distribution Methodology	NEDO-10958-A	NEDO-10958-A
Power Distribution Uncertainty	NEDO-10958-A	NEDO-10958-A
Core Monitoring System	GARDEL	GARDEL
R-Factor Calculation Methodology	NEDC-32505P-A	NEDC-32505P-A

Table 3. Monte Carlo Calculated SLMCPR vs. Estimate

Description	**Cycle 25 1775 MWt 47.46 Mlb/hr**	**Cycle 25 1775 MWt 57.60 Mlb/hr**	**Cycle 26 1653 MWt 33.06 Mlb/hr**	**Cycle 26 2004 MWt 46.08 Mlb/hr**	**Cycle 26 2004 MWt 57.60 Mlb/hr**	**Cycle 26 2004 MWt 60.48 Mlb/hr**

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Table 3. Monte Carlo Calculated SLMCPR vs. Estimate

Description	Cycle 25 1775 MWt 47.46 Mlb/hr	Cycle 25 1775 MWt 57.60 Mlb/hr	Cycle 26 1653 MWt 33.06 Mlb/hr	Cycle 26 2004 MWt 46.08 Mlb/hr	Cycle 26 2004 MWt 57.60 Mlb/hr	Cycle 26 2004 MWt 60.48 Mlb/hr

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Table 4. Non-Power Distribution Uncertainties

Description	Nominal (NRC-Approved) Value ± σ (%)	Cycle 25 1775 MWt 47.46 Mlb/hr	Cycle 25 1775 MWt 57.60 Mlb/hr	Cycle 26 1653 MWt 33.06 Mlb/hr	Cycle 26 2004 MWt 46.08 Mlb/hr	Cycle 26 2004 MWt 57.60 Mlb/hr	Cycle 26 2004 MWt 60.48 Mlb/hr
GETAB							
Feedwater Flow Measurement	1.76	1.76	1.76	1.76	1.76	1.76	1.76
Feedwater Temperature Measurement	0.76	0.76	0.76	0.76	0.76	0.76	0.76
Reactor Pressure Measurement	0.50	0.50	0.50	0.50	0.50	0.50	0.50
Core Inlet Temperature Measurement	0.20	0.20	0.20	0.20	0.20	0.20	0.20
Total Core Flow Measurement	6.0 SLO 2.5 TLO	6.0 SLO 2.5 TLO	6.0 SLO 2.5 TLO	6.0 TLO	6.0 TLO	6.0 SLO 2.5 TLO	2.5 TLO
Channel Flow Area Variation	3.0	3.0	3.0	3.0	3.0	3.0	3.0
Friction Factor Multiplier	10.0	10.0	10.0	10.0	10.0	10.0	10.0
Channel Friction Factor Multiplier	5.0	5.0	5.0	5.0	5.0	5.0	5.0

Table 4. Non-Power Distribution Uncertainties

Description	Nominal (NRC-Approved) Value ± σ (%)	Cycle 25 1775 MWt 47.46 Mlb/hr	Cycle 25 1775 MWt 57.60 Mlb/hr	Cycle 26 1653 MWt 33.06 Mlb/hr	Cycle 26 2004 MWt 46.08 Mlb/hr	Cycle 26 2004 MWt 57.60 Mlb/hr	Cycle 26 2004 MWt 60.48 Mlb/hr
NEDC-32601P-A							
Feedwater Flow Measurement	[[ ]]	N/A	N/A	N/A	N/A	N/A	N/A
Feedwater Temperature Measurement	[[ ]]	N/A	N/A	N/A	N/A	N/A	N/A
Reactor Pressure Measurement	[[ ]]	N/A	N/A	N/A	N/A	N/A	N/A
Core Inlet Temperature Measurement	0.20	N/A	N/A	N/A	N/A	N/A	N/A
Total Core Flow Measurement	6.0 SLO/2.5 TLO	N/A	N/A	N/A	N/A	N/A	N/A
Channel Flow Area Variation	[[ ]]	N/A	N/A	N/A	N/A	N/A	N/A
Friction Factor Multiplier	[[ ]]	N/A	N/A	N/A	N/A	N/A	N/A
Channel Friction Factor Multiplier	5.0	N/A	N/A	N/A	N/A	N/A	N/A

