UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 7, 2009

Mr. Timothy J. O'Connor
Site Vice President
Monticello Nuclear Generating Plant
Northern States Power Company - Minnesota
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - ISSUANCE OF AMENDMENT REGARDING RECIRCULATION RISER DIFFERENTIAL PRESSURE (TAC NO. MD6864)

Dear Mr. O'Connor:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 161 to Renewed Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The amendment consists of changes to the Technical Specifications in response to your application dated September 25, 2007, as supplemented by letters dated September 8, 2008, November 6, 2008, January 20, 2009 and April 2, 2009.

The amendment revised the allowable value and channel calibration frequency for Function 2.j, Recirculation Riser Differential Pressure - High Function (Break Detection), in Table 3.3.5.1-1, "Emergency Core Cooling System Instrumentation."

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Karl Feintuch for

Peter S. Tam, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosures:
1. Amendment No. 161 to DPR-22
2. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY - MINNESOTA*

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 161
License No. DPR-22

1. The U. S. Nuclear Regulatory Commission (the Commission) has found that:

    A. The application for amendment by Nuclear Management Company, LLC* (the licensee), dated September 25, 2007, as supplemented by letters dated September 8, November 6, 2008, January 20, 2009 and April 2, 2009, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

    B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

    C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

    D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

    E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.2 of Renewed Facility Operating License No. DPR-22 is hereby amended to read as follows:

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*On September 22, 2008, Nuclear Management Company, LLC, transferred its operating authority to its parent, Northern States Power Company - Minnesota (NSPM). By letter dated September 3, 2008 (Accession No. ML082470648), NSPM stated that it accepts responsibility for all actions before the NRC staff which were previously initiated or addressed by Nuclear Management Company.

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 161, are hereby incorporated in the license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Lois M. James

Lois M. James, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to Renewed Facility Operating License and Technical Specifications

Date of Issuance: April 7, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 161

RENEWED FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Replace the following page of Renewed Facility Operating License DPR-22 with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

REMOVE	INSERT
3	3

Replace the following pages of Appendix A, Technical Specifications, with the revised pages attached. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3.3.5.1-8	3.3.5.1-8
3.3.5.1-9	3.3.5.1-9

2. Pursuant to the Act and 10 CFR Part 70, NSPM to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operations, as described in the Final Safety Analysis Report, as supplemented and amended, and the licensee's filings dated August 16, 1974 (those portions dealing with handling of reactor fuel) and August 17, 1977 (those portions dealing with fuel assembly storage capacity);

3. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, NSPM to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

4. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, NSPM to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

5. Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to possess, but not separate, such byproduct and special nuclear material as may be produced by operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission, now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

1. Maximum Power Level

NSPM is authorized to operate the facility at steady state reactor core power levels not in excess of 1775 megawatts (thermal).

2. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 161 are hereby incorporated in the license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. Physical Protection

NSPM shall implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search

Table 3.3.5.1-1 (page 3 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System					
e. Reactor Steam Dome Pressure Permissive - Bypass Timer (Pump Permissive)	1, 2, 3	2	C	SR 3.3.5.1.7 SR 3.3.5.1.8	≥ 18 minutes and ≤ 22 minutes
	4[a], 5[a]	2	B	SR 3.3.5.1.7 SR 3.3.5.1.8	≥ 18 minutes and ≤ 22 minutes
f. Low Pressure Coolant Injection Pump Start - Time Delay Relay	1, 2, 3, 4[a], 5[a]	4 per pump	B	SR 3.3.5.1.7 SR 3.3.5.1.8	
Pumps A, B					≤ 5.33 seconds
Pumps C, D					≤ 10.59 seconds
g. Low Pressure Coolant Injection Pump Discharge Flow - Low (Bypass)	1, 2, 3, 4[a], 5[a]	1 per pump	E	SR 3.3.5.1.2 SR 3.3.5.1.7 SR 3.3.5.1.8	≥ 360 gpm and ≤ 745 gpm
h. Reactor Steam Dome Pressure - Low (Break Detection)	1, 2, 3,	4	B	SR 3.3.5.1.2 SR 3.3.5.1.7 SR 3.3.5.1.8	≥ 873.6 psig and ≤ 923.4 psig
i. Recirculation Pump Differential Pressure - High (Break Detection)	1, 2, 3	4 per pump	C	SR 3.3.5.1.2 SR 3.3.5.1.7 SR 3.3.5.1.8	≥ 63.5 inches wc
j. Recirculation Riser Differential Pressure - High (Break Detection)	1, 2, 3	4	C	SR 3.3.5.1.2 SR 3.3.5.1.7[c][d] SR 3.3.5.1.8	≤ 100.0 inches wc

(a) When associated ECCS subsystem(s) are required to be OPERABLE per LCO 3.5.2.

