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 August 15, 1995

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 William J Hill
William J Hill
Plant Manager
Monticello Nuclear Generating Plant

On this 15 day of August 95 before me a notary public in and for said County, personally appeared William J Hill, Plant Manager, Monticello Nuclear Plant, 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.

Stephen Blegen

STEPHEN R. BLEGEN
NOTARY PUBLIC - MINNESOTA
My Comm. Exp. Jan. 31, 2000

Exhibit A

MONTICELLO NUCLEAR GENERATING PLANT

License Amendment Request Dated August 15, 1995

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 Change (Part 1)- Combined MSIV Leakage:

Revise specifications as noted in Exhibits B and C to have main steam line isolation valve leak rate test acceptance criterion based on combined maximum flow path leakage for all four main steam lines of 46 scfh in lieu of the current limit of 11.5 scfh per valve.

Reason for Change:

The purpose of the proposed change is to establish a more appropriate acceptance criterion for main steam line isolation valve leak rate testing. The current acceptance criterion is unnecessarily restrictive and results in unwarranted personnel radiation exposure and valve repair work.

Background:

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 four main steam lines which penetrate the primary containment each have two 18" diameter isolation valves installed in series (one inside and one outside of the drywell), for a total of eight main steam line isolation valves. Type C leak rate testing of the main steam line isolation valves and other primary containment isolation valves (with certain approved exemptions) is performed in accordance with the requirements of 10 CFR Part 50 Appendix J to verify that leakage through these paths is within acceptable limits. 10 CFR Part 50 Appendix J requires that the combined leakage of all penetrations and valves subject to type B and C tests shall be less than 0.60 L_a. L_a represents the maximum allowable containment leak rate, which for Monticello is 1.2% (by weight)

of the containment air per day at pressure P_a. P_a is the calculated peak containment internal pressure related to the design basis loss-of-coolant accident, which for Monticello is calculated to be 42 psig. However, Technical Specification 3.7.2.b.2 excludes the contribution of main steam isolation valve leakage from the 0.60 L_a combined limit. Instead, the Technical Specification limits each main steam line isolation valve to a maximum leak rate of 11.5 scfh at 25 psi.

Safety Evaluation:

From an analytical standpoint, there is essentially no difference between the current main steam line isolation valve leak rate (11.5 scfh/valve) and the proposed criterion (46 scfh maximum combined flow path leakage), since in either case the maximum allowable leak rate for the 4 main steam lines combined is limited to 46 scfh at 25 psig. The overall allowable leakage with the proposed change is no different than presently exists, thus post accident doses remain unchanged. From a practical standpoint, the proposed criterion is superior to the old in several respects.

1. The proposed criterion will result in a higher probability of a successful test by, in essence, allowing an "averaging" of the leak rate. Valves that would have marginally failed the test in the past could now be accepted by taking credit for other valves that are below the maximum leak rate. In the extreme case, the combined leakage could be due to a single steam line (i.e., one steam line with a maximum pathway leakage rate of 46 scfh, and the remaining three lines at 0.0 scfh). Such a case, although unlikely, would be acceptable and does not represent excessive valve leakage through the main steam line isolation valves.

   The BWR Owners Group has evaluated the feasibility of increasing the Technical Specification limit for the main steam line isolation valve leakage for several years. The BWR Owners Group provided funding for General Electric to research this issue, and NEDC-31858P Revision 2, September 1993 (BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems) was issued to report the results of this effort. NEDC-31858P Revision 2 concluded, among other things, that main steam line isolation valve leakage could be increased to 200 scfh per main steam line without inhibiting the safety function of the valve. Section 4.2 of the report notes that valve manufacturers have stated that "large and fast acting main steam line isolation valves are expected to have no substantial defects at leakage rates up to 200 scfh". In other words, a leak rate of 200 scfh does not represent abnormal or excessive leakage for a valve of this size and type. Valve leakage of 46 scfh would not, therefore, indicate a serious condition.

   Section 4.2 of NEDC-31858P Revision 2 also states that the BWROG found that disassembly and refurbishment of main steam line isolation valves to meet unnecessarily low leakage limits frequently contributes to repeated failures. The report goes on to list several examples of possible maintenance induced defects, such as machining induced seat cracks, machining of guide ribs, excessive pilot valve seating machining, and mechanical defects induced by assembly and disassembly. The report also suggests that, by attempting to correct non-existent or minimal defects in the valves, it is likely that some defects may actually be introduced that lead to later valve failures.

