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 JULY 13, 1992

Northern States Power Company, a Minnesota corporation, requests authorization for changes to Appendix A of the Prairie Island Operating License as shown on the attachments labeled Exhibits A, B, and C. Exhibit A describes the proposed changes, reasons for the changes, and a significant hazards evaluation. Exhibits B and C are copies of the Prairie Island Technical Specifications incorporating the proposed changes.

This letter contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By Thomas M Parker
Thomas M Parker
Manager
Nuclear Support Services

On this 13th day of July 1992 before me a notary public in and for said County, personally appeared Thomas M Parker, Manager Nuclear Support Services, 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

Prairie Island Nuclear Generating Plant
License Amendment Request Dated July 13, 1992

Evaluation of Proposed Changes to the
Technical Specifications Appendix A of
Operating License DPR-42 and DPR-60

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 Appendix A, Technical Specifications:

1. Refueling Containment Integrity Requirements

Background

This license amendment request proposes clarification of the applicability of the Prairie Island Technical Specifications requirements for containment closure during refueling. Specification 3.8.A.1.a of the Prairie Island Technical Specifications specifies that at least one isolation valve shall be operable or locked closed in each line which penetrates the containment and provides a direct path from containment atmosphere to the outside.

There is no guidance provided in the limiting condition for operation or the bases for Specification 3.8.A.1.a as to the meaning of "outside". Current interpretation of this specification is to apply the requirements to all lines which penetrate the containment. Because of the protection provided outside the majority of the containment penetrations by the Auxiliary Building Special Ventilation System and the Shield Building Ventilation System, we believe this interpretation is overly restrictive. The proposed changes described below clarify the requirements of Specification 3.8.A.1.a such that the containment closure requirements during refueling are only applied to lines which do not exit the containment into areas protected by operable automatic safeguards ventilation systems which actuate on high radiation and filter all releases to the environment through particulate-absolute-charcoal filters.

The current wording of Specification 3.8.A.1.a is also confusing with respect to the requirements for automatic isolation valve operability and closure. Therefore, in addition to the changes described above, the wording of Specification 3.8.A.1.a has been revised to be consistent with the guidance provided in the Standard Technical Specifications.

Proposed Changes and Reasons for Change

The proposed changes to the Prairie Island Technical Specifications being submitted to clarify the requirements for containment closure during refueling are described below, and the specific wording changes to Technical Specifications are shown in Exhibits B and C.

A. Proposed changes to Technical Specification 3.8.A.1.a

Current Specification 3.8.A.1.a is being replaced as shown in Exhibit B. The proposed new Specifications 3.8.A.1.a.1, 2 and 3 are consistent with the guidance provided in the Standard Technical Specifications for containment closure during core alterations. Proposed Specifications 3.8.A.1.a.4 and 5 clarify the closure requirements for penetrations which do not exit the containment into areas protected by operable automatic safeguards ven 'lation systems. The proposed wording more clearly defines the requirements for automatic isolation valve operability and closure.

Proposed Specifications 3.8.A.1.a.1 and 2 are consistent with the requirements of the current Specification 3.8.A.1.a except that the current specification does not require the equipment hatch to be held in place by a minimum of four bolts.

The current wording of Specification 3.8.A.1.a is interpreted to require that an isolation valve required for closure of the containment during core alterations is operable if it is capable of being closed by a manual containment isolation signal from the control room. This interpretation leads to confusion with respect to what valves must be in the closed position. The proposed Specifications provides improved guidance with respect to which valves are required to be closed. The requirements of the proposed Specification 3.8.A.1.a.3 are more restrictive than the current re... rements with respect to which automatic isolation valves are required to be closed.

The incorporation of the Standard Technical Specification wording into Specification 3.8.A.1.a results in "outside" being changed to "outside atmosphere" in proposed Specification 3.8.A.1.a.3. This change, in combination with the proposed Specifications 3.8.A.1.a.4 and 5, and the bases changes described below, will clarify that the requirement of Specification 3.8.A.1.a.3 only apply to lines which penetrate the containment into areas where leakage through a penetration could be released unfiltered to the environment.

B. Proposed changes to the Bases for Technical Specification Section 3.8

The Bases to Technical Specification Section 3.8 are being revised, as shown in Exhibit B, to provide the bases for the requirements for containment closure during refueling and to clarify that those requirements are only applied to lines which do not exit the containment into areas protected by operable automatic safeguards ventilation systems whi h actuate on high radiation and filter all releases to the environment through particulate-absolute-charcoal filters.

Safety Evaluation

The containment serves to contain fission-product radioactivity that may be released during an accident such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. The intent of the requirements for containment penetration closure during refueling is to ensure that a release of fission-product radioactivity within the containment as a result of a fuel-handling accident will be restricted from escaping directly to the environment.

