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

February 27, 2013

Mr. Mark A. Schimmel
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 TO REVISE AND RELOCATE PRESSURE TEMPERATURE CURVES TO A PRESSURE TEMPERATURE LIMITS REPORT (TAC NO. ME7930)

Dear Mr. Schimmel:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 172 to Renewed Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant (MNGP). The amendment consists of changes to the technical specifications (TSs) in response to your application dated January 20, 2012, as supplemented on December 7, 2012.

The amendment approves a revision to the MNGP TS Section 1.0, "Definitions," Section 3.4.9, "RCS Pressure and Temperature (P-T) Limits," and Section 5.6, "Administrative Controls." The amendment revises the P-T limits based on a methodology documented in the SIR-05-044-A report, "Pressure-Temperature Limits Report [PTLR] Methodology for Boiling Water Reactors," and relocates the revised P-T limits from the TS to the MNGP PTLR.

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,

Terry A. Beltz, 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. 172 to DPR-22
2. Safety Evaluation

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 172
License No. DPR-22

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

   A. The application for amendment by Northern States Power Company (NSPM, the licensee), dated January 20, 2012, as supplemented on December 7, 2012, 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:

Enclosure 1

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 172, 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 180 days after start-up from the 2013 Refueling Outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert D. Carlson, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed
Operating License DPR-22
and Technical Specifications

Date of Issuance: February 27, 2013

ATTACHMENT TO LICENSE AMENDMENT NO. 172

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 attached revised pages. The revised pages are identified by amendment number and contain a marginal line indicating the area of change.

| REMOVE | INSERT |
|---|---|
| 1.1-4 | 1.1-4 |
| 3.4.9-1 | 3.4.9-1 |
| 3.4.9-2 | 3.4.9-2 |
| 3.4.9-3 | 3.4.9-3 |
| 3.4.9-4 | --- |
| 3.4.9-5 | --- |
| 3.4.9-6 | --- |
| 3.4.9-7 | --- |
| 5.6-3 | 5.6-3 |

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. 172, 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

## 1.1 Definitions

| | |
|---|---|
| OPERABLE – OPERABILITY | A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s). |
| PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) | The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.5. |
| RATED THERMAL POWER (RTP) | RTP shall be a total reactor core heat transfer rate to the reactor coolant of 1775 MWt. |
| REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME | The RPS RESPONSE TIME shall be that time interval from initiation of any RPS channel trip to the de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. |
| SHUTDOWN MARGIN (SDM) | SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that:<br>a. The reactor is xenon free;<br>b. The moderator temperature is 68°F; and<br>c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM. |

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.9 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.9 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation pump starting temperature requirements shall be maintained within the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. ------------NOTE------------<br>Required Action A.2 shall be completed if this Condition is entered.<br>--------------------------------<br>Requirements of the LCO not met in MODE 1, 2, or 3. | A.1 Restore parameter(s) to within limits.<br><br>AND<br><br>A.2 Determine RCS is acceptable for continued operation. | 30 minutes<br><br><br><br>72 hours |
| B. Required Action and associated Completion Time of Condition A not met. | B.1 Be in MODE 3.<br><br>AND<br><br>B.2 Be in MODE 4. | 12 hours<br><br><br><br>36 hours |
| C. ------------NOTE------------<br>Required Action C.2 shall be completed if this Condition is entered.<br>--------------------------------<br>Requirements of the LCO not met in other than MODES 1, 2, and 3. | C.1 Initiate action to restore parameter(s) to within limits.<br><br>AND<br><br>C.2 Determine RCS is acceptable for operation. | Immediately<br><br><br><br>Prior to entering MODE 2 or 3 |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.9.1 | -----NOTES----- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ----- <br><br> Verify: <br><br> a. RCS pressure and RCS temperature are within the applicable limits specified in the PTLR; and <br><br> b. RCS heatup and cooldown rates are within the the limits specified in the PTLR. | 30 minutes |
| SR 3.4.9.2 | Verify RCS pressure and RCS temperature are within the criticality limits specified in the PTLR. | Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality |
| SR 3.4.9.3 | -----NOTE----- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. ----- <br><br> Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is within the limits specified in the PTLR. | Once within 15 minutes prior to each startup of a recirculation pump |
| SR 3.4.9.4 | -----NOTE----- Only required to be performed when tensioning the reactor vessel head bolting studs. ----- <br><br> Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR. | 30 minutes |

