September 15, 1994

Mr. Roger O. Anderson, Director
Licensing and Management Issues
Northern States Power Company
414 Nicollet Mall
Minneapolis, MN 55401

Dear Mr. Anderson:

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - ISSUANCE OF AMENDMENT RE: CHANGE IN SAFETY RELIEF VALVES TESTING REQUIREMENTS (TAC NO. M88603)

The Commission has issued the enclosed Amendment No. 92 to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated

The amendment revises the requirements of TS 4.6.E.1.a, which previously specified that a minimum of seven safety/relief valves shall be bench checked or replaced with a bench checked valve each refueling outage. The proposed amendment changes this specification to require the valves to be tested in accordance with the Section XI Inservice Testing Requirements of the ASME Boiler and Pressure Vessel Code. The proposed change is consistent with the Improved Standard Technical Specifications, NUREG-1433.

A copy of our related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by

Beth A. Wetzel, Acting Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosures: 1. Amendment No. 92 to DPR-22
2. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION:
See next page

(with 2 changes)

OFFICE	(A)LA:PD31-1	(A)PM:PD31-1	BC:EMEB	OGC	PD:PD31-1
NAME	LLessler	BWetzel:gll	RWessman		LMarsh
DATE	09/7/94	09/7/94	09/7/94	09/12/94	09/14/94

OFFICIAL RECORD COPY FILENAME: G:\WPDOCS\MONTICEL\MON88603.AMD

DF01

cc:

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Site General Manager
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DATED: September 15, 1994

AMENDMENT NO. 92 TO FACILITY OPERATING LICENSE NO. DPR-22-MONTICELLO

Docket File
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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 FACILITY OPERATING LICENSE

Amendment No. 92
License No. DPR-22

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

   A. The application for amendment by Northern States Power Company (the licensee) dated January 3, 1994, 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 Facility Operating License No. DPR-22 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 92 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

L B Marsh

Ledyard B. Marsh, Director
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 15, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 92

FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

REMOVE	INSERT
127	127

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3.0 LIMITING CONDITIONS FOR OPERATION

E. Safety/Relief Valves

1. During power operating conditions and whenever reactor coolant pressure is greater than 110 psig and temperature is greater than 345°F:

   a. The safety valve function (self-actuation) of seven safety/relief valves shall be operable.

   b. The solenoid activated relief function (Automatic Pressure Relief) shall be operable as required by Specification 3.5.E.

   c. The Low-Low Set Function for three non-Automatic Pressure Relief Valves shall be operable as required by Specification 3.2.H.

2. If Specification 3.6.E.1.a is not met, initiate an orderly shutdown and have reactor coolant pressure and temperature reduced to 110 psig or less and 345°F or less within 24 hours.

4.0 SURVEILLANCE REQUIREMENTS

E. Safety/Relief Valves

1. a. Safety/relief valves shall be tested or replaced each refueling outage pursuant to Specification 4.15.B. The nominal self-actuation setpoints are specified in Section 2.4.B.

   b. At least two of the safety/relief valves shall be disassembled and inspected each refueling outage.

   c. The integrity of the safety/relief valve bellows shall be continuously monitored.

   d. The operability of the bellows monitoring system shall be demonstrated at least once every three months.

2. Low-Low Set Logic surveillance shall be performed in accordance with Table 4.2.1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 92 TO FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

By letters dated January 3, 1994 and August 29, 1994, the Northern States Power Company (the licensee) requested an amendment to the Technical Specifications (TS) appended to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The August 29, 1994, letter only provided additional test data and did not affect the staff's initial no significant hazards determination contained in the original Federal Register notice. The proposed amendment would change the surveillance requirement for safety relief valves (SRV) to be consistent with the Section XI Inservice Testing Requirements of the ASME Code. ASME Section XI, Subarticle IWV-3200, states that pressure relief devices shall be tested in accordance with the requirements of ANSI/ASME OM-1-1981. Paragraph 1.3.3.1.2 of ANSI/ASME OM-1-1981 addresses the applicable test frequency of Class 1 pressure relief devices and requires that, "all valves of each type and manufacture shall be tested within each subsequent 5 year period with a minimum of 20% of the valves tested within any 24 months. This 20% shall be previously untested valves, if they exist." Currently, Monticello TS require that a minimum of seven SRVs be bench tested or replaced with a bench tested valve every refueling outage.

