UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT DOCKET NO. 50-282
50-306

REQUEST FOR AMENDMENT TO
OPERATING LICENSES DPR-42 & DPR-60

LICENSE AMENDMENT REQUEST DATED August 30, 1994

Northern States Power Company, a Minnesota corporation, requests authorization for changes to the Prairie Island Operating License and Appendix A as shown on the attachments labeled Exhibits A, B, and C. Exhibit A describes the proposed changes, reasons for the changes, and the supporting safety evaluation/significant hazards determination. Exhibit B contains current Prairie Island Technical Specification pages marked up to show the proposed changes. Exhibit C contains the revised Operating License and Technical Specification pages.

This letter contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By E J Watzl
E. L. Watzl
General Manager
Prairie Island Nuclear Generating Plant

On this 31st day of August 1994 before me a notary public in and for said County, personally appeared E. L. Watzl, General Manager, Prairie Island Nuclear Generating 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.

Marcia K LaCore

MARCIA K. LaCORE
NOTARY PUBLIC-MINNESOTA
HENNEPIN COUNTY
My Commission Expires Jan. 31, 2000

LICENSE AMENDMENT REQUEST DATED August 30, 1994
Line-Item Technical Specification Improvements to Reduce
Surveillance Requirements for Testing During Power Operation

EXHIBIT A

Description of the Proposed Changes, The Reasons for Requesting the Changes, and the Supporting Safety Evaluation/Significant Hazards Determination

Pursuant to 10 CFR Part 50, Sections 50.59 and 50.90, the holders of Operating Licenses DPR-42 and DPR-60 hereby propose the following changes to the Facility Operating Licenses and Appendix A, Technical Specifications:

Background

The NRC issued NUREG-1366, "Improvements to Technical Specifications Surveillance Requirements", in December 1992 to provide the results of a comprehensive examination of surveillance testing required by Technical Specifications. This document found that while some testing at power is essential to verify equipment and system operability, safety can be improved, equipment degradation decreased, and unnecessary personnel burden relaxed by reducing the amount of testing at power.

On September 27, 1993 the NRC issued Generic Letter 93-05,"Line-Item Technical Specification Improvements to Reduce Surveillance Requirements for Testing During Power Operation". Using this Generic Letter, licensees are encouraged to propose Technical Specification changes that are consistent with the guidance provided.

Of the forty-seven line item changes included in Generic Letter 93-05, Northern States Power chooses to request six Technical Specification changes at this time for which the recommendations of NUREG-1366 are compatible with Prairie Island operating experience. These six Technical Specification amendment requests, described further below, are compatible with NUREG-1366 recommendations and are consistent with the guidance of Generic Letter 93-05.

Additional changes are also proposed for the hydrogen recombiner testing which eliminate unnecessary requirements and bring Prairie Island Technical Specifications into conformance with Standard Technical Specifications.

Proposed Changes and Reasons for Changes

The proposed changes to Prairie Island Operating License Appendix A, Technical Specifications are described below, and the specific wording changes are shown in Exhibits B and C.

1. Table TS.4.1-1 (Page 3 of 5), MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS: Delete "Accumulator Level and Pressure" and corresponding frequency interval designations.

   Justification: This change deletes the accumulator water level and pressure channel Checks, Calibrations and Functional Tests from the Technical Specifications and places them into licensee controlled test procedures. These changes are consistent with industry recognition that accumulator instrumentation operability is not directly related to the capability of the accumulators to perform their safety function. This requested Technical Specification amendment implements Generic Letter 93-05 item 7.4 Accumulator Water Level and Pressure Channel Surveillance Requirements (PWR).

2. Table TS.4.1-2A, MINIMUM FREQUENCIES FOR EQUIPMENT TESTS: Revise the frequency for partial movement of all control rod assemblies from every 2 weeks to once per quarter.

   Justification: This change is requested because industry experience has shown that this test has caused reactor trips, dropped rods, and unnecessary challenges to safety systems. The purpose of control rod movements tests is to detect rods that cannot move, however, most stuck rods are discovered during plant startup during initial pulling of the rods or during rod drop testing. This requested Technical Specification amendment implements Generic Letter 93-05 item 4.2 Control Rod Movement Test.

3. Technical Specification 4.3, PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES: Extend the amount of time the plant can be shut down before pressure isolation valve (PIV) testing will be required from 72 hours to 7 days.

   Justification: This change is requested because industry experience has shown that the present surveillance requirement is burdensome and can result in occupational exposure. The current short surveillance test interval places unnecessary restrictions on the recovery from short forced outages where equipment reliability does not show such testing is required. This requested Technical Specification amendment implements Generic Letter 93-05 item 6.1 Reactor Coolant System Isolation Valves (PWR).

4. Technical Specification 4.4, CONTAINMENT SYSTEMS, I. Electric Hydrogen Recombiners: Revise the containment hydrogen recombiner testing surveillance frequency from every six months to every refueling interval. Delete the specific requirement to perform CHANNEL CALIBRATION of recombiner instruments and control circuits. Delete the requirement to sequentially perform the resistance to ground test following the functional test.

