UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT DOCKET NO. 50-263

REQUEST FOR AMENDMENT TO
OPERATING LICENSE DPR-22

LICENSE AMENDMENT REQUEST DATED January 3, 1994

Northern States Power Company, a Minnesota corporation, requests authorization for changes to Appendix A of the Monticello Operating License as shown on the attachments labeled Exhibits A, B, and C. Exhibit A describes the proposed changes, describes the reasons for the changes, and contains a Safety Evaluation, a Determination of Significant Hazards Consideration and an Environmental Assessment. Exhibit B contains current Technical Specification pages marked up with the proposed changes. Exhibit C is a copy of the Monticello Technical Specifications incorporating the proposed changes.

This letter contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By Roger O Anderson

Roger O Anderson
Director
Licensing & Management Issues

On this 3rd day of January 1994 before me a notary public in and for said County, personally appeared Roger O Anderson, Director, Licensing and Management Issues, 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.

Judy L Klapperick

JUDY L. KLAPPERICK
NOTARY PUBLIC-MINNESOTA
ANOKA COUNTY
My Commission Expires Sept. 29, 1997

Exhibit A

MONTICELLO NUCLEAR GENERATING PLANT

License Amendment Request Dated January 3, 1994

Evaluation of proposed changes to the Technical Specifications for Operating License DPR-22

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

Proposed Changes:

The reactor pressure relief system consists of eight safety/relief valves (RV-2-71A thru H) located on the main steam lines within the drywell between the reactor vessel and the first main steam isolation valve. The eight safety/relief valves are self-actuating at high pressure to provide overpressure protection for the reactor vessel and associated piping. Certain groups of the valves may also be manually (remotely) or automatically actuated to control reactor pressure under accident or transient conditions.

Technical Specification Surveillance Requirement 4.6.E.1.a currently specifies that a minimum of seven safety/relief valves shall be bench checked or replaced with a bench checked valve each refueling outage. The basis for this specification, as discussed on page 150 of the Technical Specifications, is to ensure that any valve deterioration is detected.

The proposed amendment involves changing this surveillance requirement to simply require the valves to be tested in accordance with Section XI of the ASME Boiler and Pressure Vessel Code. Specifically, we propose revising Specification 4.6.E.1.a to read as follows:

> "4.6.E.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."

Technical Specification 4.15.B addresses Inservice Testing requirements. The valves are currently within the scope of the Inservice Testing requirements of the ASME Code, which contains specific surveillance testing requirements for Class 1 pressure relief devices by invoking ANSI/ASME OM-1-1981. The proposed change is consistent with Surveillance Requirement 3.4.3.1 of the Improved Standard Technical Specifications, NUREG-1433, which also specifies safety/relief valve surveillance testing in accordance with Inservice Testing requirements.

Reason for Changes:

The proposed change was identified as a Cost Beneficial Licensing Action (CBLA) in that it will allow the plant staff to take advantage of the flexibility currently contained in the ASME Code concerning relief valve testing frequency. 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."

This requirement is less prescriptive than the current requirements of Specification 4.6.E.1.a, and its application will allow the plant staff to apply more discretion in scheduling the replacement and testing of individual safety/relief valves. This will provide several benefits, such as;

- Improved ALARA performance through a reduction in the required frequency of safety/relief valve topworks replacement and testing. Much of this takes place in high dose rate areas within the drywell. The dose received performing this surveillance requirement during the 1993 refueling outage was in excess of 23 man-Rem. We feel the accumulation of such a high dose is not warranted for this surveillance, since our operating experience leads us to conclude that it does not result in a commensurate increase in safety/relief valve reliability. The proposed change would allow us more flexibility to reduce the scope of this work for any given outage, and to shift this work between outages when prudent, in order to take advantage of factors that can reduce drywell dose such as changes in plant chemistry and scheduled decontamination efforts.

- Improved performance and reliability of the safety/relief valves. Although we cannot quantify this benefit, root cause analysis of past valve problems supports the conclusion that it is preferable to minimize any disassembly or handling of the valves, since work of this nature has the potential for introducing foreign material into the valve and increases the risk of damaging the valve internals or valve seats. Any of these conditions can cause seat leakage and/or disturb valve setpoints. Furthermore, it is desirable to minimize the frequency of full flow steam testing in order to reduce wear and tear on the valves, since each valve is typically lifted more than 10 times during the course of as-found and as-left testing.

