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 July 7, 1993

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 and Management Issues

On this 7th day of July 1993 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 July 7, 1993

Evaluation of Proposed Changes to the Technical Specifications
for Operating License DPR-22

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

Page	Section	Current Specification
126	3.6.D.5	At least one of the leakage measurement instruments associated with each sump shall be operable and the drywell particulate radioactivity monitoring system shall be operable or a sample of the containment atmosphere shall be taken and analyzed at least every four hours. Otherwise, initiate an orderly shutdown of the reactor and reduce reactor water temperature to less than 212°F within 24 hours.
		Proposed Change
	3.6.D.5	Any time irradiated fuel is in the reactor vessel and reactor water temperature is above 212°F, at least one of the leakage measurement instruments associated with each sump shall be operable. If no leak rate measurement instruments associated with a sump are operable, then: a. Perform manual leak rate measurements once per 12 hours and restore a measurement instrument to operable status within 30 days. b. Otherwise, initiate an orderly shutdown of the reactor and reduce reactor water temperature to less than 212°F within 24 hours.
	3.6.D.6	Any time irradiated fuel is in the reactor vessel and reactor water temperature is above 212°F, the drywell particulate radioactivity monitoring system shall be operable. If the drywell particulate radioactivity monitoring system is not operable, then: a. Analyze grab samples of the primary containment atmosphere once per 12 hours.

b. Otherwise, initiate an orderly shutdown of the reactor and reduce reactor water temperature to less than 212°F within 24 hours.

Reason for Proposed Change

Staff position number three of Generic Letter 88-01, Supplement 1 provides for an alternative means to quantitatively measure reactor coolant system leakage via manually pumping the sump or measuring differences in sump level for a period of 30 days while restoring the drain sump monitoring system. The alternative staff position is reflected in the revised wording of Limiting Condition for Operation requirement 3.6.D.5.a. The proposed change reduces the burden placed on plant operations personnel to record indications of potential coolant system leakage and reduces the potential for unnecessary plant shutdowns by allowing manual determination of coolant system leakage consistent with NRC Staff positions stated in Generic Letter 88-01, Supplement 1.

Requirement 3.6.D.6 incorporates Limiting Conditions for Operation for the drywell particulate radioactivity monitoring system which were previously contained in requirement 3.6.D.5. Requirement 3.6.D.6 revises the grab sample analyses frequency to once every 12 hours during drywell radioactive particulate monitor inoperability to be consistent with standard technical specification requirements contained in NUREG-1433, BWR/4 Improved Standard Tech Specs.

Page	Section	Current Specification
126	4.6.D.1	Any time irradiated fuel is in the reactor vessel and coolant temperature is above 212°F, the following surveillance program shall be carried out: a. Unidentified and Identified Leakage rates shall be recorded at least once every 4 hours using primary containment floor and equipment drain sump monitoring equipment. b. Primary containment atmospheric particulate radioactivity shall be recorded at least once every 4 hours. c. Drywell pressure and temperature shall be recorded at least once every 12 hours.

Proposed Change

Any time irradiated fuel is in the reactor vessel and coolant temperature is above 212°F, the following surveillance program shall be carried out:

a. Unidentified and Identified Leakage rates shall be recorded once per shift not to exceed 12 hours using primary containment floor and equipment drain sump monitoring equipment.

Reason for Proposed Change

Staff position number one of Generic Letter 88-01, Supplement 1, endorses a frequency of once per shift not to exceed 12 hours for monitoring reactor coolant system leakage. The revised frequency for Surveillance Requirement 4.6.D.1.a and the deletion of requirements 4.6.D.1.b and 4.6.D.1.c reduces the burden placed on plant operations personnel to record indications of potential coolant system leakage consistent with NRC staff guidance provided in Generic Letter 88-01, Supplement 1, and NUREG-1433, BWR/4 Improved Standard Tech Specs.

Current Specification

126 4.6.D.2 b. Primary containment sump leakage measurement system - performance of a sensor check at least once per 4 hours and a channel calibration test at least once per cycle.

Proposed Change

b. Primary containment sump leakage measurement system - performance of a sensor check at least once per shift not to exceed 12 hours and a channel calibration test at least once per cycle.

Reasons for Proposed Change

Surveillance requirement 4.6.D.2.b has been revised to require a sump leakage measurement system sensor check to be performed once per shift not to exceed twelve hours. The proposed change reduces the burden placed on plant operations personnel to record indications of potential coolant system leakage consistent with NRC staff guidance provided in NUREG-1433, BWR/4 Improved Standard Tech Specs.

Safety Evaluation:

Requirements governing reactor coolant system leakage detection were added to the plant Technical Specifications on December 10, 1982 via License Amendment 14 and 17. License Amendment 14 was issued as part of the corrective actions and justification for returning the Monticello Nuclear Plant to power following a Confirmatory Action Letter issued by the Commission on October 19, 1982. The October 19, 1982 Confirmatory Action Letter was issued regarding proposed corrective actions for crack indications found in welds on the Monticello Nuclear Plant recirculation system. Since that time, the Monticello Nuclear Plant has either replaced piping susceptible to Intergranular Stress Corrosion Cracking (IGSCC) in the recirculation system, the residual heat removal system, and the core spray system with materials resistant to IGSCC, or protected the piping with a cladding of resistant weld metal. To further reduce susceptibility to IGSCC, a Hydrogen Water Chemistry System was placed in operation in 1988. As a result of these modifications, the potential for IGSCC (and therefore the safety significance of the leak detection system) has been greatly reduced.