The proposed wording of Specifications 3.8.A.1.a.1, 2 and 3 is consistent with the guidance provided in the Standard Technical Specifications.

The proposed Specifications 3.8.A.1.a.4 and 5 and the changes to the bases for Specification 3.8.A.1.a are based on the premise that containment closure requirements are unnecessary for lines which exit into areas serviced by the Auxiliary Building Special Ventilation System and the Shield Building Ventilation System. The Auxiliary Building Special Ventilation System is designed to reliably collect any potential containment leakage into the Auxiliary Building Special Ventilation Zone and pass that leakage through absolute particulate and charcoal filters before it reaches the environment. The Shield Building Ventilation System is designed to collect leakage from the containment into the annulus between the containment vessel and the shield building, and discharge that leakage to the environment through particulate-absolute-charcoal filters. Therefore, any leakage from the containment into the Auxiliary Building Special Ventilation Zone or Shield Building Annulus will be filtered prior to release to the environment, greatly reducing the potential off-site exposure.

The analysis of a fuel handling accident inside containment, described in Section 14.5.1.4 of the Prairie Island Updated Safety Analysis Report, takes no credit for containment isolation or filtration of releases. The entire release from the complete rupture of a single fuel assembly is assumed to be taken into the high capacity purge system and released to the environment untreated. No credit is taken for the automatic isolation of the purge system. The resulting whole body and thyroid doses are well below the requirements of 10 CFR Part 100.

If a fuel handling accident were to occur under normal conditions, the containment purge systems would isolate on high radiation and the majority of the fission product release would remain in the containment. Because there is no containment pressure increase associated with a fuel handling accident, leakage out through any open containment penetrations to the Auxiliary Building Special Ventilation Zone or the Shield Building Annulus would be small and would take place over a long period of time. There would be ample time to take actions to close any open penetrations. The majority of the fission product release would be retained in the containment and would be diluted within the containment.

Any releases that reached the Auxiliary Building Special Ventilation Zone or the Shield Building Annulus would be further diluted and would ultimately actuate the Auxiliary Building Special Ventilation System and the Shield Building Ventilation System. These safeguards ventilation systems would draw the releases through particulate-absolute-charcoal filters prior to release to the environment. The combination of dilution, time delay and filtration would maintain offsite doses from releases to the Auxiliary Building Special Ventilation Zone or Shield Building Annulus to a small fraction of the design basis fuel accident described in the Updated Safety Analysis Report and therefore significantly below the 10 CFR Part 100 requirements.

In conclusion, Northern States Power believes there is reasonable assurance that the health and safety of the public will not be adversely affected by the proposed Technical Specification changes.

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 the probability or consequences of an accident previously evaluated.

   The requirements of Technical Specification 3.8.A.1.a are only intended to mitigate the consequences of an accident. Therefore, the proposed changes, which clarify the requirements for operability and closure of isolation valves during core alterations, have no impact on the probability of an accident.

   Technical Specification 3.8.A.1.a is intended to mitigate the consequences of a fuel handling accident in the containment. The analysis of a fuel handling accident in containment, described in the Prairie Island Updated Safety Analysis Report, assumes the fission product release is transmitted untreated to the environment via a high capacity purge system. The proposed clarification of the requirements of Specification 3.8.A.1.a would allow fission product releases to the Auxiliary Building Special Ventilation Zone or Shield Building Annulus not allowed by the present interpretation of the Specification. However, provided the associated safeguards ventilation systems are operable, any releases to the Auxiliary Building Special Ventilation Zone or Shield Building Annulus would be filtered prior to release to the environment and would result in offsite exposures that would be a small fraction of those resulting from the design basis fuel handling accident described in the Updated Safety Analysis Report.

   Therefore, for the reasons discussed above, 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 modification of plant equipment or any changes in operational limits. The proposed changes only modify and clarify a specification designed to mitigate the consequences of a fuel handling accident in the containment. The proposed clarification may affect the release path for fission products released during a fuel handling accident in containment, but no new or different kind of accident will result.

   Therefore, for the reasons discussed above, 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. <u>The proposed amendment will not involve a significant reduction in the margin of safety.</u>

   Technical Specification 3.8.A.1.a is intended to mitigate the consequences of a fuel handling accident in the containment. The analysis of a fuel handling accident in containment, described in the Prairie Island Updated Safety Analysis Report, assumes the fission product release is transmitted untreated to the environment via a high capacity purge system. The proposed clarification of the requirements of Specification 3.8.A.1.a would allow fission product releases to the Auxiliary Building Special Ventilation Zone or Shield Building Annulus not allowed by the present interpretation of the Specification.