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.9.5 | ------NOTE------<br>Not required to be performed until 30 minutes after RCS temperature ≤ 80°F in MODE 4.<br>------------<br><br>Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR. | 30 minutes |
| SR 3.4.9.6 | ------NOTE------<br>Not required to be performed until 12 hours after RCS temperature ≤ 100°F in MODE 4.<br>------------<br><br>Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR. | 12 hours |

## 5.6 Reporting Requirements

5.6.4 Post Accident Monitoring Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.5 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

   1. Limiting Conditions for Operation Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits"

   2. Surveillance Requirements Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits"

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

   1. SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," Revision 0, dated April 2007.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplements thereto.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 172 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

## 1.0 INTRODUCTION

By letter to the U.S. Nuclear Regulatory Commission (NRC, the Commission) dated January 20, 2012, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12033A175), as supplemented by letter dated December 7, 2012 (ADAMS Accession No. ML12349A210), Northern States Power Company – Minnesota (NSPM, the licensee) submitted a license amendment request (LAR) for the Monticello Nuclear Generating Plant (MNGP). The proposed amendment would revise the MNGP Technical Specifications (TS) Section 1.0, "Definitions," Section 3.4.9, "RCS Pressure and Temperature (P-T) Limits," and Section 5.6, "Administrative Controls." The licensee would revise the pressure-temperature (P-T) limits based on a methodology documented in report SIR-05-044-A, "Pressure-Temperature Limits Report [PTLR] Methodology for Boiling Water Reactors." In addition, the licensee proposed to relocate the revised P-T limits from the TS to the MNGP PTLR.

The supplement dated December 7, 2012, 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 April 17, 2012 (77 FR 22815).

The NRC staff reviewed the submittals to ensure that the information provided in the proposed PTLR and the revised TS pages is consistent with the SIR-05-044-A report and in accordance with the guidance in NRC Generic Letter (GL) 96-03, "Relocation of Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," and Technical Specifications Task Force (TSTF) Traveler TSTF-419, "Revise PTLR Definition and References in ISTS [Improved Standard Technical Specification] 5.6.6, RCS [Reactor Coolant System] PTLR." The staff also verified that the proposed P-T limits have been developed appropriately using the SIR-05-044-A methodology.

## 2.0 REGULATORY EVALUATION

The NRC has established requirements in Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 50) to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The NRC staff evaluated the acceptability of a facility's proposed PTLR based on the following NRC regulations and guidance:

- Appendix G to 10 CFR Part 50

  Appendix G to 10 CFR Part 50 requires that facility P-T limits for the reactor pressure vessel (RPV) be at least as conservative as those obtained by applying the linear elastic fracture mechanics methodology of Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Code.

- Appendix H to 10 CFR Part 50

  Appendix H to 10 CFR Part 50 establishes requirements related to facility RPV material surveillance programs.

- Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials"

  RG 1.99, Revision 2, contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy resulting from neutron radiation.

- GL 92-01, Revision 1, "Radiation Damage to Reactor Vessel Materials"

  GL 92-01, Revision 1, requested that licensees submit the RPV data for their plants to the NRC for review.

- GL 92-01, Revision 1, Supplement 1

  GL 92-01, Revision 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations.

- Standard Review Plan (SRP), Section 5.3.2

  SRP Section 5.3.2 provides an acceptable method for determining the P-T limits for ferritic materials in the beltline of the RPV based on the ASME Code, Section XI, Appendix G, methodology.

  Appendix G to Section XI of the ASME Code is endorsed in 10 CFR 50.55a, and therefore by reference in 10 CFR Part 50, Appendix G. Additionally, Appendix G to 10 CFR Part 50 imposes minimum head flange temperatures when system pressure is at or above 20 percent (%) of the preservice hydrostatic test pressure.