Table 5. Power Distribution Uncertainties

Description	Nominal (NRC-Approved) Value ± σ (%)	Cycle 25 1775 MWt 47.46 Mlb/hr	Cycle 25 1775 MWt 57.60 Mlb/hr	Cycle 26 1653 MWt 33.06 Mlb/hr	Cycle 26 2004 MWt 46.08 Mlb/hr	Cycle 26 2004 MWt 57.60 Mlb/hr	Cycle 26 2004 MWt 60.48 Mlb/hr
GETAB/NEDC-32601P-A							
GEXL R-Factor	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]
Random Effective TIP Reading	2.85 SLO 1.2 TLO	2.85 SLO 1.2 TLO	2.85 SLO 1.2 TLO	2.85 TLO	2.85 TLO	2.85 SLO 1.2 TLO	1.2 TLO
Systematic Effective TIP Reading	8.6	8.6	8.6	8.6	8.6	8.6	8.6
NEDC-32694P-A, 3DMONICORE							
GEXL R-Factor	[[ ]]	N/A	N/A	N/A	N/A	N/A	N/A
Random Effective TIP Reading	2.85 SLO 1.2 TLO	N/A	N/A	N/A	N/A	N/A	N/A
TIP Integral	[[ ]]	N/A	N/A	N/A	N/A	N/A	N/A
Four Bundle Power Distribution Surrounding TIP Location	[[ ]]	N/A	N/A	N/A	N/A	N/A	N/A

Table 6. Power Distribution Uncertainties

Description	Nominal (NRC-Approved) Value ± σ (%)	Cycle 25 1775 MWt 47.46 Mlb/hr	Cycle 25 1775 MWt 57.60 Mlb/hr	Cycle 26 1653 MWt 33.06 Mlb/hr	Cycle 26 2004 MWt 46.08 Mlb/hr	Cycle 26 2004 MWt 57.60 Mlb/hr	Cycle 26 2004 MWt 60.48 Mlb/hr
Contribution to Bundle Power Uncertainty Due to LPRM Update	[[ ]]	N/A	N/A	N/A	N/A	N/A	N/A
Contribution to Bundle Power Due to Failed TIP	[[ ]]	N/A	N/A	N/A	N/A	N/A	N/A
Contribution to Bundle Power Due to Failed LPRM	[[ ]]	N/A	N/A	N/A	N/A	N/A	N/A
Total Uncertainty in Calculated Bundle Power	[[ ]]	N/A	N/A	N/A	N/A	N/A	N/A

Table 7. Critical Power Uncertainties

Description	Nominal (NRC-Approved) Value ± σ (%)	Cycle 25 1775 MWt 47.46 Mlb/hr	Cycle 25 1775 MWt 57.60 Mlb/hr	Cycle 26 1653 MWt 33.06 Mlb/hr	Cycle 26 2004 MWt 46.08 Mlb/hr	Cycle 26 2004 MWt 57.60 Mlb/hr	Cycle 26 2004 MWt 60.48 Mlb/hr

