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 26, 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 26th 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 31, 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:

10 CFR Part 50, Appendix J (Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors) specifies the requirements for periodic primary containment leak rate testing. The leak rate testing program at Monticello substantially conforms to these requirements, however, the design of the Monticello plant predates Appendix J. As discussed on page 184 of the Monticello Technical Specifications (Section 4.7.A Bases), the design of the plant was reviewed to determine where compliance with Appendix J was impossible or impractical. In each case where a departure from the requirements of Appendix J was identified, a request for exemption from the requirements of Appendix J or a plant modification was proposed and submitted for NRC Staff review. Exemptions were proposed in those cases where compliance with Appendix J would have provided no meaningful improvement in plant safety. In their review of Appendix J compliance, the NRC Staff approved a number of exemption requests, denied others, and provided necessary interpretation and clarification of the requirements of Appendix J. The current Technical Specification surveillance requirements reflect the results of this review.

One of the exemptions granted involved low pressure testing of the main steam isolation valves. The four main steam lines which penetrate the primary containment each have two 18 inch diameter isolation valves installed in series (one inside and one outside of the drywell), for a total of eight main steam line isolation valves. Appendix J would normally require that such valves be subjected to a leakage test at a test pressure of P_a, the calculated peak containment internal pressure related to the design basis loss-of-coolant accident (for Monticello, P_a is calculated to be 42 psig). However, Technical Specifications 3.7.A.2.b.3 and 4.7.A.2.b.5 provide an exemption from this requirement and specify a test pressure of greater than or equal to 25 psig for the main steam line isolation valves.

The original reason for this exemption, as discussed in the NRC's Appendix J Safety Evaluation Report for Monticello dated June 3, 1984, concerned the test methodology used at the time. Leak rate testing of the main steam isolation valves was performed by applying test pressure between the two isolation

valves in a main steam line and measuring the total combined leakage past the two valves. The test method caused the test pressure to act on the underside of the inboard valve disc from below the seat. Test pressures in excess of 25 psig tended to unseat the inboard valve, rendering the test meaningless. In addition, since it was not possible to allocate the portion of the total leakage attributable to individual valves, it was conservatively assumed that each valve in the pair was leaking at the combined rate. This conservatism made it more difficult to meet the test acceptance criteria of 11.5 SCFH per valve since it tended to overestimate the leak rate for individual valves, thus any added leakage induced by the test method could adversely bias the test results.

Since that time, the development of inflatable main steam line plugs capable of holding against a pressure in excess of P_a has enabled us to develop and implement improved test methodology whereby the full test pressure can be applied in the proper direction (above the disc) for all eight main steam isolation valves. As an added benefit, we can now measure the leak rate for each valve individually. As a result of these improvements, there is no longer any need or justification for an exemption from the requirements of Appendix J to allow low pressure testing of the main steam isolation valves.

In consideration of the above, we propose to amend Technical Specifications 3.7.A.2.b and 4.7.A.2.b, as well as the associated Bases for Section 3.7 and 4.7, to reflect the fact that the main steam isolation valves can now be tested at a pressure of greater than or equal to P_a (42 psig). The test leakage acceptance criteria would be increased from 11.5 SCFH to 25.3 SCFH to reflect the higher test pressure. The revised leakage criterion of 25.3 SCFH has been calculated using the methodology of Appendix A to Technical Evaluation Report TER-C5257-30, "Containment Leakage Testing - Northern States Power Company, Monticello Nuclear Generating Plant", dated April 7, 1982. This TER was used by the NRC as the bases for reviewing our current Appendix J program, and is referenced by the NRC in the June 3, 1984 Monticello Appendix J Review Safety Evaluation Report and associated Appendix J Exemption Request approval.

The change in test pressure and acceptance criteria differ from the equivalent surveillance requirement of NUREG-1433, but this is because the NUREG assumes an exemption from Appendix J is needed to provide a practical alternative to the Appendix J test criteria. This assumption is no longer true for Monticello. The specific changes we are proposing are detailed in Exhibits B and C.

Reason for Changes:

The proposed changes to the main steam isolation valve leak rate test pressure and acceptance criteria are being submitted to eliminate an exemption to 10 CFR Part 50, Appendix J that is no longer technically necessary or justifiable, thereby ensuring Appendix J requirements are met to the extent practical.

