UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

Docket No. 50-263

REQUEST FOR AMENDMENT TO
OPERATING LICENSE NO. DPR-22

License Amendment Request Dated May 29, 1984

Northern States Power Company, a Minnesota corporation, requests authorization for changes to the Technical Specifications as shown on the attachments labeled Exhibit A and Exhibit B. Exhibit A describes the proposed changes along with reasons for the change. Exhibit B is a set of Technical Specification pages incorporating the proposed changes. Exhibit C is a Safety Evaluation prepared by the General Electric Company in support of the proposed changes.

This letter contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By David Musolf
David Musolf
Manager - Nuclear Support Services

On this 29th day of May, 1984 before me a notary public in and for said County, personally appeared David Musolf, Manager - Nuclear Support Services, and being first duly sworn acknowledged that he is authorized to execute this document on behalf of Northern States Power Company, that he knows the contents thereof and that to the best of his knowledge, information and belief, the statements made in it are true and that it is not interposed for delay.

Betty J. Dean

BETTY J DEAN
NOTARY PUBLIC - MINNESOTA
RAMSEY COUNTY
My Commission Expires Dec 13 1987

EXHIBIT A

Monticello Nuclear Generating Plant

License Amendment Request Dated May 29, 1984

Proposed Changes to the Technical Specifications
Appendix A of Operating License DPR-22

Purusant to 10 CFR Part 50, Section 50.59 and Section 50.90, the holders of Operating License DPR-22 hereby propose the following changes to Appendix A, Technical Specifications:

1. RHR Intertie Line Addition

PROPOSED CHANGES

Add proposed Limiting Conditions for Operation and Surveillance Requirements applicable to the new RHR intertie line as shown on the attached Exhibit B pages 104, 105a and 116a.

Revise the layout of the text on associated pages 103, 105, 106 and 107 as shown in Appendix B to improve the readability of the material. No changes in text are proposed.

REASONS FOR CHANGE

Northern States Power Company is currently replacing the recirculation system piping at the Monticello Nuclear Generating Plant with material resistant to intergranular stress corrosion cracking (IGSCC). A description of this replacement program was submitted for the information of the NRC Staff on July 29, 1983. The work is being performed under the provisions of 10 CFR Part 50 Section 50.59 since an unreviewed safety question is not involved and no Technical Specification changes are directly involved with the piping replacement.

In addition to the piping replacement, Northern States Power Company intends to install a four-inch intertie line between the RHR suction line and the RHR return lines. Installation of this line will reduce the potential for water hammer in the recirculation system and the RHR system when required to cool down with an isolated or idle recirculation loop. The line also eliminates the need to fill the RHR suction and return lines from the condensate service system prior to initiating shutdown cooling (Exhibit C, Figure 1).

REASONS FOR CHANGE (continued)

The line will be constructed of carbon steel. Three motor operated gate valves will be installed for isolation purposes. RHR return line isolation valves will receive a closure signal on Low Pressure Coolant Injection (LPCI) actuation to prevent a reduction in LPCI flow delivered to the reactor under accident conditions. Power for the RHR return line isolation valves (MO-4085A and MO-4085B) will be supplied from the plant bus currently used to power the LPCI injection valves. Power for the RHR shutdown cooling supply isolation valve (MO-4086) will be supplied from an essential motor control center which receives power from No. 15 safeguards bus. Relays formerly used to provide isolation signals to the recirculation system equalizer and equalizer bypass valves will be used to provide isolation signals for the return line isolation valves. The recirculation equalizer, equalizer bypass and discharge bypass valves are being removed in conjunction with the recirculation piping replacement project.

The new intertie line would be normally isolated during plant operation with MO-4085A and MO-4085B closed and MO-4086 open. The line would be used:

1. When the plant is proceeding to cold shutdown, the valves would be opened after the reactor is out of the run mode and prior to depressurizing the reactor.

2. To prevent a recirculation loop water hammer in a hot isolated loop during primary system depressurization. In this case, MO-4085A and MO-4085B would be opened and MO-4086 would be either open or closed, depending on which loop is isolated.