(c) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(d) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the nominal trip setpoint; otherwise, the channel shall be declared inoperable. The nominal trip setpoint and the methodology used to determine the as-found tolerance and the as-left tolerance are specified in the TRM.

Table 3.3.5.1-1 (page 4 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System					
k. Reactor Steam Dome Pressure - Time Delay Relay (Break Detection)	1, 2, 3	2	B	SR 3.3.5.1.7 SR 3.3.5.1.8 SR 3.3.5.1.9	≤ 2.97 seconds
l. Recirculation Pump Differential Pressure - Time Delay Relay (Break Detection)	1, 2, 3	2	C	SR 3.3.5.1.7 SR 3.3.5.1.8 SR 3.3.5.1.9	≤ 0.75 seconds
m. Recirculation Riser Differential Pressure - Time Delay Relay (Break Detection)	1, 2, 3	2	C	SR 3.3.5.1.7 SR 3.3.5.1.8 SR 3.3.5.1.9	≤ 0.75 seconds
3. High Pressure Coolant Injection (HPCI) System					
a. Reactor Vessel Water Level - Low Low	1, 2[(e)], 3[(e)]	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.7 SR 3.3.5.1.8	≥ -48 inches
b. Drywell Pressure - High	1, 2[(e)], 3[(e)]	4	B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.8	≤ 2 psig
c. Reactor Vessel Water Level - High	1, 2[(e)], 3[(e)]	2	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.7 SR 3.3.5.1.8	≤ 48 inches
d. Condensate Storage Tank Level - Low	1, 2[(e)], 3[(e)]	2	D	SR 3.3.5.1.7 SR 3.3.5.1.8	≥ 29.3 inches
e. Suppression Pool Water Level - High	1, 2[(e)], 3[(e)]	2	D	SR 3.3.5.1.5 SR 3.3.5.1.6 SR 3.3.5.1.8	≤ 3.0 inches
f. High Pressure Coolant Injection Pump Discharge Flow - Low (Bypass)	1, 2[(e)], 3[(e)]	1	E	SR 3.3.5.1.5 SR 3.3.5.1.6 SR 3.3.5.1.8	≥ 362 gpm and ≤ 849 gpm

(e) With reactor steam dome pressure > 150 psig.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 161 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY*

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

By application dated September 25, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML072760401), Nuclear Management Company, LLC (NMC*, the licensee), requested changes to the Technical Specifications (TS) for the Monticello Nuclear Generating Plant (MNGP). By letters dated September 8, November 6, 2008, January 20, 2009 and April 2, 2009 (ADAMS Accession Nos. ML082600347, ML083150026, ML090280053 and ML090930499), the licensee supplemented the original application. The licensee proposed to revise the allowable value and channel calibration frequency for the Recirculation Riser Differential Pressure - High Function (break detection), a setpoint used by the Low Pressure Coolant Injection (LPCI) loop select logic to reliably select the unbroken recirculation loop for LPCI injection. To justify the proposed amendment, the licensee submitted an analysis of the small-break loss-of-coolant accident (SBLOCA) which determined a new minimum detectable break size for the LPCI loop select logic.

The SBLOCA analysis reflected a methodology change, which considered the potential for axial power shape to influence the accident results. The licensee stated that this methodology change resulted in the SBLOCA being the limiting accident with respect to reactor fuel peak cladding temperature (PCT) rather than the large recirculation line break LOCA (LBLOCA). Therefore, in conjunction with the TS change, the licensee also requested to re-zero the accumulation of PCT changes and errors required by 10 CFR 50.46(a)(3) at the new value of 1990°F as determined by the analysis. However, in its November 6, 2008, letter, the licensee withdrew the request to re-zero the licensing basis PCT.