   Justification: The change in surveillance frequency is requested because industry experience has shown that the containment hydrogen

recombiners have high reliability and there are redundant recombiners which can perform the required function. Specifically, Prairie Island recombiners have performed reliably. Furthermore, the hydrogen recombiners are manually started many hours after a loss of coolant accident occurs allowing sufficient time to effect repairs if required on the portion of the system located outside containment before the system would be needed. This requested Technical Specification amendment implements Generic Letter 93-05 item 8.5 Hydrogen Recombiner (PWR,BWR).

Deletion of containment hydrogen recombiner CHANNEL CALIBRATION is an administrative change which brings the Technical Specifications into conformance with Standard Technical Specifications. The calibration requirements will be relocated to licensee controlled procedures.

The requirement to perform a resistance to ground test following the functional test does not have a technical basis. The resistance to ground test should be performed before the functional test since another functional test is required after the resistance to ground test. Accordingly this requirement is unnecessarily restrictive. Removal of the requirement for sequential testing will bring this Technical Specification into conformance with the Standard Technical Specifications.

5. Technical Specification 4.5 ENGINEERED SAFETY FEATURES, A. Containment Spray System: Revise the containment spray system nozzle testing surveillance frequency from once every five years to once every ten years.

   Justification: This change is requested because industry experience has shown in general that the system consistently has flow through the header and nozzles. The test does not give quantitative data on flow rates exiting the nozzles, it only verifies that the flow path is unobstructed. Prairie Island's Containment Spray System is constructed of stainless steel and does not have any protective coatings which would clog the spray nozzles. This requested Technical Specification amendment implements Generic Letter 93-05 item 8.1 Containment Spray System (PWR).

6. Technical Specification 4.8, STEAM AND POWER CONVERSION SYSTEMS, A. Auxiliary Feedwater System: Revise the testing frequency for the auxiliary feedwater pumps from intervals of one month to semi-quarterly on a STAGGERED TEST BASIS.

   Justification: This change is requested because the current test frequency exceeds the industry recommended (through ASME Code) once per quarter test frequency. The industry has recognized that at some point an increase in surveillance testing will not contribute to an increase in availability, and in fact could contribute to equipment unavailability. As noted in NUREG-1366, auxiliary feedwater pump availability is increased by testing once per quarter (semi-quarterly on a STAGGERED TEST BASIS). This requested Technical Specification

amendment implements Generic Letter 93-05 item 9.1 Auxiliary Feedwater Pump and system Testing (PWR).

Safety Evaluation

The staff of the U.S. Nuclear Regulatory Commission (NRC) performed a comprehensive examination of Technical Specification surveillance requirements that require testing during power operation as part of the NRC Technical Specification Improvement Program. The results of this work are reported in NUREG-1366,"Improvements to Technical specifications Surveillance Requirements," December 1992.

The staff found that while the majority of the testing at power is important, safety can be improved, equipment degradation decreased, and unnecessary burden on personnel resources eliminated by reducing the amount of testing that the Technical Specifications require at power operating conditions.

The NRC staff used four criteria to screen the surveillance requirements. The criteria are as follows:

1. The surveillance could lead to a plant transient.
2. The surveillance results in unnecessary wear to equipment.
3. The surveillance results in radiation exposure to plant personnel which is not justified by the safety significance of the surveillance.
4. The surveillance places an unnecessary burden on plant personnel because the item required is not justified by the safety significance of the surveillance.

On September 27, 1993 the NRC issued Generic Letter 93-05, "Line Item Improvements to Reduce Surveillance Requirements for Testing during Power Operation". The Generic Letter encourages licensees to present Technical Specification changes consistent with the guidance provided.

As directed by Generic Letter 93-05, since Prairie Island does not have Standard Technical Specifications, the changes proposed by this License Amendment Request are consistent with the intent of NUREG-1366 recommendations, the guidance provided in Generic Letter 93-05, and the format of the Prairie Island Technical Specifications. All proposed Technical Specification changes are compatible with the Prairie Island Nuclear Generating Plant operating experience. The requested amendments do not adversely affect public health and safety. They will, however, reduce equipment degradation, reduce the potential for inadvertent safety system actuations and reduce potential for reactor trips due to testing activities.

Determination of Significant Hazards considerations

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 <u>the probability or consequences of an accident previously evaluated</u>

   Except for hydrogen recombiner changes to conform to Standard Technical Specifications, the requested changes were extensively reviewed by the NRC during the preparation of NUREG-1366 and Generic Letter 93-05. For the sake of clarity each proposed change is discussed separately in the order appearing in the Prairie Island Technical Specifications.

   A. This Technical Specification amendment removes the accumulator water level and pressure channel surveillance from the Technical Specifications and places them into a licensee controlled test procedure. These changes are consistent with industry recognition that accumulator instrumentation operability is not directly related to the capability of the accumulators to perform their safety function.