   However, provided the associated safeguards ventilation systems are operable, any releases to the Auxiliary Building Special Ventilation Zone or the Shield Building Annulus would be filtered prior to release to the environment and would result in offsite exposures that would be a small fraction of those resulting from the design basis fuel handling accident described in the Updated Safety Analysis Report. The margin of safety for the fuel handling accident in containment, as described in the Updated Safety Analysis Report, would be unaffected.

   Therefore, for the reasons discussed above, the proposed changes will not result in any reduction in the plant's 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 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.

<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.

2. Fuel Handling Crane Specifications

Background

This license amendment request proposes changes to Prairie Island Technical Specification Sections 3.8 and 4.1 which will clarify the requirements for operability and load testing of the fuel handling cranes. These changes were initiated in response to comments from the NRC Resident Inspector for Prairie Island on the October 4, 1991 License Amendment Request which requested changes related to the upgrade of the Auxiliary Building Crane.

Significant changes were made to Section 3.8 of the Prairie Island Technical Specifications by License Amendment Nos. 91 and 84, dated October 27, 1989. Those changes included the relocation of the fuel handling crane load test requirements from old Section 3.8.A, which defined refueling operations requirements, to the current Section 3.8.B, which defines fuel handling requirements. The location of the fuel crane load test requirements in Section 3.8.B can lead to confusion with respect to how the requirements apply to the containment manipulator crane during core alterations. The proposed changes to Section 3.8 will provide specific operability requirements for both the spent fuel pool fuel handling crane and the manipulator crane.

Additionally, the fuel handling crane load test requirements have relocated to Table TS.4.1-2A. Specific load test surveillance requirements, with time limitations for the validity of the testing, are specified for the spent fuel pool fuel handling crane and for the manipulator crane.

Proposed Changes and Reasons for Change

The proposed changes to the Prairie Island Technical Specifications being submitted to clarify the requirements for operability and load testing of the fuel handling cranes are described below, and the specific wording changes to Technical Specifications are shown in Exhibits B and C.

A. Proposed changes to Technical Specification 3.8.A.1

New Specification 3.8.A.1.k is being incorporated into Section 3.8.A.1 to clearly specify that the manipulator crane shall be operable for the movement of fuel assemblies during core alterations.

B. Proposed changes to Technical Specification 3.8.B.1

The phrase "in the spent fuel pool area" is being incorporated into Specification 3.8.B.1 to clarify that the specified requirements only apply during fuel handling in the spent fuel pool area.

C. Proposed changes to Technical Specification 3.8.B.1.b

The current load test requirements are being deleted from Specification 3.8.B.1.b. They are being relocated to Section 4.1 as described below. The load test requirements in Specification 3.8.B.1.b are being replaced with a Specification that clearly specifies that the spent fuel pool fuel handling crane shall be operable for the movement of fuel assemblies.

D. Relocation of Fuel Handling Crane Load Test Requirements

The fuel handling crane load test requirements for verification of the operability of limit switches, interlocks and alarms are being relocated from Specification 3.8.B.1.b to Table TS.4.1-2A as shown in Exhibit B. This relocation will separate the fuel handling crane load test surveillance requirements from the limiting conditions for operation. Specific load test surveillance requirements, with time limitations for the validity of the testing, are specified for the spent fuel pool fuel handling crane and for the manipulator crane in Table TS.4.1-2A.

The time limitations for the manipulator crane specifies that it be load tested to verify operability of limit switches, interlocks and alarms once each refueling outage prior to the start of core alterations. Testing of the manipulator crane prior to the start of core alterations will ensure that any equipment degradation that has occurred since the previous refueling outage will be identified before the crane is used for handling fuel. The proposed specification that the testing be performed once during each refueling outage, will avoid unnecessary testing of the manipulator crane if core alterations are interrupted during a refueling outage. If core alterations are interrupted due to inoperability of the manipulator crane, post maintenance testing would be performed on the crane as necessary to ensure crane operability.

There are no requirements in the Standard Technical Specifications for the operability or testing of the spent fuel pool fuel handling crane. However, a requirement for load testing to verify operability of limit switches, interlocks and alarms is proposed to help ensure the safe handling of fuel in the spent fuel pool. The 3 month time limitation is included to clarify when testing is required and to avoid unnecessary testing of the crane if its use is not continuous.

E. Editorial Changes to Table TS.4.1-2A

Several editorial changes to Table TS.4.1-2A are included in the proposed changes. A second page was added to the table to accommodate the relocation of the crane test requirements.

The reference to specific FSAR sections was eliminated. Those references were outdated and provided little useful information that couldn't be found in the Table of Contents of the Updated Safety Analysis Report.

The following editorial changes noted in the proposed revision to Table TS.4.1-2A have been previously submitted as part the License Amendment Request dated June 25, 1991 which is still under review. These changes are included in this amendment request for consistency with the earlier amendment request:

a. The word "floodes" is being corrected to "flooded" in Item 5 of the table.

b. The word "Quaterly" is being corrected to "Quarterly" in Item 6 of the table.

c. Item 12, which is just a reference to a previously deleted item is deleted from the first page of the table to provide additional room for the first page and to allow consistent numbering for the items on the proposed second page.