The LPCI loop select logic is designed to reliably detect the unbroken recirculation loop for LPCI injection. The logic is initiated upon the receipt of either a Reactor Vessel Water Level - Low signal or a Drywell Pressure - High signal. The selection of the unbroken recirculation loop is done by comparing the pressure of the two recirculation riser loops. The broken recirculation loop is indicated by a lower pressure than the unbroken loop. The recirculation loop with the higher pressure is used for LPCI injection.

*On September 22, 2008, Nuclear Management Company, LLC, transferred its operating authority to its parent, Northern States Power Company - Minnesota (NSPM). By letter dated September 3, 2008 (Accession No. ML082470648), NSPM stated that it accepts responsibility for all actions before the Nuclear Regulatory Commission staff which were previously initiated or addressed by Nuclear Management Company.

Enclosure

Recirculation riser differential pressure signals are used by the LPCI loop select logic to determine which, if any, recirculation loop is broken by comparing the pressure of the two recirculation loops. For a LBLOCA, the analysis assumes that the LPCI loop select logic successfully identifies and directs LPCI flow to the unbroken recirculation loop, so that core reflooding is accomplished in time to ensure that the PCT remains below the limits of 10 CFR 50.46. However, for a SBLOCA, there is a minimum break size where any break smaller in size would be beyond the measurement capability of the instrumentation to be reliably sensed by the LPCI loop select logic. Therefore, the SBLOCA analysis must assume that the broken loop is selected for LPCI injection for any break size less than the minimum detectable break size.

The licensee proposed to increase the minimum detectable break size of the LPCI loop select logic from the current 0.1 ft^2 to 0.4 ft^2. The licensee submitted an analysis of the SBLOCA ranging from 0.05 ft^2 to 0.5 ft^2 to justify the new allowable minimum detectable break size.

The licensee's September 8, November 6, 2008, January 20, 2009 and April 2, 2009, supplements provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on November 20, 2007 (72 FR 65368).

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, establishes the fundamental regulatory requirements with respect to the domestic licensing of nuclear production and utilization facilities. Specifically, Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 provides, in part, the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety.

The requirements of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," in combination with General Design Criteria (GDC) 35, "Emergency Core Cooling," and Appendix K, "ECCS Evaluation Models," were used by the NRC staff to evaluate the licensee's SBLOCA analysis.

GDC 13, "Instrumentation and Control," requires that instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

GDC 20, "Protective System Functions," requires the protection system be designed (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences; and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

In 10 CFR 50.36, "Technical Specifications," states, "Each applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed technical specifications in accordance with the requirements of this section." Specifically, 10 CFR 50.36(d)(3) states, "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

Regulatory Guide (RG) 1.105, "Setpoints for Safety-Related Instrumentation," describes a method acceptable to the NRC staff for complying with the NRC's regulations for ensuring that setpoints for safety-related instrumentation are initially within and remain within the TS limits.

3.0 TECHNICAL EVALUATION

The LPCI system is designed, as part of the emergency core cooling system (ECCS), to restore and maintain the coolant inventory in the reactor vessel so that the core is adequately cooled after a LOCA. The LPCI system operates in conjunction with the high pressure coolant injection system, the core spray system, and the automatic depressurization system to achieve this goal.

The LPCI system is designed to operate at low pressure and high flow to cover the reactor core to at least two-thirds core height and to maintain this level. The system operates after the reactor pressure vessel has depressurized. At MNGP, the LPCI system uses the residual heat removal pumps to draw suction from the suppression pool and discharge water to one of the recirculation loops. The purpose of the LPCI loop select logic is to determine the differences in pressure between the two recirculation loops and select the intact recirculation loop for injection.

3.1 Instrumentation Evaluation

The proposed amendment would revise Table 3.3.5.1-1, "Emergency Core Cooling System Instrumentation," as follows: Revise the allowable value for Function 2.j, LPCI Recirculation Riser Differential Pressure – High (Break Detection), from less than or equal to 24.0 inches water column (wc) to less than or equal to 100 inches wc; replace Surveillance Requirement (SR) 3.3.5.1.6 (channel calibration at 12 month frequency) with SR 3.3.5.1.7 (channel calibration at 24 month frequency).