      Relocating the instrumentation surveillance requirements is an administrative change which will not affect equipment testing, availability, or operation. Therefore, it will not have an effect on the probability or consequences of an accident.

   B. This Technical Specification amendment changes control rod movement from every two weeks to once every quarter. Control rod movement testing is performed to determine if the control rods are immovable. Control rods may be electrically stuck due to a problem in the control rod drive circuitry or mechanically stuck. Electrical problems with the control rod drive system, in general, do not prevent insertion of a control rod into the core when the reactor trip breakers are opened.

      NUREG-1366 determined that control rod movement testing is not effective in determining immovable control rods. Most of the mechanically immovable control rods are discovered during plant startup during initial pulling of the rods or during rod drop testing. Extending the surveillance interval will not affect this failure discovery method.

      The accident analyses assume that the single highest worth rod is stuck while fully withdrawn and will not insert. One immovable control rod will still bound this accident analysis. For these reasons, the extension of the surveillance frequency from once every two weeks to once every quarter will not involve a significant increase in the probability or consequences of a previously evaluated accident.

   C. This Technical Specification amendment will require Reactor Coolant System Pressure Isolation Valves(PIV) to be surveillance

tested after seven days at cold shutdown instead of after three days at cold shutdown.

The PIVs are important in preventing over pressurization and rupture of the Emergency Core Cooling System low pressure piping which could result in a LOCA that bypasses containment. Allowable leakage from any PIV is sufficiently low to ensure early detection of possible in-series check valve failure. This change will not change the refueling outage surveillance, nor will it change the required testing to be performed after maintenance, repair, or replacement. The proposed level of surveillance is appropriate for these valves.

These valves have had very good operating performance and should continue to have the same performance record with continuation of the same maintenance and testing program. Furthermore, these valves are backed by motor or air-operated valves which have performed reliably.

For these reasons, the extension of the amount of time from three days to seven days before pressure isolation valve testing is required will not result in a significant increase in the probability or consequences of a previously evaluated accident.

D. This Technical Specification amendment will revise the containment hydrogen recombiner testing surveillance from every six months to every refueling interval.

The two independent containment hydrogen recombiners provide post-accident hydrogen control of the containment atmosphere. The recombiners are designed to be passive until an accident occurs.

Industry experience and in particular, Prairie Island experience has demonstrated that this equipment is highly reliable. Since the recombiners are not required until after an accident, there would likely be time to effect accessible repairs if the equipment were not operable.

Relocation of the recombiner calibration is an administrative change which will not affect recombiner operability. Deletion of specific testing sequence will not affect the performance of recombiner testing.

Equipment redundancy, reliability and time for repairs ensures post-accident control. For these reasons, these changes will not result in a significant increase in the probability or consequences of a previously evaluated accident.

E. This Technical Specification amendment will revise the containment spray system nozzle testing surveillance from once every five

years to once every ten years.

Two independent containment spray systems provide post-accident cooling of the containment atmosphere and provide a mechanism for removing iodine from the containment atmosphere. This surveillance test verifies by air flow test that the spray nozzles are unobstructed. The extension of the surveillance frequency does not affect administrative controls that preclude entry of foreign material into the nozzles.

At Prairie Island the piping headers and nozzles are fabricated from austenitic stainless steel. There have been no reported in-service problems noted with spray nozzle testing from plants with stainless steel headers and nozzles and there is no indication that the lines would corrode and become obstructed.

For these reasons, this change will not result in a significant increase in the probability or consequences of a previously evaluated accident.

F. This Technical Specification amendment will revise the frequency for testing the Auxiliary Feedwater Pumps (AFWP) from monthly to semi-quarterly on a STAGGERED TEST BASIS.

Two 100% redundant, diverse pumps provide an emergency source of feedwater to the steam generators. The Prairie Island AFWPs have performed reliably. However, frequent testing of the pumps and associated equipment wears out the equipment resulting in equipment unavailability. AFWP availability will be increased by semi-quarterly surveillance testing on a STAGGERED TEST BASIS.

For these reasons, this change will not result in a significant increase in the probability or consequences of a previously evaluated accident.

Therefore, the probability or consequences of an accident previously evaluated are not affected by any of the proposed amendments.

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

The extension of facility surveillance intervals as discussed previously will not result in changes in plant configuration or operation. The changes in recombiner calibration and testing will not result in changes in plant configuration or operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated would not be created.

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

The amendments proposed in this License Amendment Request do not reduce the ability of any system or component to perform its safety related function. The basis of NUREG-1366, Generic Letter 93-05, and the analysis performed in support of this License Amendment Request is that the reduction in surveillance testing can improve safety by reducing challenges to plant systems, personnel exposure, and equipment wear or degradation. The proposed changes to surveillance frequencies do not change the method of performing any surveillance. The operation of systems and equipment remains unchanged. Therefore, a significant reduction in the margin of safety would not be involved.

Based on the evaluation described above, and pursuant to 10 CFR Part 50, Section 50.91, Northern States Power Company has determined that operation the Prairie Island 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 Company has evaluated the proposed changes and determined that:

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

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.