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 LICENSE NOS. DPR-42 & DPR-60

(License Amendment Request Dated July 11, 1984)

Northern States Power Company, a Minnesota corporation, requests authorization for changes to the Technical Specifications as shown on the attachments labeled Exhibit A and Exhibit B. Exhibit A describes the proposed changes along with reasons for the change. Exhibit B is a set of Technical Specification pages incorporating the proposed changes. Exhibit C is a report NSPNAD-8406. Exhibit D is a report XN-NF-84-03.

This letter contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By David Musolf
David Musolf
Manager - Nuclear Support Services

On this 11th day of July, 1984 before me a notary public in and for said County, personally appeared David Musolf, 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.

Dody A. Brose

DODY A. BROSE
NOTARY PUBLIC – MINNESOTA
HENNEPIN COUNTY
My Commission Expires Dec. 26, 1989

EXHIBIT A

Prairie Island Nuclear Generating Plant

License Amendment Request - Dated July 11, 1984

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

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

1. $F^N_{\Delta H}/F_Q$ Limit Changes

   Proposed Changes (TS-x; TS.2.1-2; TS.2.1-3; Figure TS.2.1-1; TS.3.10-1, 2, 9, 10, 11 and 13; Figure TS.3.10-7.)

   The changes discussed here are associated with new peaking factor limit functions. Rather than have one F_Q and $F^N_{\Delta H}$ limit, two functions are proposed. The F_Q limit could then be lowered within the range of 2.32 to 2.28, gaining $F^N_{\Delta H}$ margin within the range of 1.55 to 1.65.

   Replace Figure TS.2.1-1 with the one contained in Exhibit B. New safety limit curves were developed for the new $F^N_{\Delta H}$ limit. Change the text and delete a reference to WCAP 8091.

   In Section 3.10, replace the numerical limits for F_Q and $F^N_{\Delta H}$ with functions identifying each limit as a function of the other. These two new functions F_Q $(F^N_{\Delta H})$ and $F^N_{\Delta H}$ (F_Q) are shown in Figure TS.3.10-7. Change the title of Figure TS.3.10-7 in the List of Figures.

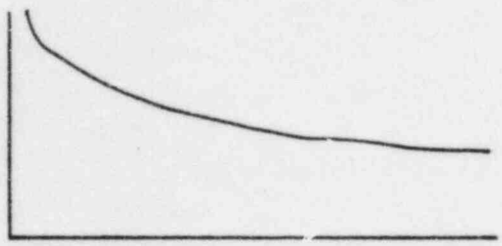
   Change Section 3.10 BASES.

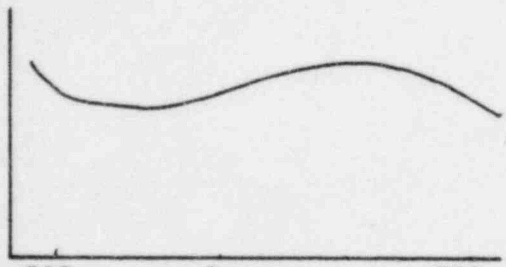
   Delete the BU (E_i) function and associated Bases. Add a statement to the bases identifying the burnup range covered by the LOCA and offsite dose analyses as 0 to 55,000 MWD/MTU.

   In Section 3.10.B.3.(a), increase the setpoint reduction from 1% to 5% for each percent the measured $F^N_{\Delta H}$ exceeds the limit.

EXHIBIT A

Reason for Change
This change will allow more operational flexibility.





As shown on the above curve F_Q typically peaks sharply at the beginning of the cycle and generally decreases throughout the cycle. $F^N_{\Delta H}$ will generally increase as Gadolinia burns out in mid-cycle. Therefore, F_Q can be limiting at the beginning of cycle but $F^N_{\Delta H}$ can limit mid-cycle operations. The requested changes will allow flexibility by allowing, for example, a higher F_Q limit at beginning of cycle and a higher $F^N_{\Delta H}$ limit in mid-cycle.

Section 3.10.B.3.(a) is being changed to increase the setpoint reduction from 1% to 5% for $F^N_{\Delta H}$. A 5% reduction in power, instead of 1% as currently exists in the Technical Specification, is necessary to increase the $F^N_{\Delta H}$ limit 1%.

The F_Q limit is proportional to 1/P.

The $F^N_{\Delta H}$ limit is proportional to [1 + 0.2(1-P)].

If power is reduced by 1% the F_Q limit will increase ≅1%, e.g.:

$$\frac{1}{P-0.01P} = \frac{1}{P(1-0.01)} \cong \frac{1}{P}(1+0.01)$$

If power is reduced by 5% the $F^N_{\Delta H}$ limit will increase ≅1%, e.g.:

$$[1 + 0.2(1-(P-.05P))] = [1 + 0.2(1-P)] + 0.01P$$

When power (P) is high, this approximately equals [1 + 0.2(1-P)] (1 + 0.01).