NSP concurs with the above assessment, and is further concerned that frequent and repeated seat lapping performed to achieve unnecessarily low leak rates results in premature depletion of the available seat material, requiring major valve repair or replacement efforts. This was the case during the 1994 refueling outage. Since 1970, Monticello has reworked 39 main steam line isolation valves at an average rebuild cost of roughly $40,000 per valve. If the proposed 46 scfh combined maximum flow path leakage criterion had been applied, 9 of these repairs could have been avoided. It is reasonable to project that similar benefits (i.e., avoidance of roughly 1 valve repair per refueling cycle) can be expected throughout the remainder of the plant operating life.

It should be noted that, to date, main steam line isolation valve rework has not been critical path for Monticello outages, however, there have been occasions when this nearly occurred. As Monticello's outages become shorter, critical path repair of MSIV's is likely to occur. Section 4.2 of NEDC-31858P Revision 2 notes a case at another utility where 56 days of critical path time were required to correct main steam line isolation valve seat damage caused by lapping being performed to meet the low leakage limit. Such an occurrence could increase the ultimate cost of repairing even a single valve to several million dollars.

2. Repair of main steam line isolation valves involves working in radiation areas, and the performance of unnecessary work is not consistent with ALARA principles. Although dose rates around the main steam line isolation valves vary from outage to outage, it is estimated that rework of a single valve represents approximately 1 man-Rem of exposure. Actual dose will vary based on current dose rates and the location of the valve being repaired. For example, dose rates near the inboard valve (located inside the drywell) are higher than those near the outboard valves (located outside the drywell). As discussed above, application of the proposed criterion would avoid roughly 1 valve repair per refueling cycle and thus represents a potential savings of approximately 1 man-Rem per refueling cycle.

<u>Proposed Change (Part 2) - Drywell spray header and nozzle air test frequency:</u>

Technical Specification 4.5.C (page 104) presently requires an operability test of the drywell spray header and nozzles with an air test during each 5 year period. It is proposed that the operability test interval be revised from a 5 year interval, to a 10 year interval.

<u>Reason for Changes:</u>

The purpose of the operability test is to demonstrate the spray header and nozzles are unobstructed. Industry experience has shown the headers and nozzles to be reliable, as stated by Bases section SR 3.6.1.7.4 of the BWR/6 Improved Standard Technical Specification (Rev. 0, dated 09/28/92), which reads:

> "This Surveillance is performed every 10 years to verify that the spray nozzles are not obstructed and that flow will be provided when required. The 10 year frequency is adequate to detect degradation in performance due to the passive nozzle design and

its normally dry state and has been shown to be acceptable through operating experience."

Operability testing experience at Monticello has likewise been successful with no observed nozzle blockage during either periodic testing, or specific inspections. Additionally, since the test will only be run half as often, this change will result in an associated 50% cost reduction, and a 50 % reduction in exposure.

Safety Evaluation:

Due to the passive nature of the drywell spray header and nozzles, and its normally dry state, there are very few malfunctions that could occur to this equipment. Industry experience has shown them to be reliable and free from plugging. The large number of nozzles also provide redundancy should one become plugged.

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.

Proposed Changes (Part 3)- Utilization of Appendix J test interval criteria for Primary Containment:

In various locations of Monticello Technical Specification sections 3/4.7.a.2 (Primary Containment Integrity), information specific to the primary containment leakage rate testing program is being removed and replaced with statements to abide by the requirements of Appendix J of 10 CFR Part 50.

Reason for Changes:

Removing the details of how primary containment testing is performed at Monticello and replacing this information with more generalized statements that testing will be per Appendix J of 10 CFR Part 50 is in compliance with, and following the desires of the NRC to migrate all plants with Custom Technical Specifications towards NUREG-1433 (Standard Technical Specifications (STS) for General Electric Plants, BWR/4). Monticello has not yet determined if it will fully change over to the STS, but where possible, we will comply with the STS when making license amendment requests such as this.

Because of the changes made to section 4.7.A.2.b (discussed below), section 4.7.A.2.a (page 158) which makes reference to Surveillance Requirement 4.7.A.2.b.5 is too prescriptive. To still control the location of the requirement, yet remain accurate, it is proposed the reference be broadened to 4.7.A.2.b by dropping the ".5".

In sections 3.7.A.2.b.2 and 3, reference to all penetrations and valves except for main steam isolation valves is being revised to delete the exceptions to the MSIV's. This is because Monticello has already been including the MSIV leakage into the combined leakage rate of less than or equal to 0.6La. Also, there is not a specific exemption allowing exclusion of MSIV

leakage from the combined Type B and C leakage limit of 0.6La. Deleting this conflicting statement makes the Technical Specifications consistent with Appendix J.