- Reductions in outage costs. Safety/relief valve testing and replacement is costly (roughly $23,000 per valve each outage). Much of the work must be performed inside the drywell, which reduces the productivity and efficiency of maintenance personnel due to the need to wear full anti-contamination clothing. As noted previously, this is also a high dose

rate area, which limits personnel stay times and further reduces efficiency. In addition, it is currently necessary to ship the valve topworks off-site to one of only a few specialized test facilities that can accommodate full steam flow testing of the valves in order to determine the as-found and as-left setpoints. These factors contribute significantly to increasing the overall cost of this work. Implementation of the proposed Technical Specification change would allow us to avoid and/or defer some of these costs by allowing the option of spreading valve testing out over two or possibly three consecutive refueling outages, rather than replacing a minimum of 7 of 8 valve topworks each outage.

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 Technical Specification 4.6.E.1.a only requires this to be done for seven of the valves. This practice was adopted because the differences between the Technical Specification requirements and ASME Code requirements added complexity to this issue that made it difficult to ensure compliance with both documents if less than all 8 valves were replaced. It was deemed preferable to incur the incremental cost of performing maintenance on the eighth valve rather than run the risk of violating either the Technical Specifications or the ASME Code due to a program administration oversight. The proposed change will eliminate this concern by making the ASME Code the governing requirement. The proposed change will also present the plant staff with the option of significantly reducing the scope of safety/relief valve work for a given outage. For example, the plant staff could choose to bench check or replace four of the eight valves each outage, such that all eight valves would be tested within a 2 cycle period. If warranted by unusual circumstances, such as abnormally high drywell dose rates or unusually short outage windows, the plant staff could opt to disassemble and test or replace as few as two valves during a given outage provided the minimum required test frequency specified in ANSI/ASME OM-1-1981 was satisfied. It should also be noted that the proposed change would not preclude the plant staff from continuing the current practice of bench checking or replacing all eight safety/relief topworks during each refueling outage, since this option is also allowed by ANSI/ASME OM-1-1981 and may prove to be the best choice in some situations.

Safety Evaluation:

As discussed in the basis for the Monticello Technical Specifications, the reactor coolant system safety/relief valves assure that the reactor coolant system pressure safety limit is never reached. In compliance with Section III of the ASME Boiler and Pressure Vessel Code, 1965 Edition, the safety/relief valves must be set to open at a pressure no higher than 105 percent of design pressure, and they must limit the reactor pressure to no more than 110 percent of design pressure. The safety/relief valves are sized according to the Code for a condition of Main Steam Isolation Valve (MSIV) closure while operating at 1670 MWt, followed by a scram from indirect (high flux) means. With the safety/relief valves set as specified, the maximum vessel pressure remains below the 1375 psig ASME Code limit. Only seven of the eight valves are

assumed to be operable in this analysis and the valves are assumed to open at 1% above their setpoint with a 0.4 second delay.

In addition to the high pressure self-actuation function, all eight valves can be remotely operated manually from the control room. Three of these valves (RV-2-71A, C & D) are also used by the Automatic Depressurization System (ADS) as a back-up to the HPCI system during a small-break Loss of Coolant Accident (LOCA). ADS allows rapid depressurization of the reactor vessel, thereby enabling the low head Core Spray and Low Pressure Coolant Injection (LPCI) systems to inject water into the vessel to protect the fuel cladding. Another three of the safety/relief valves (RV-2-71E, G & H) provide a Low-Low set function that controls the opening and closing setpoint of the valves following a scram to maximize the time between pressurization transients. A different valve (RV-2-71F) can be controlled from the Alternate Shutdown System (ASDS) control panel to facilitate plant shutdown in the event of a control room fire. The proposed changes will have no impact on the function or operation of the safety/relief valves as described above or in the Monticello Updated Safety Analysis Report (USAR) or the Technical Specifications, therefore the change will have no impact on any accident or transient analysis. The health and safety of the public will not be affected by this change.

As discussed in Section 4.4.4 of the USAR, it is recognized that it is not feasible to test the safety/relief valve setpoints while the valves are in place or during normal plant operation. The valves are mounted on 6-inch-diameter 1500-lb primary service rating flanges so that they may be removed for maintenance or bench checks and reinstalled during normal plant shutdowns. In addition, the main seats of at least two safety/relief valves are inspected during each refueling outage as required by Technical Specification 4.6.E.1.b.