3.1.1 Instrumentation Setpoint Methodology

The licensee proposed to revise ECCS instrumentation to reflect a new allowable value based on analysis of the SBLOCA which determined a new minimum detectable break area for the LPCI loop select logic.

The licensee applied the General Electric-Hitachi (GEH) Instrument Setpoint Methodology (ISM) to determine the allowable value for LPCI loop select logic Function 2.j. The GEH methodology, which was previously approved by the NRC staff, is based on the Instrument Society of America Standard 67.04, which is endorsed with clarifications by RG 1.105. The calculation uses the square-root-sum-of-the-squares method and includes all errors to include process measurement accuracy, primary element accuracy, instrument loop accuracy (A_T), instrument calibration errors (C), and drift (D). The staff found that the licensee's setpoint calculation methodology is consistent with the guidance in RG 1.105.

In accordance with Technical Specification Task Force (TSTF) letter to the NRC, "Industry Plan to Resolve TSTF-493, Clarify Application of Setpoint Methodology for LSSS Functions," dated February 23, 2009 (ADAMS Accession No. ML090540849), the licensee has proposed the addition of TS notes to address the controls to ensure operability. The proposed notes for TS Table 3.3.5.1-1, Function 2.j, "LPCI Recirculation Riser Differential Pressure – High (Break Detection)," are as follows:

> "If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service."
>
> "The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the nominal trip setpoint; otherwise, the channel shall be declared inoperable. The nominal trip setpoint and the methodology used to determine the as-found tolerance and the as-left tolerance are specified in the TRM."

3.1.2 Channel Calibration

For Function 2.j, the licensee proposed to replace an associated surveillance SR 3.3.5.1.6, Channel Calibration 12-month frequency, with SR 3.3.5.1.7, Channel Calibration 24-month frequency. The licensee performed a drift analysis, and determined that, based on the additional margin calculated by the setpoint analysis, the channel calibration frequency could be changed from 12 months to 24 months. The 24-month frequency is based upon:

a. The assumptions of the 24-month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

b. Revised minimum detectable break area for the LPCI loop select logic.

The licensee also provided information regarding data found outside specified limits during surveillance testing. The licensee performs surveillances to verify that specific settings are within an acceptable range. If the instrument is found outside the As-Found Tolerance (AFT) or As-Left Tolerance (ALT), it is considered a condition adverse to quality and recorded into the Corrective Action Program (CAP). The ALT band is acceptance criteria for the As-Left value. If the As-Found value is within the ALT band, the calibration is considered acceptable. Any values found outside the ALT band, but within the AFT, must be reset to a value within the ALT band for the surveillance to be considered completed satisfactorily. If the As-Found value is within the allowable value, but outside the AFT band, the instrument is reset to the Nominal Trip Setpoint within the ALT. If the channel is operating correctly, it is returned to service, logged into the licensee's CAP, and the surveillance is considered complete. If the channel is not operating correctly, it is declared inoperable. If at anytime it cannot be determined that an instrument is functioning as required, the instrument would be declared inoperable and the associated TS action requirements followed.

3.1.3 Summary of Instrumentation Evaluation

Based on the above evaluation and the evaluation below regarding SBLOCA break size and probabilistic insight, the NRC staff concludes that the licensee's proposed allowable value change for Function 2.j is consistent with the guidance in RG 1.105, ensures that this function complies with the requirements of 10 CFR 50.36, and is therefore, acceptable. The licensee's procedures will ensure that the instrument will be capable of performing its specified safety function with the surveillance interval changed from 12 months to 24 months; therefore, the NRC staff also finds the proposed channel calibration frequency of 24 months acceptable.

3.2 Change of Minimum Detectable Break Size Evaluation

The licensee proposed to increase the minimum detectable break size of the LPCI loop select logic from the current 0.1 ft^2 to 0.4 ft^2. The licensee submitted an analysis of the SBLOCA for break sizes ranging from 0.05 ft^2 to 0.5 ft^2 to justify the new allowable minimum detectable break size. The SBLOCA analysis also reflected a methodology change, which considered the potential for axial power shape to influence the accident results.