Safety Evaluation:

The Monticello primary containment system consists of a drywell, which encloses the reactor vessel and recirculation pumps, a pressure suppression chamber which stores a large amount of water, a connecting vent system between the drywell and the suppression chamber, and isolation valves. One of the functions of the primary containment system is to provide a barrier which, in the event of a loss-of-cooling accident, controls the release of fission products to the secondary containment. In order to verify that the primary containment system is capable of fulfilling this function, periodic leak rate testing is performed using methodology that complies, to the extent practical, with the requirements of 10 CFR Part 50, Appendix J. The design, function and testing of the primary containment system is discussed in Section 5.2 of the Monticello Updated Safety Analysis Report.

The proposed amendment will have no adverse impact on the ability of the main steam isolation valves to perform their intended safety function. The proposed test criteria will continue to provide a means to detect valve degradation and will not impact the validity of the test results, since the test will simulate challenging the valves under worst case accident conditions. Utilization of the higher test pressure P_a (42 psig) would also comply with 10 CFR Part 50, Appendix J requirements.

Use of a higher allowable leakage (25.3 SCFH per valve) for a test conducted at P_a (42 psig) is also appropriate, since this is equivalent to the previously specified leakage rate of 11.5 SCFH when tested at 25 psig. The methodology of Appendix A to TER-C5257-30, which was used to determine the new leakage criteria, was previously reviewed by the NRC and found to be acceptable. The revised leak rate criteria will not result in higher main steam isolation valve leakage rates under actual service or accident conditions. The proposed change will therefore have no impact on the plant's ability to meet Technical Specification 3.7.A.2.b.1, which requires that the Primary Containment overall integrated leakage rate be less than or equal to L_a, 1.2 percent by weight of the containment air per 24 hours at P_a, 42 psig.

The proposed changes will have no impact on the function, operation, or reliability of the main steam isolation valves as described above or in the Monticello Updated Safety Analysis Report (USAR) or the Bases for the Technical Specifications, therefore the change will have no impact on any accident or transient analysis, nor will it increase the likelihood of an accident occurring. The proposed change will not result in any increase in post accident off-site or on-site radiation dose, since the adjusted leakage limit is consistent with inputs previously established for the dose analyses.

Based on the above discussion, we conclude that the proposed change is technically acceptable and does not adversely impact public health or 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 (test pressure and allowable leakage criteria) applicable to the main steam line isolation valves. The proposed criteria are equivalent to the current criteria with respect to monitoring main steam isolation valve performance to ensure that leakage past the valves would be within acceptable limits under accident conditions. This surveillance test is performed while the plant is in a cold shutdown condition at a time when the main steam isolation valves are not required to be operable. Performance of the test itself 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.

The proposed amendment will not adversely affect the function, operation, or reliability of the valves, nor will it diminish the capability of the valves to perform as required during an accident. There will be no increase in post accident off-site or on-site radiation dose, since the adjusted leakage limit is consistent with inputs previously established for the dose analyses. The proposed amendment is consistent with regulatory requirements (10 CFR Part 50, Appendix J) and guidance (TER-C5257-30) that has been previously reviewed by the NRC and found to be acceptable. Therefore, the amendment will not increase the consequences of any accident previously evaluated.

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 main steam isolation valve failure modes that have not been previously considered. The proposed amendment is limited to a change in the surveillance test pressure & acceptance criteria used to leak test the valves. This test is performed while the plant is in a cold shutdown condition at a time when the valves are not required to be operable. 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 will result in the main steam isolation valves being subjected to the maximum pressure (P_a, 42 psig) calculated to occur under worst case accident conditions, and will therefore provide a more realistic and challenging test of valve performance under those conditions. The leakage rate criteria for the test has been adjusted upward to be commensurate with the higher test pressure, but this does not represent any increase in actual leakage under accident conditions. On-site and off-site dose analyses will not be affected. The proposed amendment does not involve any change in operability requirements or limiting conditions for operation beyond the replacement of the old test pressure & acceptance criteria with equivalent criteria consistent with 10 CFR Part 50, Appendix J, NUREG-1433, and TER-C5257-30. 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.

Exhibit B

Monticello Nuclear Generating Plant

License Amendment Request Dated January 26, 1994

Technical Specification Pages Marked Up
with Proposed Wording Changes

Exhibit B consists of the existing Technical Specification pages marked up with the proposed changes. Existing pages affected by this change are listed below:

Page
159
160
184