The proposed change will have no significant adverse impact on the reliability of the safety/relief valves. Although any single valve may ultimately be tested or changed out less frequently, we expect overall reliability may actually increase since the most common performance problems associated with these valves, seat leakage and setpoint drift, can be induced by valve handling or disassembly work. The proposed change would enable us to minimize such work to the extent permitted by Section XI of the ASME Code and ANSI/ASME OM-1-1981. Based on our experience with the safety/relief valves, as well as balance of plant relief valves in similar applications, we conclude that increasing the interval between bench tests will not significantly affect valve performance. This is supported by the fact that 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. As before, all eight safety/relief valves will be exercised each refueling outage in accordance with ASME Section XI requirements (as modified by Monticello Third 10-Year IST Program Relief Request NB-1) during plant startup.

The proposed change is consistent with Surveillance Requirement 3.4.3.1 of the Improved Standard Technical Specifications, NUREG-1433, which specifies the frequency of safety/relief valve testing to be in accordance with Inservice Testing requirements. Thus, the acceptability of using the Inservice Testing

requirements of Section XI of the ASME Code for safety and relief valve testing has already been confirmed.

If the proposed change is approved, it will be necessary to process a revision to Third Ten-Year Inservice Testing Program Relief Request NB-1. Relief Request NB-1 provides an exception from the quarterly valve exercising tests specified in the ASME Code (Section XI, Subarticle IWV-3410) in recognition of the fact that testing of the subject valves during power operation is impractical. However, the language of the relief request reflects our past practice by stating that all eight safety/relief valves are bench checked or replaced each outage, which would not necessarily be true if the proposed Technical Specification amendment is approved. The necessary changes to Relief Request NB-1 would be handled via a separate submittal on a routine basis.

Based on the above discussion, we conclude that the proposed change is technically acceptable and does not adversely affect safety. In addition, we conclude that this amendment does not involve any significant increase in the types or amounts of effluents released from the site and therefore has no significant environmental impact.

Determination of Significant Hazards Consideration:

This proposed change to the Operating License has been evaluated to determine if it constitutes a significant hazards consideration as required by 10 CFR Part 50, Section 50.91 using the standards provided in Section 50.92. This analysis is provided below:

a. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment is limited to changes to the surveillance testing requirements (bench checking or replacement) applicable to the main steam system safety/relief valves. This surveillance requirement is performed while the plant is in a cold shutdown condition at a time when the safety/relief valves are not required to be operable. The performance of this evolution is not an input or consideration in any accident previously evaluated, thus the proposed change will not increase the probability of any such accident occurring. Current safety analyses conclude that the pressure relief capabilities of the Safety Relief valves are adequate assuming that one of the eight safety/relief valves fails to open upon demand. The proposed change will not adversely affect the reliability of the valves and will therefore not reduce the conservatism of this assumption.

Similarly, the proposed amendment specifies testing requirements consistent with accepted industry codes and regulatory guidance to provide assurance that the valves will function as designed. The amendment will not diminish the capability of the safety/relief valves to perform as required during any accident previously evaluated and will therefore not increase the consequences of any such accident.

b. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

The proposed amendment does not involve any modification to plant equipment or operating procedures, nor will it introduce any new safety/relief valve failure modes that have not been previously considered. The net result of the proposed amendment will be to allow the plant staff the option of decreasing the frequency of safety/relief valve testing to a level that has been acknowledged as acceptable by the ASME Code and NUREG-1433. We therefore conclude the proposed changes will not create the possibility of a new or different kind of accident from any accident previously analyzed.

c. The proposed amendment will not involve a significant reduction in the margin of safety.

The proposed amendment does not involve a decrease in the number or capacity of safety/relief valves that are provided in the system, nor does it involve any change in safety/relief valve setpoints, operability requirements, or limiting conditions for operation. Based on these considerations, we conclude the proposed amendment will not involve a significant reduction in the margin of safety

Based on the evaluation described above, and pursuant to 10 CFR Part 50, Section 50.91, Northern States Power Company has determined that operation of the Monticello Nuclear Generating Plant in accordance with the proposed license amendment request does not involve any significant hazards considerations as defined by NRC regulations in 10 CFR Part 50, Section 50.92.

Environmental Assessment:

Northern States Power has evaluated the proposed changes and determined that:

1. The changes do not involve a significant hazards consideration,

2. The changes do not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or

3. The changes do not involve a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR Part 51 Section 51.22(c)(9). Therefore, pursuant to 10 CFR Part 51 Section 51.22(b), an environmental assessment of the proposed changes is not required.