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<u>**ENCLOSURE 4**</u>

MONTICELLO NUCLEAR GENERATING PLANT

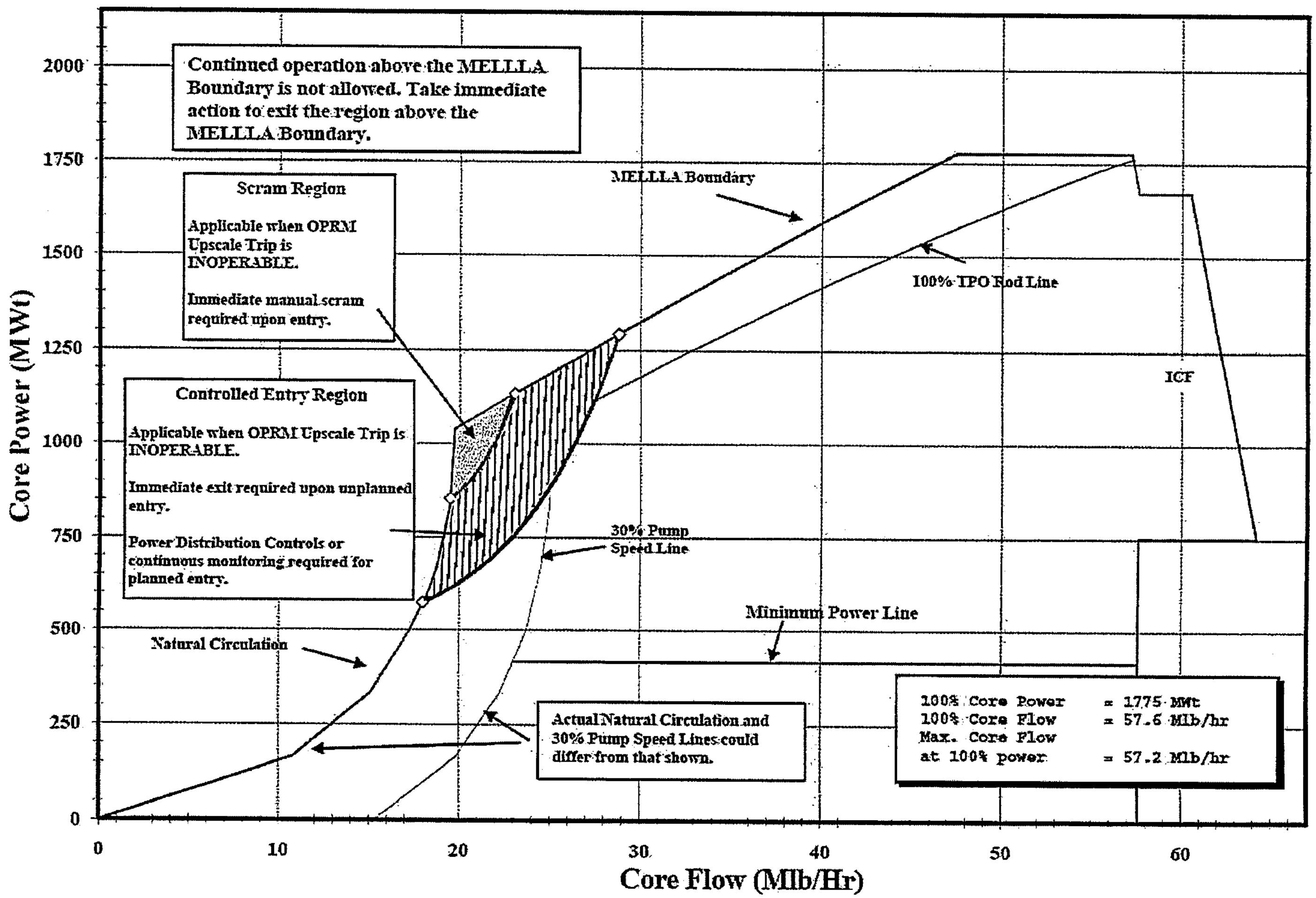
LICENSE AMENDMENT REQUEST

REVISE THE MINIMUM CRITICAL POWER RATIO SAFETY LIMIT IN REACTOR CORE SAFETY LIMIT 2.1.1.2

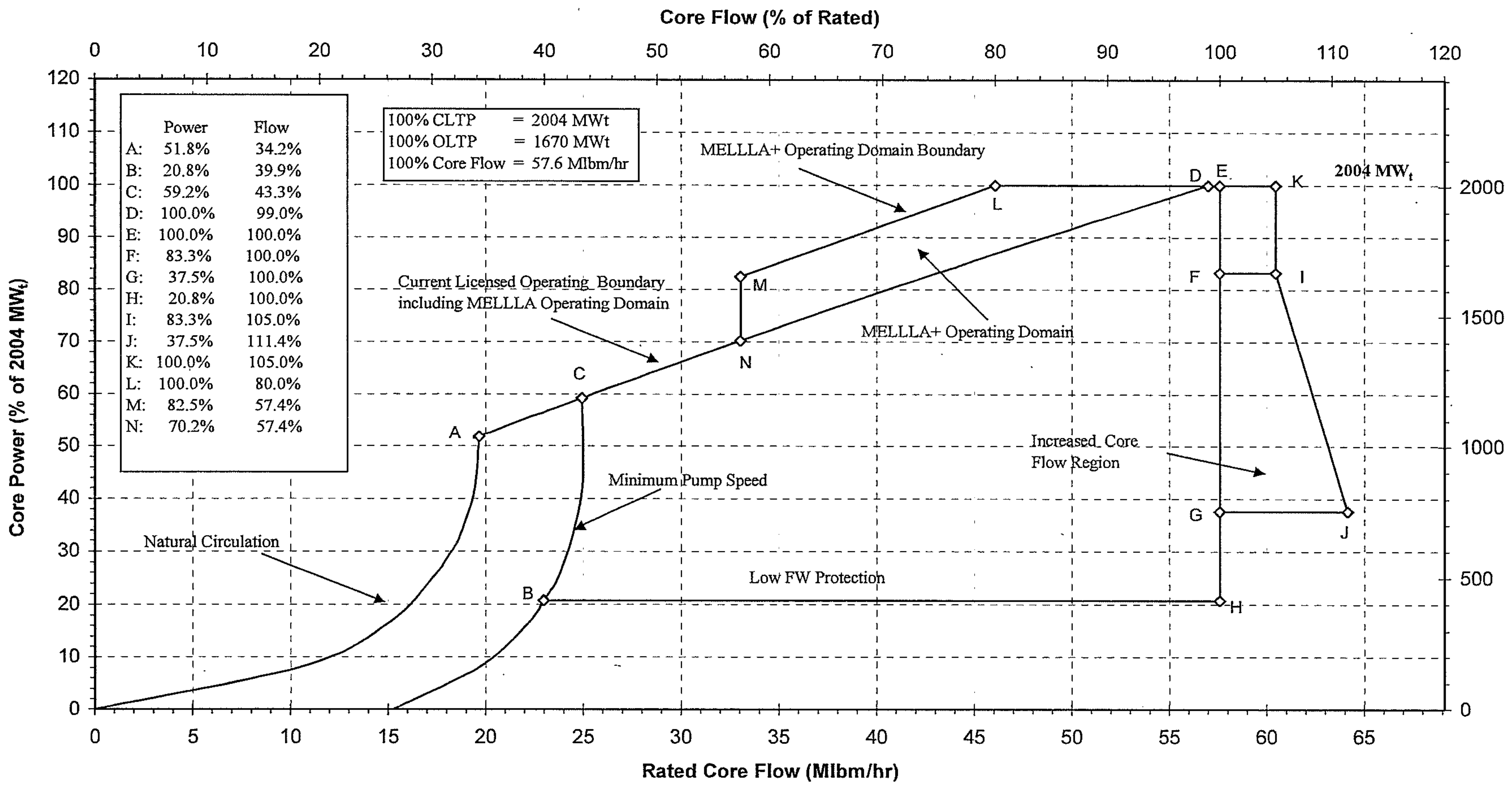
MONTICELLO POWER / FLOW MAPS FOR CYCLES 25 AND 26