- GL 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits"

  GL 96-03 addresses the technical information necessary for a licensee's implementation of a PTLR. GL 96-03 establishes the information which must be included in: (1) an acceptable PTLR methodology (with the P-T limit methodology as its subset) and, (2) the PTLR itself.

TSTF-419 provides additional guidance which provides an alternative format for documenting the implementation of a PTLR in the "Administrative Controls" section of a facility's TS.

The licensee requested the initial implementation of a PTLR for MNGP; therefore, the NRC staff's review focused on both the implementation of the MNGP PTLR and the appropriate application of the SIR-05-044-A methodology to generate the proposed MNGP P-T limits. The related neutron fluence calculation was reviewed by the NRC's Division of Safety Systems.

## 3.0 TECHNICAL EVALUATION

### 3.1 Licensee's Evaluation

#### 3.1.1 PTLR Implementation

Attachment 1 of GL 96-03 contains seven technical criteria that the contents of a proposed methodology should conform to if license amendments requesting PTLRs are to be approved by the NRC staff. The NRC staff's evaluations of the contents of the Boiling Water Reactor Owners' Group methodology against the seven criteria in Attachment 1 of GL 96-03 are provided in Section 3.1 of the safety evaluation report for the SIR-05-044-A report. The licensee further stated that the proposed TS changes are consistent with the guidance provided in GL 96-03.

#### 3.1.2 P-T limits

MNGP is a member of the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP). The BWRVIP ISP Data Source Book (BWRVIP-135) is periodically updated to include the results of additional surveillance capsule data. The BWRVIP-135 report updated the chemistry factor associated with plate heat number C2220 from 130.8 degrees Fahrenheit (°F) to 180 °F. In addition, based on the analysis of surveillance data, the margin for plate heat number C2220 lowered from 34 °F to 17 °F.

The adjusted reference temperature (ART) values and P-T limits valid for 36, 40, and 54 Effective Full Power Years (EFPY) of facility operation using the SIR-05-044-A methodology were documented in the proposed MNGP PTLR. The minimum required leak test temperature (Curve A, hydrostatic pressure tests and leak tests) is above 200 °F at 54 EFPY. The licensee included additional Curve A limits at 36 and 40 EFPY, since the 54 EFPY Curve A limits present operational challenges. The licensee identified the limiting beltline materials as the lower intermediate shell plates (Heat Nos. C2220-1 and C2220-2) and the recirculation inlet nozzle (N2) at low temperature/low pressure conditions. The key parameters in determining the licensee's ART value for the limiting material at the one-quarter of the RPV wall thickness (1/4T) location are shown in Tables 6, 7, and 8, of the PTLR (Enclosure 4 of the January 20, 2012,

submittal) for 36, 40, and 54 EFPY, respectively. Since the upper-intermediate shell plates were projected to accumulate neutron fluence values in excess of 1 x $10^{17}$ n/$cm^2$ (neutrons per centimeters squared) (E > 1 MeV) [energy greater than 1 megaelectron volt] and therefore considered to be RPV beltline materials, the parameters for calculation of the ART for the upper-intermediate shell plates were also included in Tables 6, 7, and 8 of the PTLR.

3.2 Staff Evaluation

3.2.1 PTLR Implementation

As mentioned in Section 3.1.1 of this safety evaluation (SE), Attachment 1 of GL 96-03 requires the licensee to evaluate and document seven technical criteria to demonstrate the acceptability of its PTLR methodology. The NRC staff examined the proposed PTLR and determined that it was developed from the template PTLR of the SIR-05-044-A report and meets the seven technical criteria:

(1) The PTLR methodology describes the transport calculation methods including computer codes and formula used to calculate neutron fluences (Section 3.0, "Methodology," Page 4 of the MNGP PTLR).

(2) The PTLR methodology describes the surveillance program (Appendix A, "Monticello Reactor Vessel Materials," Page 38 of the MNGP PTLR).