2.0 EVALUATION

The eight main steam SRVs at Monticello are 3-stage Target Rock design which consist of a 2-stage pilot valve section, remote air actuator section and the main valve section. The SRVs are designed to ensure that the reactor coolant system pressure safety limit is never reached. The SRVs are set to open at a pressure no higher than 105% of the vessel design pressure and they limit the reactor pressure to no more than 110% of the reactor pressure vessel design pressure. Three of the SRVs are assigned to the automatic depressurization system (ADS). The function of the ADS is to provide a backup to the high pressure coolant injection (HPCI) system for automatically depressurizing the reactor vessel so that the low pressure coolant injection (LPCI) and the core spray system can operate to protect fuel cladding.

The SRVs are sized for overpressure protection during a main steam isolation valve (MSIV) closure at full reactor power. As the system isolates, pressure rises in the vessel until the SRVs open to mitigate the accident. The evaluation assumes that only seven of the eight SRVs are operable and that they open at 1% over their setpoint with a 0.4 second time delay.

As discussed in Updated Safety Analysis Report Section 4.4.4, it is not feasible to test the SRV setpoints while the valves are installed in the plant or during normal plant operation. The valves are removed for maintenance or bench checks and reinstalled during normal plant shutdowns. In the past, normal practice has been for all eight pilot assemblies to be bench checked or replaced with spare bench checked pilot assemblies during each refueling outage, even though TS 4.6.E.1.a only requires this to be done for seven of the eight valves. This was done in order to avoid an administrative oversight in missing a valve required to be bench checked during an outage. In addition, the main seats of two SRVs are inspected during each refueling outage as required by TS 4.6.E.1.b.

The proposed change reduces the test frequency from seven of the valves being bench tested every refueling outage to 20% of the valves being tested within any 24 months. The decreased test frequency is permitted by ANSI/ASME OM-1-1981, which has been reviewed and approved by the ASME Code Committee for industry-wide use and has been incorporated by reference through the 1986 Edition of Section XI of the ASME Code in 10 CFR 50.55a, May 5, 1988 (Federal Register, Vol. 53, page 16051). The current Inservice Test Program for Monticello was developed to the 1986 Edition of Section XI. The proposed change is also consistent with Surveillance Requirement 3.4.3.1 of NUREG-1433, "Standard Technical Specifications, General Electric Plant, BWR/4," which specifies the frequency of SRV testing to be in accordance with inservice testing requirements.

The licensee states in its evaluation that the change will have no adverse impact on the reliability of the SRVs. The licensee also states that, although the change may result in any single valve being tested or changed out less frequently, the overall reliability of the valves may increase because the most common performance problems associated with the valves can be induced by handling or disassembly work. The proposed change will minimize such work on the valves. The licensee also stated in its January 3, 1994, submittal that as before, all eight SRVs will continue to be exercised each refueling outage in accordance with ASME Section XI requirements during plant startup.

The licensee has not experienced problems with setpoint drift that plants with the different design, 2-stage Target Rock SRVs, have experienced as verified by the data submitted by the licensee in its letter dated August 29, 1994. The letter contains the as-found SRV lift pressures for SRV testing performed during the past 5 years. The results of the data show that only three valves in the past 5 years lifted at pressures outside of the as-found acceptance criteria (1076-1131 psig). However, the valves are only designed to maintain their setpoint tolerance over the operating cycle to +/- 3% (1075-1142 psig). Therefore, the as-found acceptance criteria is more stringent than what the valves are designed to provide. All the lift pressures for SRV tests performed during the past 5 years were within the design tolerances of the valves.

The staff has determined that the proposed change in SRV testing frequency is acceptable. This conclusion is based on: (a) compliance with the 1986 Edition of Section XI of the ASME Code and ANSI/ASME OM-1-1981, (b) compliance with NUREG-1433, "Standard Technical Specifications, General Electric Plant, BWR/4," surveillance requirements, and (c) reliable as-found setpoint data for the past 5 years of bench testing.

3.0 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.

4.0 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. The 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 (59 FR 10011). 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.

5.0 CONCLUSION

The staff 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) 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: B. Wetzel

Date: September 15, 1994