3.2.1 Use of NRC-Approved Methodology and Limitation

The licensee described the methodology used to comply with 10 CFR 50.46 and 10 CFR Part 50, Appendix K. The SBLOCA was analyzed using the SAFER/GESTR-LOCA methodology and associated code set (Topical Report NEDE-23785-1-P-A Rev. 1, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Vol. 3, SAFER/GESTR Application Methodology," October 1984). The NRC staff has previously approved the application of the SAFER/GESTR-LOCA methodology at MNGP in Amendment No. 102, dated September 16, 1998 (Accession No. ML020920138).

The NRC staff approval of the original SAFER/GESTR-LOCA evaluation model imposed a restriction of 1600°F on the acceptable upper bound PCT calculation result for analyses using this methodology. Subsequently, NRC approved removal of this constraint from the general SAFER/GESTR methodology per NRC staff's approval (Topical Report NEDO-23785-A, Supplement 1, Rev. 1, "GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Vol. 3, Supplement 1, Additional Information for Upper Bound PCT Calculation," April 2002). However, the licensee stated that plant-specific application is required for an individual licensee to receive NRC approval for upper bound PCT limit removal. The licensee elected to retain this limitation for this amendment. Therefore, the limitation remains in effect as part of the present analysis basis.

The NRC staff finds that the Monticello SBLOCA analysis is based on an NRC-staff approved methodology, and complies with the associated limitation. Therefore, the staff finds this acceptable.

3.2.2 SBLOCA Analysis

The licensee analyzed a series of SBLOCA cases to investigate the ECCS response under the assumption that the minimum detectable break size of the LPCI loop selection logic system is changed from the current 0.1 ft^2 to 0.4 ft^2. The licensee calculated PCTs for LOCA transients initiated by hypothetical small-breaks ranging from 0.05 ft^2 to 0.5 ft^2, applying assumptions consistent with 10 CFR Part 50, Appendix K. The analysis assumed the more limiting top-peaked axial power shape.

The previous MNGP ECCS-LOCA analysis assumed that the LPCI loop selection logic system was capable of selecting the intact recirculation loop for breaks down to 0.1 ft^2. The proposed analysis assumed failure of the LPCI loop selection logic system for all break sizes lower than the proposed minimum detectable break area of 0.4 ft^2. This assumption was in addition to the limiting single-failure required by regulations.

The most limiting single-failure for small recirculation line breaks is the battery failure, which eliminates one division of low-pressure ECCS capacity along with eliminating high pressurecore injection capacity. The licensing basis PCT was calculated to be 1990°F, demonstrating compliance with 10 CFR 50.46. These results are discussed in the General Electric licensing report (Enclosure 4 of the September 25, 2007 application (ADAMS Accession No. ML072820487)) prepared for MNGP supporting this proposed amendment. The NRC staff finds these results acceptable.

3.2.3 Effect of Top-peaked Axial Power Shape assumption on LBLOCA

To demonstrate compliance with 10 CFR 50.46, the analysis must consider the most severe postulated accidents. The licensee stated that with the present analysis, SBLOCA became the limiting accident when it considered the top-peaked axial power shape. In response to an NRC staff question, the licensee submitted additional analysis results showing that the mid-peaked axial power shape remained limiting for LBLOCA at the maximum extended loadline limit analysis conditions. The NRC staff finds that this substantiates the licensee's claim that the licensing basis PCT is established by the SBLOCA rather than the LBLOCA for this proposed amendment.

3.2.4 Refined Model for Nominal PCT Calculation

In response to an NRC staff question regarding the limiting break location (i.e. recirculation suction line vs. discharge line), the licensee stated that certain model refinements were made for the nominal PCT calculation in order to comply with the upper bound PCT limitation. In its January 20, 2009 letter, the licensee discussed application of modeling credits associated with detailed recirculation system modeling. The licensee submitted analysis results showing the impact of the individual modeling effects. The licensee further stated that the refinements were purely to address the upper bound PCT limitation and that they will be removed upon receiving approval of the application to remove the upper bound PCT limitation. For example, the extended power uprate (a separate licensing action currently under review) analysis basis will not contain the refined model described with this proposed amendment. The licensee confirmed that the refined model was implemented with input changes and no changes were made to the licensing codes themselves. The licensee also confirmed that the refinements were not applied to the Appendix K model, which influences the licensing basis PCT more significantly.