Therefore, when the F_Q limit is exceeded, power should be reduced 1% for each 1% the limit is exceeded but when the $F^N_{\Delta H}$ limit is exceeded, power should be reduced 5% for each 1% the limit is exceeded.

Significant Hazards Evaluation

As a result of adding the option to raise the $F^N_{\Delta H}$ limit to 1.65 and subsequently lower F_Q to 2.28, several analyses had to be evaluated or re-analyzed. The following were considered:

Transient Analysis
Rod Bow Penalty
Safety Limit Curves
V(Z) Analysis
LOCA Analyses
K(Z) Curve

The transient analyses were evaluated using the higher $F^N_{\Delta H}$ limit of 1.65. The slow rod withdrawal transient was re-anlayzed and indicated that the MDNBR criteria (greater than 1.3) was still met. The other transient analyses did not need to be re-analyzed. This evaluation is documented in Exhibit C.

A new rod bow penalty computational method was used to verify that rod bow would not reduce the MDNBR below 1.3. Margin to the limit was verified as documented in Exhibit C.

The safety limit curves currently in the Technical Specification were based on an $F^N_{\Delta H}$ of 1.58 and an F_Q of 2.71. Therefore, new curves were calculated for an $F^N_{\Delta H}$ of 1.65 and an F_Q of 2.32. The new curves are very similar to the existing curves, but slightly more restrictive. Since the new curves are more restrictive, the overtemperature and overpower ΔT trip setpoints were evaluated to show that the trip would occur before a safety limit was violated. The overtemperature ΔT trip was compared for 1700 and 2400 psig (ΔI=0) and shown to provide adequate protection. In order to account for operation with non-zero ΔI's, the f(ΔI) function was evaluated verifying that non-zero ΔI's would meet the acceptance criteria also. It was concluded that the overpower ΔT trip did not need to be reverified. Therefore, the current trip setpoints are adequate.

For the purpose of the safety limit curve evaluation the acceptance criteria of 1.3 for MDNBR was raised 5% to account for cycle to cycle variations. However, the f(ΔI) will be evaluated as part of each cycle's Reload Safety Evaluation process. For more details see Exhibit C.

The V(Z) function is not dependent upon the $F^N_{\Delta H}$ limit and no change in PDC-II methodology is necessary.

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A new LOCA analyses was calculated for $F^N_{\Delta H}$ = 1.65 and F_Q = 2.28. The results of this analyses are shown in Exhibit D and met all acceptance criteria.

As a result of the LOCA analyses the K(Z) curve was evaluated and found to be adequate in its present form.

Associated with this change in limits, the calculated margin to acceptance criteria has been decreased slightly in some evaluations. However, all results are clearly within all acceptable criteria. The change to paragraph 3.10.B.3.(a) increases an existing restriction.

For these reasons, operation of the Prairie Island Nuclear Generating Plant in accordance with this proposed change will not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) involve a significant reduction in a margin of safety.

2. Reporting Changes

Proposed Changes (pages TS.1-1, TS.3.1-12, TS.3.9-1, TS.3.9-3, TS.3.14-1, TS.3.14-2, TS.3.14-3, TS.3.14-4, TS.3.15-1, TS.4.12-5, TS.6.2-3, TS.6.2-6, TS.6.7-3a thru TS.6.7-8)

Add a definition for "Reportable Event". Change the phrases "reportable occurrence report", "prompt notification with written followup" and other similar phrases to "special report".

Add a requirement for the Safety Audit and Operations Committee to review all reportable events and special reports. Delete requirements associated with the previous reporting system.

Delete Section 6.7.B dealing with reportable occurrences and add a paragraph on reportable events.

Change the name of "Special Reports" in the Environmental Reports section to "Environmental Special Reports".

Add a section 6.7.D concerning Special Reports (non-Environmental).

Reason for Change
These changes are being made as a result of Generic Letter No. 83-43 and recent changes to 10 CFR 50.73.

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Significant Hazards Evaluation
These changes were suggested by Generic Letter 83-43 and are purely administrative in nature.

For these reasons, operation of the Prairie Island Nuclear Generating Plant in accordance with this proposed change will not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) involve a significant reduction in a margin of safety.

3. New Table of Contents

Proposed Changes (TS.i thru TS.viii)
Substitute new Table of Contents for the existing one.

Reason for Change
The new Table of Contents contains more listings, making it easier to locate items.

Significant Hazards Evaluation
This change is purely administrative in nature.