(3) The PTLR methodology describes how the low temperature overpressure protection system limits are calculated applying system/thermal hydraulics and fracture mechanics (not applicable to boiling-water reactors).

(4) The PTLR methodology describes the method for calculating the ART values using RG 1.99, Revision 2 (Section 3.0, "Methodology," Page 4 of the MNGP PTLR).

(5) The PTLR methodology describes the application of fracture mechanics in the construction of P-T limits based on ASME Code, Section XI, Appendix G, and the guidance in the NRC's SRP. The MNGP PTLR provided information regarding the finite element analyses performed to generate part of the P-T limits. The submittal stated that the equations and values were calculated in accordance with the SIR-05-044-A report.

(6) The PTLR methodology describes how the minimum temperature requirements in Appendix G to 10 CFR Part 50 are applied to P-T limits for bolt-up temperature and hydrotest temperature (Page 3 of the MNGP PTLR stated that the P-T limits were calculated in accordance with the SIR-05-044-A report. This description is sufficient because the SIR-05-044-A report contained detailed information regarding the minimum temperature requirements for bolt-up temperature and hydrotest temperature.)

(7) The PTLR methodology describes how the data from multiple surveillance capsules are used in the ART calculation (Appendix A of the MNGP PTLR.)

Hence, the NRC staff finds that implementation of the MNGP PTLR is acceptable.

3.2.2 P-T Limits

The licensee states on Page 9 of Enclosure 1 to the January 20, 2012, submittal letter, the following:

> The General Electric – Hitachi (GEH) fluence methodology, NEDC-32983P-A, "Licensing Topical Report, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," was used to perform the fluence calculations. The GEH methodology is NRC approved and Regulatory Guide 1.190 [Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence] compliant.
>
> These calculations determined the projected fluence accumulation for the reactor vessel, considering EPU power level (2004 MWth [megawatts thermal]) and EPU [extended power uprate] operation in the MELLLA+ [Maximum Extended Load Line Limit Analysis Plus] operating domain. The calculations were performed for the assumed fluence at the end of the renewed facility operating license period (i.e., 54 EFPY) on September 8, 2030. A multiplication factor of 1.3 was applied by GEH to the fluence values to account for potential variation in future cycles of operation. The neutron source distribution used in the calculations is based on the EPU equilibrium core, not on an extrapolation of the original licensed thermal power source distribution.

The neutron fluence values were calculated in accordance with the NRC-approved method described in licensing topical report GEH-NEDO-32983-A, Revision 2 (the NEDO designator refers to the open distribution version of the NEDC report (ADAMS Accession No. ML072480121)). The NRC staff's SE in approving NEDO-32983-A concludes that the plant-specific neutron fluence values calculated following this methodology would be adherent to the RG 1.190 guidance and hence acceptable. RG 1.190 provides guidance concerning the calculation of acceptable reactor pressure vessel neutron fluence values. Since the fluence calculations were performed in accordance with an NRC-approved methodology and using the guidance in RG 1.190, the NRC staff finds the fluence calculations acceptable insofar as they support the requested PTLR implementation.

The NRC staff notes that operation with an EPU and MELLLA+ is anticipated at MNGP. These operating domain changes cause the steady-state void fractions in the core to increase, reducing the neutron moderation and resulting in the neutron flux energy spectrum at the vessel wall to increase. The result is a slight increase to the reactor vessel neutron fluence above 1 MeV. Since the licensee's calculations include both this change and the multiplication factor of 1.3, the fluence values are conservative relative to operation in the currently licensed operating domain. This conservatism is acceptable to the NRC staff.

Operation in the MELLLA+ operating domain, in particular, is addressed by a separate NRC review activity. The NRC SE approving the MELLLA+ licensing topical report (NEDO-33006-A, Revision 3, "Maximum Load Line Limit Analysis Plus" (ADAMS Accession No. ML091800512)) states that demonstration of the performance of the reactor vessel materials will be dependent on plant-specific evaluations under MELLLA+ conditions using plant-specific design and as-built information.

