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Northern States Power Company

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March 13, 1991

10 CFR Part 50
Section 50.90

U S Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

License Amendment Request Dated March 13, 1991
Fuel Assembly Design Features Changes

Attached is a request for a change to the Technical Specifications, Appendix A of the Operating Licenses, for the Prairie Island Nuclear Generating Plant. This request is submitted in accordance with the provisions of 10 CFR Part 50, Section 50.90.

The Prairie Island Technical Specifications include design requirements for fuel assemblies in Section 5, Design Features. On a plant-specific basis, the NRC Staff has approved changes to these requirements that provide flexibility for improved fuel performance by permitting timely removal of fuel rods that are found to be leaking during a refueling outage or are determined to be probable sources of future leakage.

Because improvements in fuel performance will result in lower occupational radiation exposure and plant radiological releases, this alternative was made available to all plants as a line-item Technical Specification improvement by Generic Letter 90-02, "Alternative Requirements for Fuel Assemblies in the Design Features Section of Technical Specifications". The proposed changes to Technical Specification 5.3.A.1 are being submitted in response to Generic Letter 90-02.

Exhibit A contains a description of the proposed changes, the reasons for requesting 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 Technical Specification pages.

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Please contact us if you have any questions related to this License Amendment Request.

Thomas M Parker
Manager
Nuclear Support Services

c: Regional Administrator-III, NRC
NRR Project Manager, NRC
Senior Resident Inspector, NRC
MPCA
Attn: J W Ferman
J E Silberg

Attachments:

Affidavit

Exhibit A - Evaluation of Proposed Changes to the Technical Specifications

Exhibit B - Proposed Changes Marked Up on Existing Technical Specification Pages

Exhibit C - Revised Technical Specification Pages

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 March 13, 1991

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 13 day of March 1991 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, 1991

Exhibit A

Prairie Island Nuclear Generating Plant
License Amendment Request Dated March 13, 1991

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:

Proposed Changes

Revise specification 5.3.A.1, as shown on page TS.5.3-1 in Exhibit B, to allow Zircaloy-4 filler rods, stainless steel filler rods or open water channels to be substituted for fuel rods in fuel assemblies.

Reason For Changes

The Prairie Island Technical Specifications include design requirements for fuel assemblies in Section 5, Design Features. On a plant-specific basis, the NRC Staff has approved changes to these requirements that provide flexibility for improved fuel performance by permitting timely removal of fuel rods that are found to be leaking during a refueling outage or are determined to be probable sources of future leakage. Because improvements in fuel performance will result in lower occupational radiation exposure and plant radiological releases, this alternative was made available to all plants as a line-item Technical Specification improvement by Generic Letter 90-02.

Generic Letter 90-02 provided specific guidance for the modification of the fuel assembly design features specification. Technical Specification 5.3.A.1 has been modified in accordance with this guidance.

The proposed changes will allow Zircaloy-4 filler rods, stainless steel filler rods or open water channels to be substituted for fuel rods in fuel assemblies as long as that replacement is justified by cycle-specific reload analyses performed using NRC-approved methodology. Prior to each fuel cycle an analysis is performed to ensure that, with each reload of fuel, all core design safety criteria are met. This analysis is performed using the NRC approved methodology listed in Technical Specification 6.7.A.6.b.

In the case where fuel assemblies are repaired or fuel rods replaced, appropriate safety analysis will be conducted in conjunction with the normal reload analysis, verifying that all applicable core safety limits for fuel rods in the vicinity of the missing or substituted rods are still met. By modeling based on the exact substitution, an accurate and complete safety analysis can be performed, and conformance with established safety margins will be ensured. These analyses and the core reload changes are reviewed by Northern States Power as required by 10 CFR Part 50, Section 50.59. If a change to Technical Specifications or an unreviewed safety question is identified, then appropriate changes and analyses will be provided to the NRC for review and approval.

The qualifier "using an NRC-approved methodology" is included in the proposed specification to ensure that the effects of fuel rod removal/replacement will be analyzed or evaluated on a cycle specific basis using the same NRC-approved methodology and design limits that apply to any reload core.

The requirement to report fuel rod replacement for more than 30 rods in the core or 10 rods in any fuel assembly per refueling is included to ensure the NRC is advised of abnormal fuel performance. The reporting threshold criteria is consistent with the guidance provided in Generic Letter 90-02. The proposed changes would require fuel rod replacement in excess of these numbers to be reported within 30 days after cycle startup.

Safety Evaluation and 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 modifications to the Design Features section of the Prairie Island Technical Specifications require that operation with fuel assemblies that have been repaired or have had fuel rods replaced, be justified by cycle-specific reload analyses using an NRC-approved methodology. This justification will be provided by the performance of an appropriate safety analysis in conjunction with the normal reload analysis, verifying that all applicable core safety limits for fuel rods in the vicinity of the missing or substituted rods are still met. By modeling based on the exact substitution, an accurate and complete safety analysis will be performed, and conformance with established safety margins will be assured. These analyses and the core reload changes will be reviewed as required by 10 CFR Part 50, Section 50.59.

   The evaluations and analyses described above will provide adequate assurance that the repair of a fuel assembly by the replacement or removal of one or more fuel rods 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. While fuel assemblies containing a substitute rod or vacancies represent a change in the physical core configuration, it is a physical change which is no more significant than, for example, using fuel of a different enrichment from a previous cycle. All such changes will be accounted for by the reload analysis as described above. Given successful completion of such an analysis, it is not possible to create a new or different kind of accident 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.

   As stated above, the proposed changes require that operation with fuel assemblies that have been repaired or have had fuel rods replaced, be justified by cycle-specific reload analyses using an NRC approved methodology. If the physical parameters of the reload core are evaluated as being within previously defined acceptance criteria, then a reduction in the margin of safety is precluded. Therefore, the proposed changes will not result in any reduction in the plant's margin of safety.

The Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 for determining whether a significant hazards consideration exists by providing certain examples of amendments that will likely be found to involve no significant hazards considerations. These examples were published in the Federal Register on March 6, 1986.

The changes to the Prairie Island Technical Specifications proposed above are equivalent to NRC example (vi), because they involve changes which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan. Based on this guidance and the reasons discussed above, we have concluded that the proposed changes do not involve a significant hazards consideration.

Environmental Assessment

This license amendment request does not change effluent types or total effluent amounts nor does it involve an increase in power level. Therefore, this change will not result in any significant environmental impact.

Exhibit B

Prairie Island Nuclear Generating Plant

License Amendment Request Dated March 13, 1991

Proposed Changes Marked Up
On Existing Technical Specification Pages

Exhibit B consists of an existing Technical Specification page with the proposed changes written on that page. The existing page affected by this License Amendment Request is listed below:

TS.5.3-1