The licensee also clarified that the aforementioned model refinements were a part of the original staff approval of the SAFER/GESTR methodology. Therefore, they do not constitute, in the staff's view, departures from the approved method.

The NRC staff also evaluated how the licensing basis PCT may be affected by the refined model. Due to the manner in which the licensing-basis PCT is calculated, the PCT improvement in the nominal PCT will have a minimal effect on the licensing basis PCT. For example, if the model refinement resulted in a 100°F improvement in the nominal PCT, the improvement in the licensing basis PCT would be significantly below the 50°F significance level for reportability per 10 CFR 50.46. In addition, the purpose of the refined model was to meet the upper bound PCT limitation for which the NRC staff had generically approved its elimination (see NEDO-23785-A, Supplement 1, Rev. 1). Therefore, the NRC staff finds that this limited refinement in the model does not affect the ability of MNGP to comply with the regulations.

3.2.5 Re-zeroing of the Licensing Basis PCT

Subsequent to the initial application, by its November 6, 2008 letter, the licensee elected to withdraw the portion of the application that requested the re-zeroing of the licensing basis PCT. Therefore, this safety evaluation should not be interpreted as approval of the initial request to re-zero the licensing basis PCT.

3.2.6 Summary of Minimum Detectable Break Size Evaluation

In consideration of the information discussed above, the NRC staff finds that the proposed change to minimum detectable break size is acceptable. The licensee has provided adequate justification for the proposed change and the NRC staff concludes that the licensee's analysis of the SBLOCA provides reasonable assurance that proposed change in minimum detectable area from the current 0.1 ft^2 to 0.4 ft^2 will not adversely affect MNGP's ability to comply with the regulatory requirements.

3.3 Evaluation of Risk Insight

The proposed license amendment was not risk-informed, but was based on deterministic analysis. However, the licensee did provide risk insights from a plant-specific MNGP probabilistic risk assessment (PRA) of the proposed change to extend the surveillance test interval (STI).

The scope of the NRC staff's review was limited to the evaluation of the risk impact and potential risk implications of the licensee's amendment request per the guidance of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Chapter 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," Appendix D, "Use of Risk Information in Review on Non-Risk-Informed License Amendment Requests." Appendix D provides review and assessment guidance on whether a "special circumstance" exists such that the normal presumption of adequate protection is no longer met by compliance with existing regulatory requirements for license amendment requests that are not risk-informed.

Per the guidance given in Appendix D, the NRC staff used the risk-informed decision-making process in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," in its review. Although the RG 1.174 acceptance guidelines by themselves do not constitute a definition of adequate protection, they do provide an appropriate set of criteria to be used in the initial process of evaluating adequate protection and provide a basis for finding that there is reasonable assurance of adequate protection.

Function 2.j historically has been calibrated on a once per year frequency. As part of the 2-year fuel cycle extension project, the licensee previously proposed to revise the calibration interval from 12-months to once per cycle (24-months), and performed a drift analysis to determine the acceptability in accordance with the MNGP implementation of the GEH ISM. The licensee determined then that due to instrument drift and small available analytical margin, the calibration interval could not be extended at that time.

The previous ECCS-LOCA analysis assumed the LPCI loop select logic was capable of selecting the intact recirculation loop for break sizes down to 0.1 ft^2. The calculated pressure differential between the recirculation loops corresponding to this 0.1 ft^2 minimum detectable break area provides the current analytical basis for the recirculation riser differential pressure measuring instrumentation within the plant TS.

As discussed above, the licensee performed an analysis of the SBLOCA, which conservatively postulated the failure of the LPCI loop selection logic to select the unbroken (correct) recirculation loop for all break sizes lower than a minimum detectable break area of 0.4 ft^2. Calculation of the pressure differential between the recirculation loops corresponding to this increased minimum detectable break area of 0.4 ft^2 provides a new analytical basis for the recirculation riser differential pressure measuring instrumentation within the plant TS.