Based on this statement, the NRC staff concludes that the fluence calculations are acceptable insofar as they conservatively reflect currently licensed plant operation. The acceptability of the fluence values for EPU and MELLLA+ is not within the scope of the current review activity and will be revisited as required during the MELLLA+ review.

To evaluate the proposed P-T limits for the MNGP RPV, the NRC staff confirmed the licensee's ART calculations. The licensee identified the limiting beltline materials as the lower intermediate shell plates (Heat No. C2220-1 and C2220-2) and recirculation inlet nozzle (N2) at low temperature and low pressure conditions. The staff performed independent calculations of the ART values for all RPV beltline materials using the RG 1.99, Revision 2, methodology. The chemistry factor for lower intermediate shell plates (Heat No. C2220-1 and C2220-2) was revised based on updated information in BWRVIP-135, Revision 2. As documented in NUREG-1944, the staff determined that the licensee's participation in the BWRVIP ISP satisfies the requirements of 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements." The chemistry factors for the remaining MNGP beltline materials were determined by the tables in RG 1.99, Revision 2. The staff's ART values for the limiting beltline material at the 1/4T location are consistent with the values of 147.4 °F, 156.0 °F, and 186.6 °F, reported for the lower intermediate shell plates (Heat No. C2220-1 and C2220-2) for 36, 40, and 54 EFPY, respectively. The licensee did not calculate the ART value at the 3/4T location, which is relevant to the heatup P-T limit calculation, because the SIR-05-044-A report determined that P-T limits for the cooldown transient are bounding.

The P-T limits calculations for ferritic reactor coolant pressure boundary components that are not RPV beltline shell materials (have projected neutron fluence values less than $1 \times 10^{17}$ n/cm$^2$ E > 1 MeV) may define P-T curves that are more limiting than those calculated for the RPV beltline shell materials because RPV nozzles, penetrations and other discontinuities have complex geometries that may exhibit higher stresses than those for the RPV beltline shell region. These higher stresses can potentially result in more restrictive P-T limits, even if the reference temperature ($RT_{NDT}$) for these components is not as high as that of the RPV beltline shell materials that have simpler geometries. .

The proposed MNGP P-T limits are composite curves, representing the most limiting P-T limits for the RPV beltline, the bottom head, feedwater and recirculation inlet nozzles, and the upper vessel. The NRC staff has verified that, for a cooldown of 100 °F per hour (°F/hr), the bottom head P-T limits are the least limiting. For the RPV beltline P-T limit segment, incorporating plant-specific information, the staff utilized the ASME Code, Section XI, Appendix G, methodology in its independent evaluation, using the $K_{IC}$ curve as resistance and the pressure-dependent $K_{Im}$ formula and the cooldown rate dependent $K_{It}$ formula as driving forces. The NRC staff produced almost identical beltline P-T limits for the two ends of the upper segment. Enclosures 6 and 7 of the January 20, 2012, submittal contained a three-dimensional finite element stress analysis of the RPV feedwater nozzle and recirculation inlet nozzle, respectively. The feedwater nozzle analysis was performed to determine through-wall and pressure stress distributions due to a bounding thermal transient. A thermal shock representing the maximum thermal shock for the feedwater nozzle during normal operating conditions, and a thermal ramp were analyzed. The thermal ramp of 100 °F/hour was determined to produce higher tensile stresses at the 1/4T location than the thermal shock. In developing a three-dimensional finite element analysis of the feedwater nozzle, conservative heat transfer coefficients were used to bound the actual operating conditions and material properties from the MNGP Code of Record

were primary inputs. In addition, a plant-specific analysis of the recirculation inlet nozzle was performed. Stresses representing the bounding stresses associated with the 100 °F/hour heatup and cooldown limits for beltline nozzles were used as inputs to the analysis. Heat transfer coefficients were based on the full temperature difference of the transient, rather than the RPV to coolant temperature difference. The NRC staff concluded that this is a conservative basis for the heat transfer coefficients.

The feedwater nozzle was confirmed to be the bounding component in the upper vessel due to the fact that it is a stress concentrator and typically experiences more severe thermal transients in comparison to the rest of the upper vessel region.