Also in these same sections, several clarifications are being made. In 3.7.A.2.b.2, it is being explained that Pa is 42 psig. In 3.7.A.2.b.3, "g" is being added to 25 psi making it 25 psig. These are minor revisions that will improve statement clarity.

In section 4.7.A.2.b. and c, much of the existing wording is being deleted as the prescriptiveness of the existing wording limits primary containment and MSIV testing program flexibility without adding safety value. Included in this change is removal of the specific requirement to test the MSIV's at 18 month intervals. In its place will be reference to the fact that leak rate testing will be in accordance with 10 CFR Part 50, Appendix J, as modified by approved exemptions. The last statement, "as modified by approved exemptions" clarifies that there are exemptions on leak rate testing at Monticello that have been approved by the NRC. This statement, now standard in the STS, was missing from the Monticello Technical Specifications.

Also being deleted is the statement on the bottom of page 159 that: "The second test of the second 10-year service period may be conducted during the 1989 refueling outage". Since this outage has long since passed, this statement is no longer controlling and should be removed.

On page 161, sections 4.7.A.2.c.1 and 2 which pertain to the containment airlocks, are being revised. This reflects the referencing of Appendix J and the low pressure testing of the primary containment airlock exemption and reduced airlock testing frequency exemption already issued as stated in Basis section 4.7.A on page 184 as exemptions c and d respectively.

Safety Evaluation:

When Appendix J to 10 CFR 50 was written into the regulations, the nuclear industry had limited experience with containment performance and testing. The retesting schedule requirements of Appendix J were based on the best judgment at that time for providing assurance that containment integrity would be maintained. Since then, numerous nuclear plants have been built and operated for many years, and many containment tests have been performed. This experience provides the basis for understanding how containment integrity will be assured from test to test and allow adjusting test schedules to meet specific needs.

This request for license amendment does not in and of itself change any test schedules. It is only an administrative change that eliminates the prescriptive requirements on performing leak rate tests contained in Appendix J, and would therefore allow schedules to be changed when the Nuclear Regulatory Commission grants an exemption. Each request for specific exemption would require evaluation on its own merits to determine its safety impact. Therefore, this amendment does not introduce any safety concerns.

Determination of Significant Hazards Consideration (all Parts):

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

The proposed amendment is limited to changes to the surveillance testing requirements applicable to the main steam line isolation valves allowable leakage criteria, drywell nozzles test interval, and method of applying Appendix J test requirements. With respect to monitoring main steam isolation valve performance, the proposed criteria are equivalent to the current criteria ensuring that leakage past the valves would be within acceptable limits under accident conditions. These surveillance tests are performed while the plant is in a cold shutdown condition at a time when the equipment is not required to be operable. Performance of the tests themselves are not 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 equipment, nor will it diminish the capability of the equipment to perform as required during an accident. Combining the maximum per valve leakrate into an overall maximum leakage limit does not increase the overall permissible leakage and thus has no significant impact on the consequences of previously analyzed accidents since the combined leak rate of the main steam line isolation valves, and thus the contribution of the valves to overall primary containment leakage as used for analysis purposes, is unchanged. Extending the drywell nozzle test interval has been shown by industry experience to not compromise safety, and removing the specifics of primary containment leakage testing from the Technical Specifications and referencing 10 CFR Part 50 Appendix J does not alter either how actual testing is accomplished nor the acceptance criteria.

Therefore, there will be no increase in post accident off-site or on-site radiation dose as a result of this amendment. The proposed amendment requires compliance with the regulatory requirements of 10 CFR Part 50, Appendix J 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. <u>The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.</u>

The proposed amendment does not involve any modification to plant equipment or operating procedures, nor will it introduce any new equipment failure modes that have not been previously considered. The proposed amendment is limited to changes in surveillance test acceptance criteria of tests performed while the plant is in cold shutdown at a time when the associated equipment is 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. <u>The proposed amendment will not involve a significant reduction in the margin of safety</u>.

Combining the allowable leak rate for the MSIV's from a per valve limit, to an overall limit, does not change the total allowable leakage and therefore post accident dose levels remain unchanged. Extending the drywell nozzle surveillance test interval from 5 to 10 years has been shown by industry experience to be acceptable. Removing specific containment leakage test requirements from Technical Specifications and replacing it with requirements to abide by 10 CFR Part 50 Appendix J will not change the acceptance criteria nor how testing is accomplished.

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.

<u>Environmental Assessment:</u>

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.