For this reason, operation of the Prairie Island Nuclear Generating Plant in accordance with this proposed change will not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) involve a significant reduction in a margin of safety.

4. Refueling Boron Concentration Changes

Proposed Changes (TS.3.6-2, TS.3.6-5, TS.3.8-1, TS.3.8-3, TS.3.8-4)
Remove the 2000 ppm Boron requirement for the reactor coolant system during various refueling conditions and add a 10% $\Delta K/K$ requirement.

Correct minor errors on page TS.3.8-3

Add "upper internals removal/replacement" to paragraph 3.8.A.7 and to Bases section.

Reason for Change
Currently, two requirements exist in the technical specifications for shutdown reactivity during refueling conditions. 10% ΔK/K is currently referenced in the Technical Specifications (TS.1-6 and TS.3.8-3) along with various refueling conditions. Page TS.3.8-3 suggests the reason for the 2000 ppm Boron concentrations is to "maintain the reactor subcritical by approximately 10% ΔK/K ... ". In order to eliminate any confusion as to which limit must be met, the 2000 ppm Boron concentration should be deleted and replaced with the 10% ΔK/K requirement.

In order to remove/replace the upper internals the water level must be below 20 feet. The upper internals lifting rig contains a platform which must be above water level for personnel to attach and detach the internals to and from the lifting rig.

Significant Hazards Evaluation
Currently, both the 2000 ppm and 10% ΔK/K restrict refueling operations. The proposed changes will delete the 2000 ppm requirement. Generally, the Boron concentration necessary to ensure a 10% ΔK/K shutdown margin is within 100 ppm of 2000 ppm. For a core where the boron concentration to ensure 10% ΔK/K shutdown margin is below 2000 ppm, this change could be said to reduce the safety margin. However, in all cases the acceptance criteria of 10% ΔK/K shutdown margin will be met.

Upper internals removal/replacement takes less than one shift (8 hours). Therefore the probability of failure of the one required RHR pumps during removal and replacement of the internals is small. Also the water level must be greater than 15 feet for shielding concerns, so there will continue to be a significant reservoir of water available as a heat sink. This request for relief is very similar to the situation involving the latching or unlatching of control rods previously approved by the NRC.

For these reasons, operation of the Prairie Island Nuclear Generating Plant in accordance with this proposed change will not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) involve a significant reduction in a margin of safety.

5. Radioactive Source Leakage Test

Proposed Change (Page TS.4.11-2)
Change paragraph 4.11.D on page TS.4.11-2 to read:

Tests resulting in 0.005 microcuries or more of removable contamination on the test sample shall be reported to the Commission on an annual basis.

Correct an error in the Bases.

Reason for Change
This change is requested to correct an improper reference to TS.6.7. Technical Specification Section 6.7, Reporting Requirements, does not describe the reporting requirements for source leak test results.

Significant Hazards Evaluation
The proposed change corrects an inconsistency in the Technical Specifications and is a purely administrative change. For this reason, operation of the Prairie Island Nuclear Generating Plant in accordance with this proposed change will not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) involve a significant reduction in a margin of safety.

6. SRO Shift Requirements

Proposed Changes (Table TS.6.1-1)
Add a requirement for one Senior Reactor Operator (SRO) to be in the control room at all times the reactor is above cold shutdown.

Reason for Change
This new requirement is required by 10 CFR 50.54 (m)(2).

Significant Hazards Evaluation
This change constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications.

For this reason, operation of the Prairie Island Nuclear Generating Plant in accordance with this proposed change will not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) involve a significant reduction in a margin of safety.

7. Deletion of Snubber Table

Proposed Changes (TS.ix, Table TS.3.12-1, TS.3.12-1, TS.4.13-1, thru 3).

Delete Table TS.3.12-1 and all references to Table 3.12-1.
Delete paragraph 3.12.C.

Reason for Change
The NRC Staff's policy of requiring the Technical Specificaitons to contain a detailed list of safety related snubbers has been modified as described in Generic Letter 84-13 dated May 3, 1984. The proposed changes implement the recommendations of this Generic Letter.

Paragraph 3.12.C allowed snubber modification without a prior License Amendment. Since Table TS.3.12-1 is being deleted, snubber modification will continue to be allowed without prior License Amendment. Therefore, paragraph 3.12.C can be deleted.

Significant Hazards Evaluation
This change is administrative in nature.

For this reason, operation of the Prairie Island Nuclear Generating Plant in accordance with this proposed change will not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) involve a significant reduction in a margin of safety.

8. Spray Additive Tank Change

Proposed Change (page TS.3.3-3)
Delete paragraph 3.3.B.1.C (1) and the note below paragraph 3.3.B.1.C (2).