The licensee modified the MNGP 2005 average maintenance probabilistic risk assessment (PRA) model to reflect the increased failure probability due to extending the STI from 12 to 24 months. In addition, the licensee assumed that the loop select logic always selects the wrong recirculation loop for injection, which is conservative. The LPCI loop select logic is only required for a large recirculation line break LOCA. The importance of the loop select feature is limited by the LPCI recirculation loop LOCA break sizes. For the higher probability break size of less than 3 inches, the licensee stated that adequate core cooling is maintained with the assumed failure of the LPCI loop select logic to select the unbroken recirculation loop. For breaks larger than 3 inches, the assumed failure of the LPCI loop select logic to select the unbroken recirculation loop will result in LPCI failure for recirculation LOCA events. Therefore, the impact is limited to lower probability medium- and large-break LOCAs on the recirculation line.

Based on the above, the licensee estimated a change in core damage frequency (ΔCDF) of less than 1.0E-9/year for the proposed extension of the STI from 12 months to 24 months. This is three orders of magnitude below the threshold of 1.0E-6/year ΔCDF for very small changes per RG 1.174. For this evaluation, if all core damage events were conservatively assumed to lead to a large early release, the resulting change in large early release frequency (ΔLERF) of 1.0E-9/year would be two orders of magnitude below the threshold of 1.0E-7/year ΔLERF for very small changes per RG 1.174. These changes in risk are consistent with the NRC staff's understanding of the safety significance of the LPCI loop select logic, based on previous NRC staff reviews of related license amendment requests for MNGP.

Therefore, the NRC staff concludes that the licensee's proposed extension of the STI for Function 2.j does not invoke "special circumstances," as defined in Standard Review Plan (SRP) 19.2, Appendix D.

The NRC staff finds that the licensee's proposed extension of the STI for Functional 2.j from 12 to 24 months does not reveal an unforeseen hazard or a substantially greater potential for a known hazardous event to occur such that adequate protection would be in question. The NRC staff did not identify "special circumstances" that, if reviewed on a risk-informed basis, would warrant attaching conditions to or denying the proposed changes. This conclusion is based on the very small increase in CDF and LERF (i.e., the increase in risk is within the RG 1.174

acceptance guidelines) for the proposed change. The estimated risk impacts are very small and should not significantly influence the overall results of the licensee's deterministic analysis.

The licensee did not indicate that the risk impacts played any role in its basis for the acceptability of this proposed amendment. The license's amendment application did not address the key principles of risk-informed decision making as presented in RG 1.174, and the NRC staff did not complete the full scope of risk review that would be required of a risk-informed submittal. Although RG 1.174 risk acceptance guidelines were used in evaluating the licensee's one-time amendment request, the NRC staff's review, by itself, does not provide a basis for approving the amendment based on the limited risk information available compared with a risk-informed submittal. In addition, the NRC staff did not evaluate traditional engineering insights such as maintenance recommendations, surveillance or maintenance history, setpoint methodology, or related topical reports (not specific to PRA analysis).

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Minnesota State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no-significant-hazards considerations, and there has been no public comment on the finding (72 FR 65368). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The NRC staff has concluded, on the basis of the considerations discussed above, that (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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: April 7, 2009

April 7, 2009

Mr. Timothy J. O'Connor
Site Vice President
Monticello Nuclear Generating Plant
Northern States Power Company - Minnesota
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - ISSUANCE OF AMENDMENT REGARDING RECIRCULATION RISER DIFFERENTIAL PRESSURE (TAC NO. MD6864)

Dear Mr. O'Connor:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 161 to Renewed Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The amendment consists of changes to the Technical Specifications in response to your application dated September 25, 2007, as supplemented by letters dated September 8, 2008, November 6, 2008, January 20, 2009 and April 2, 2009.

The amendment revised the allowable value and channel calibration frequency for Function 2.j, Recirculation Riser Differential Pressure - High Function (Break Detection), in Table 3.3.5.1-1, "Emergency Core Cooling System Instrumentation."

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/ Karl Feintuch for
Peter S. Tam, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosures:
1. Amendment No. 161 to DPR-22
2. Safety Evaluation

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DATE	4/7/09	4/7/09	4/3/09	2/26/09*	3/8/08*	4/3/09	4/6/09	4/7/09

*Safety evaluation transmitted by memo on the indicated date.

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