Safety Evaluation

The addition of specific operability requirements for the fuel handling cranes into the Prairie Island Technical Specifications is just a clarification of an operability requirement implied by the current load test requirements in Section 3.8.B. Specifically stating that the fuel manipulator crane and the spent fuel pool fuel handling crane shall be operable for the handling of fuel is an enhancement to plant safety.

The proposed addition of the limitation associated with the testing of the manipulator crane prior to the start of core alterations will enhance plant safety by ensuring that any equipment degradation that has occurred since the previous refueling outage will be identified before the crane is used for handling fuel.

The proposed manipulator crane testing requirements specify that the testing need only be performed once per refueling outage. The small risk associated with not repeating the manipulator crane testing, following a delay in core alterations, is offset by the increased risk associated with excessive testing of the manipulator crane components. The purpose of testing of the manipulator crane prior to use is to verify the operability of limit switches, interlocks and alarms after an extended shutdown in the adverse conditions of the containment with the reactor at full power. There is no reason to believe that an interruption in core alterations during a refueling outage, where containment conditions are significantly milder, will have any significant adverse effect on the manipulator crane components. Where core alterations are interrupted due to inoperability of the manipulator crane, post maintenance testing would be performed on the crane as necessary to ensure crane operability.

There are no requirements in the Standard Technical Specifications for the operability or testing of the spent fuel pool fuel handling crane. However, a requirement for load testing to verify operability of limit switches, interlocks and alarms associated with the spent fuel pool fuel handling crane will help ensure the safe handling of fuel in the spent fuel pool. The 3 month time limitation is included to clarify when testing is required and to avoid unnecessary testing of the crane if its use is not continuous. The 3 month time limitation proposed by this request will provide a testing frequency that ensures a reasonable balance between the risk associated with inadequate testing and risk associated with degradation of equipment due to excessive testing.

In conclusion, Northern States Power believes there is reasonable assurance that the health and safety of the public will not be adversely affected by the proposed Technical Specification changes.

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 the probability or consequences of an accident previously evaluated.

   The proposed changes to the Prairie Island Technical Specifications are associated with the testing and operability of the cranes used for handling fuel. The incorporation of operability requirements for the subject cranes will help ensure that the cranes are capable of safely handling fuel and will reduce the risk of a fuel handling accident. As discussed above the proposed time limitations for the crane testing will balance the risk of operating the crane with an inoperable component with the risks associated with excessive testing and the probability of a fuel handling accident should not be significantly affected.

   The analysis of a fuel handling accident inside containment, described in Section 14.5.1.4 of the Prairie Island Updated Safety Analysis Report, assumes the rupture of an entire fuel assembly. The resulting whole body and thyroid doses are well below the requirements of 10 CFR Part 100. Even if the proposed changes were to result in the failure of a fuel handling crane and the dropping of a fuel assembly, the conservative fuel handling accident analysis will remain bounding. There will be no effect on the consequences of the fuel handling accident evaluated in the Updated Safety Analysis Report.

   Therefore, 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 modification in operational limits.

   The analysis of a fuel handling accident inside containment, described in Section 14.5.1.4 of the Prairie Island Updated Safety Analysis Report, assumes the rupture of an entire fuel assembly. Even if the proposed changes result in the failure of a fuel handling crane and the dropping of a fuel assembly, the conservative fuel handling accident analysis in the Updated Safety Analysis Report will remain bounding. No fuel handling accident not already addressed by the Updated Safety Analysis Report will result.

Therefore, 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 to the Prairie Island Technical Specifications are associated with the testing and operability of the cranes used for handling fuel. The incorporation of operability requirements for the subject cranes will help ensure that the cranes capable of safely handling fuel and will reduce the risk of a fuel handling accident, thus increasing the plants margin of safety.

As discussed above the proposed time limitations for the crane testing will balance the risk of operating the crane with an inoperable component with the risks associated with excessive testing and the probability of a fuel handling accident, and thus the margin of safety, should not be significantly affected.

The analysis of a fuel handling accident inside containment, described in Section 14.5.1.4 of the Prairie Island Updated Safety Analysis Report, assumes the rupture of an entire fuel assembly. The resulting whole body and thyroid doses are well below the requirements of 10 CFR Part 100. Even if the proposed changes were to result in the failure of a fuel handling crane and the dropping of a fuel assembly, the conservative fuel handling accident analysis will remain bounding. There will be no effect on the consequences of the fuel handling accident evaluated in the Updated Safety Analysis Report.

Therefore, the proposed changes will not result in any reduction in the plant's 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 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 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.