In 1988, the Nuclear Regulatory Commission issued Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping." This Generic Letter provided revised Staff Positions to minimize and control intergranular stress corrosion cracking (IGSCC) in BWR piping systems made of austenitic stainless steel that is four inches or larger in nominal diameter and contains reactor coolant at a temperature above 200°F. Generic Letter 88-01 required confirmation that plant Technical Specifications related to leakage detection

would conform to the staff position on leak detection included in the Generic Letter. In our response to Generic Letter 88-01 dated July 28, 1988, we confirmed that our plant Technical Specifications conformed to the NRC staff position on leak detection as incorporated via License Amendments 14 and 17 to the Monticello Operating License. This License Amendment Request does not change the reactor coolant system leakage limits specified in Generic Letter 88-01.

Subsequent to the above correspondence, the NRC has issued Supplement 1 to Generic Letter 88-01 and NUREG-1433, BWR/4 Improved Standard Tech Specs. In these documents the NRC has provided revised guidance for monitoring reactor coolant system leakage which ensures means are available for detecting reactor coolant system leakage while reducing unnecessary hardship on plant operators and the potential for undesirable plant transients due to unnecessary plant shutdowns. This License Amendment Request incorporates the applicable revised guidance.

The revised Limiting Conditions for Operation requirements for inoperability of a sump leak rate measurement instrument and the drywell particulate radioactivity monitoring system are acceptable based on the multiple forms of leakage detection that are still available. Determination of the coolant system leakage during periods when the sump leakage rate measurement instrument is inoperable, can be accomplished by manually pumping the sump or by observation of the sump level change, using indications that are readily available to the operating staff. These instruments have an accuracy which is suitable for detecting changes in reactor coolant system leakage consistent with specified leakage limits. The 12 hour interval for manual leak rate determination or grab sample analysis provides periodic information that is adequate to detect leakage. The 30 day completion time for restoration of the sump leakage measurement instrument or the drywell particulate radioactivity monitoring system recognizes that at least one other form of leakage detection is available.

The level of safety provided to the public is maintained with extension of the frequency for recording the coolant system leakage rates and performance of sump leakage measurement system sensor check to once per shift not to exceed 12 hours, and deletion of the surveillance requirement to record drywell particulate radioactivity and drywell pressure and temperature. Alarms are provided to alert the operator to conditions indicative of a potential coolant system leak via the drywell particulate radioactivity monitoring system as well as alarms to notify the operator of conditions approaching the Technical Specification leakage limits. The drywell particulate radioactivity and drywell pressure and temperature indications provide corroboration of the primary indication of potential breaches in reactor coolant system integrity, the sump leakage measurement instruments. The established operating limits for reactor coolant system leakage as monitored by the sump leakage measurement instruments are well below the rates predicted for critical crack growth and instability.

Determination of Significant Hazards Consideration:

The proposed changes to the Operating License have been evaluated to determine whether they constitute 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:

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

   The proposed changes incorporate the guidance of Generic Letter 88-01, Supplement 1, and NUREG-1433, BWR/4 Improved Standard Tech Specs, into the Technical Specifications governing reactor coolant system leakage monitoring. Piping susceptible to Intergranular Stress Corrosion Cracking (IGSCC) in the recirculation system, the residual heat removal system, and the core spray system has been replaced with material resistant to IGSCC, or protected with a cladding or resistant weld metal. To further reduce susceptibility to IGSCC, a Hydrogen Water Chemistry System was placed in operation in 1988. The above actions taken to minimize the potential for crack initiation in conjunction with the leakage rates allowed by the Technical Specifications and the multiple leakage indication systems available, maintains a high level of confidence in the ability to monitor reactor coolant system integrity, thus the proposed changes will not significantly affect the probability or consequences of an accident previously evaluated.

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

   There are no new failure modes or mechanisms associated with the proposed changes. The proposed changes do not involve any equipment modifications or changes in operational limits. Only the means and frequency for confirming compliance with the limits are affected. The operating limits are established well below the rates predicted for critical crack growth and instability. Indications remain available to the operators to provide an early warning of a significant reactor coolant pressure boundary leak. It is therefore concluded that the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated, and the accident analyses presented in the Updated Safety Analysis Report will remain bounding.

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

   The proposed changes do not involve any modification in operational limits. Alarm functions to alert the operator of reactor coolant system leakage, and operator monitoring of key parameters is maintained at a level to assure early detection of any significant leakage. It is

therefore concluded that the proposed changes will not result in any reduction in the plant's margin of safety.

Based on the analysis above, and pursuant to 10 CFR Part 50, Section 50.59, 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 regulation in 10 CFR Part 50, Section 50.92.

Environmental Assessment

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

1. The change does not involve a significant hazards consideration.

2 The changes do not involve a significant change in the type or significant increase in the amounts of any effluent 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 met the eligibility criterion for categorical exclusion set forth in 10 CFR Part 51, Section 51.22(b), an environmental assessment of the proposed changes is not required.