Reason for Change
Modifications to the spray additive tank are operational and paragraph (1) is no longer applicable.

Significant Hazards Evaluation
This change is only administrative in nature.

For this reason, operation of the Prairie Island Nuclear Generating Plant in accordance with this proposed change will not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) involve a significant reduction in a margin of safety.

EXIBIT A

9. Discharge Canal Flow

Proposed Changes (Table TS.3.9-1 1 of 2)
Delete Item 2c. Add Item 6.

Reason for Change
The plant discharge flow rate is controlled and constant, making continuous flow rate measurement instrumentation unnecessary. The flow rate is a function of the number of circulating water pumps operating, the number of sluice gates open (either full open or fully closed), the position of the recycle gates and the difference in water level between the discharge canal and the river. The gate position and the number of pumps running are controlled by the operator. The height of the discharge canal depends upon gate positions. The height of the river varies but slowly, due to the location of lock and dam No. 3, 1½ miles downstream of the plant.

Significant Hazards Evaluation
The flow rate is determined daily and when operational changes are made using the factors discussed above. Since the flow rate is basically controlled by the operator, the function of monitoring the plant discharge flow rate will continue to be accomplished. This will have no effect on the safety margin of the plant.

For this reason, operation of the Prairie Island Nuclear Generating Plant in accordance with this proposed change will not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) involve a significant reduction in a margin of safety.

10. Radioactive Effluent Monitoring Instrumentation Requirements (Table TS.3.9-2, Table TS.4.17-1 and TS.4.17-2)

Proposed Changes
The Technical Specification Tables, TS.3.9-2, TS.4.17-1 and TS.4.17-2, are to be revised as noted in Exhibit B by deleting unnecessary tests.

Reason for Change
Table TS.4.17-1 quarterly functional tests for the liquid radwaste effluent, steam generator blowdown and discharge canal, flow instruments are to be deleted since flow instrumentation provides indication only. Indication will continue to be checked daily during releases. Flow instrumentation will continue to be calibrated during refueling shutdown intervals. The composite samples quarterly functional test and the 18 month calibration are to be deleted due to the adequacy of the daily test for normal operation and the weekly test for sample volume.

On Table TS.4.17-2 and TS.3.9-2 the sample flow monitor was replaced with the sample flow integrator and associated surveillance. The function of the Sample Flow Rate Monitor, a rotameter, is to indicate low flow. The Sample Flow Integrator is used in the calculation of effluent releases. Therefore, it is appropriate to reference the Sample Flow Integrator in the Technical Specifications.

Significant Hazards Evaluation
Removal of inappropriate surveillance and addition of appropriate testing has been requested.

The composite sampler is checked daily for normal operation. During normal plant operation no adjustments to the sampler timer is required.

We believe the requested changes are administrative in nature and do not constitute a Significant Hazards consideration since they do not:

1) Involve a significant increase in the probability of consequences of an accident previously evaluated; or

2) create the possibility of a new or different kind of accident from any accident previously evaluated; or

3) involve a significant reduction in a margin of safety.

11. Radiation Environmental Monitoring Program Sample Collection and Analysis

Proposed Change
The Technical Specification Table TS.4.10-1 is to be revised as noted in Exhibit B.

Reason for Change
The minor word change in Item 3.d from "Sediment and shoreline" to "Sediment from shoreline" is requested at this time to achieve consistency with the Standard Technical Specifications for PWR NUREG-0473 Revision 2, Table 3.12-1 page 3/4 12-4.

Additional minor changes on Table, TS.4.10-1, are also requested at this time. The additional minor changes are to correct errors.

Significant Hazards Evaluation
The proposed change in wording of Specification Table 4.10-1 is a purely administrative change to the Technical Specifications.

For this reason operation of the Prairie Island Nuclear Generating Plant in accordance with the proposed changes would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) involve a significant reduction in a margin of safety.

12. Title Change

Proposed Change (Figure TS.6.1-1 and pages TS.6.2-1, TS.6.2-3, TS.6.2-5, TS.6.7-4)

In place of "Director of Nuclear Generation" substitute "Vice President Nuclear Generation". Other titles on Figure TS.6.1-1 have been updated.

The chart has been simplified slightly by starting with the highest position that governs all groups working directly with the nuclear units, i.e., the Senior Vice President Power Supply.

Reason for Change
The Director of Nuclear Generation title has been elevated to Vice President Nuclear Generation. No other organizational changes are involved.

Significant Hazards Evaluation
This is an administrative change.

For this reason, operation of the Prairie Island Nuclear Generating Plant in accordance with this proposed change will not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) involve a significant reduction in a margin of safety.