The General Electric Company conducted an analysis to determine the best method of preventing water hammer and providing for warmup of an idle loop. The analysis indicated that a RHR intertie line was the best of several alternatives. A four-inch intertie line was shown to provide sufficient flow to minimize collection of steam bubbles in the high points of the recirculation system piping and RHR suction and discharge pipes as long as the flow is circulated continuously during depressurization.

The following Limiting Conditions and Surveillance Requirements are proposed for additions to the Technical Specifications:

1. Quarterly cycling of MO-4085A, MO-4085B and MO-4086. In addition, these valves will be included in the Monticello ASME Code, Section XI, Pump and Valve Testing Program.

2. Requirement for operability of MO-4085A and MO-4085B at all times. In the event of an inoperable valve, actions are specified to assure full LPCI flow.

3. Limitations on use of the intertie line. Flow will not be allowed in the line if the reactor is in the run mode.

In addition, appropriate changes to the basis have been provided explaining these changes.

SIGNIFICANT HAZARDS EVALUATION

The RHR intertie line has been evaluated for potential adverse effects on plant safety. The evaluation, performed by the General Electric Company, is provided in Exhibit C.

The following potential adverse effects were identified:

1. An impact of the 10 CFR Part 50, Appendix K, analysis due to an increase in design basis accident (DBA) break area equal to a four-inch line (0.08 square feet).

2. An additional flow path between the broken and unbroken recirculation loops affects core flow and may cause early boiling transition during a loss of coolant accident (LOCA).

3. An increase in containment peak pressure and temperature due to the larger DBA break size.

4. An increase in containment suppression pool loads (Mark I Long Term Program considerations) due to the larger DBA break size.

5. When the intertie line is in use, measured recirculation drive flow will be slightly greater than drive flow delivered to the jet pumps. Flow biased scrams and rod blocks may be affected.

These effects were analyzed and the following conclusions were drawn:

1. There is no impact on the Appendix K analysis. The limiting break size from the most recent analysis is the 34% DBA (1.36 square feet) which is not affected by the new intertie line.

2. Core flow reduction during a LOCA could be compensated for by changing the range of application of MAPHLGR multipliers as explained in Exhibit C. We will not use the intertie line with the reactor in the run mode, thereby eliminating this concern.

3. The increase in containment peak pressure and temperature is negligible.

4. The increase in suppression pool loading is negligible.

5. We will not use the intertie line with the reactor in the run mode. Therefore, since the flow biased scram and rod blocks are not required to be operable in this plant condition, there is no need to adjust the flow biased scram and rod blocks.

The RHR intertie line will significantly reduce the potential for water hammer events at Monticello and perform other operationally useful functions. A detailed safety evaluation has been performed to analyze potential adverse effects on plant safety. No adverse effects will result from this change.

The proposed Technical Specification changes constitute an additional limitation, restriction, or control not presently included in the Technical Specifications for a new design enhancement. This change has been found not to adversely affect plant safety. For these reasons, operation of the Monticello Nuclear Generating Plant in accordance with the proposed changes would not:

(1) Involve a signficant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

2. Recirculation System Cross-Tie

PROPOSED CHANGES
Remove Technical Specifications in Section 3.5.I and the bases as shown in Exhibit B, pages 17, 114 and 119. Retain existing restrictions on plant operation with less than two recirculation loops.

This section will be further revised in the future when other Technical Specification changes currently under review by the NRC Staff related to single loop operation are issued.

REASONS FOR CHANGE
The recirculation system cross-tie line and associated valves are being removed as part of the recirculation piping replacement program. The line is no longer used at Monticello and is not being installed in new design plants.

Technical Specification changes issued several years ago related to implementation of 10 CFR Part 50, Appendix K, have precluded use of the cross-tie line during plant operation.

Existing Technical Specifications related to the cross-tie line are therefore unnecessary and should be deleted. Limitations on plant operation with less than two recirculation loops have been retained until NRC review of single loop operation has been granted.

SIGNIFICANT HAZARDS EVALUATION

Technical Specification Limiting Conditions for Operation prohibit use of the cross-tie line during plant operation. The cross-tie line no longer serves any purpose and is being removed.

Therefore, this change is administrative in nature, reflecting removal of a plant feature that cannot be used as a result of earlier NRC requirements. For this reason, operation of the Monticello Nuclear Generating Plant in accordance with the proposed changes would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.