(2 pages follow)

Monticello Cycle 25 Power/Flow Map



Power/Flow Operating Map for MELLLA+



ENCLOSURE 5

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST

REVISE THE MINIMUM CRITICAL POWER RATIO SAFETY LIMIT
IN REACTOR CORE SAFETY LIMIT 2.1.1.2

AFFIDAVIT FOR THE GNF ADDITIONAL INFORMATION REGARDING THE REQUESTED CHANGES TO THE TECHNICAL SPECIFICATION SLMCPR

MONTICELLO CYCLE 26

(3 pages follow)

Global Nuclear Fuel - Americas LLC

AFFIDAVIT

I, Anthony P. Reese, state as follows:

(1) I am the Reload Licensing Manager, Fuel Engineering, Global Nuclear Fuels-Americas, LLC ("GNF-A"), and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the GNF-A proprietary report, GNF-0000-0092-5692-R1-P, *GNF Additional Information Regarding the Requested Changes to the Technical Specification SLMCPR, Monticello Cycle 26*, Class III, (GNF-A Proprietary Information), dated 9/7/2010. GNF-A proprietary information in GNF-0000-0092-5706-R1-P is identified by a dark red dotted underline inside double square brackets. [[This sentence is an example.[3]]] Figures and large equation objects containing GNF-A proprietary information are identified with double square brackets before and after the object. In each case, the superscript notation [3] refers to Paragraph (3) of this affidavit that provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner or licensee, GNF-A relies upon the exemption from disclosure set forth in the Freedom of Information Act (FOIA), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for trade secrets (Exemption 4). The material for which exemption from disclosure is here sought also qualifies under the narrower definition of trade secret, within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975 F2d 871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704 F2d 1280 (DC Cir. 1983).

(4) The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b. Some examples of categories of information that fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GNF-A's competitors without license from GNF-A constitutes a competitive economic advantage over GNF-A and/or other companies.

b. Information that, if used by a competitor, would reduce their expenditure of resources or improve their competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

c. Information that reveals aspects of past, present, or future GNF-A customer-funded development plans and programs, that may include potential products of GNF-A.

d. Information that discloses trade secret and/or potentially patentable subject matter for which it may be desirable to obtain patent protection.

(5) To address 10 CFR 2.390(b)(4), the information sought to be withheld is being submitted to the NRC in confidence. The information is of a sort customarily held in confidence by GNF-A, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GNF-A, not been disclosed publicly, and not been made available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary and/or confidentiality agreements that provide for maintaining the information in confidence. The initial designation of this information as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure are as set forth in the following paragraphs (6) and (7).

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, who is the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or who is the person most likely to be subject to the terms under which it was licensed to GNF-A. Access to such documents within GNF-A is limited to a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist, or other equivalent authority for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GNF-A are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary and/or confidentiality agreements.

(8) The information identified in paragraph (2) above is classified as proprietary because it contains details of GNF-A's fuel design and licensing methodology for the Boiling Water Reactor (BWR). Development of these methods, techniques, and information and their application for the design, modification, and analyses methodologies and processes was achieved at a significant cost to GNF-A. The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GNF-A asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GNF-A's competitive position and foreclose or reduce the availability of profit-making opportunities. The fuel design and licensing methodology is part of GNF-A's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GNF-A. The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial. GNF-A's competitive advantage will be lost if its competitors are able to use the results of the GNF-A experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GNF-A would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GNF-A of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 7th day of September, 2010

Anthony P. Reese
Reload Licensing Manager, Fuel Engineering
Global Nuclear Fuel – Americas, LLC