The data for beltline region, upper vessel and bottom head, were tabulated separately in the PTLR and depicted graphically for the P-T curves at 36, 40, and 54 EFPY in Figures 1, 2, and 3, respectively.

10 CFR Part 50, Appendix G, contains additional requirements for the minimum metal temperature of the closure head flange and vessel flange regions. These considerations were reflected in the "notch" of the upper vessel P-T limits. According to Appendix G to 10 CFR Part 50, the minimum bolt-up temperature is equal to the limiting material $RT_{NDT}$ of the regions affected by the bolt-up stresses. According to Section 2.8 ("Minimum Bolt-up Temperature") of SIR-05-044-A, the minimum bolt-up temperature shall not be lower than 60 °F therefore the minimum bolt-up temperature shall be 60 °F or the material $RT_{NDT}$, whichever is higher. NRC staff concluded that since the material $RT_{NDT}$ values are less than 60 °F, the initial assumed fluid temperature in the P-T curve calculation of 60°F is consistent with SIR-05-044-A and satisfy the requirements of Section XI of the ASME Code and Appendix G to 10 CFR Part 50. Hence, the licensee's proposed P-T limit curves are acceptable for operation of the MNGP RPV and are valid for 54 EFPY.

Finally, the NRC staff notes that this LAR addresses two non-conservative TS conditions: (1) TS Figure 3.4.9-1, "Core Beltline Operating Limits Versus Fluence," becomes non-conservative in approximately 2026, and (2) additional neutron fluence accumulation results in the consideration of the recirculation inlet nozzles as a beltline material and is a limiting component under low temperature and low pressure conditions. The guidance of NRC Administrative Letter 98-10, "Dispositioning of Technical Specification that are Insufficient to Assure Plant Safety," is being applied until these conditions are resolved with approval of this LAR.

## 3.3 Summary

Based on a review of the information provided in the licensee's January 20, 2012, application submittal and December 7, 2012, supplement, the NRC staff concludes the following for those topics assessed in this SE:

The proposed MNGP PTLR meets the implementation guidance as specified in GL 96-03 and, therefore, is approved as part of the MNGP licensing basis.

The MNGP RPV P-T limits are based on an acceptable methodology documented in the SIR-05-044-A report. The NRC staff performed independent evaluations and verified that the

P-T limits were developed appropriately using the SIR-05-044-A methodology, and that the proposed P-T limits for 54 EFPY satisfy the requirements of Appendix G to Section XI of the ASME Code and Appendix G to 10 CFR Part 50.

## 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 use of facility components located within the restricted area as defined in 10 CFR Part 20. 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 consideration and there has been no public comment on such finding (77 FR 22815, dated April 17, 2012). 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 Commission has concluded, based on 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) there is reasonable assurance that 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.

Principal Contributor: Carolyn Fairbanks, NRR
Benjamin Parks, NRR

Date of issuance: February 27, 2013

February 27, 2013

Mr. Mark A. Schimmel
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 TO REVISE AND RELOCATE PRESSURE TEMPERATURE CURVES TO A PRESSURE TEMPERATURE LIMITS REPORT (TAC NO. ME7930)

Dear Mr. Schimmel:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 172 to Renewed Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant (MNGP). The amendment consists of changes to the technical specifications (TSs) in response to your application dated January 20, 2012, as supplemented on December 7, 2012.

The amendment approves a revision to the MNGP TS Section 1.0, "Definitions," Section 3.4.9, "RCS Pressure and Temperature (P-T) Limits," and Section 5.6, "Administrative Controls." The amendment revises the P-T limits based on a methodology documented in the SIR-05-044-A report, "Pressure-Temperature Limits Report [PTLR] Methodology for Boiling Water Reactors," and relocates the revised P-T limits from the TS to the MNGP PTLR.

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/*
Terry A. Beltz, 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. 172 to DPR-22
2. Safety Evaluation

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**ADAMS Accession No.: ML13025A155** * Safety evaluation transmitted by memo dated 01/22/13

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