March 20, 1991

Docket Nos. 50-282
and 50-306

DISTRIBUTION
Docket File
NRC & Local PDRs
Plant R/F
GHill(8)
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JZwolinski
PShuttleworth
EJordan
DHagan

Mr. T. M. Parker, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 5401

Dear Mr. Parker:

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NOS. 1 AND 2 - AMENDMENT NOS. 94 AND 87 TO FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60 (TAC NOS. 79307 AND 79308)

The Commission has issued the enclosed Amendment No. 94 to Facility Operating License No. DPR-42 and Amendment No. 87 to the Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated November 14, 1990.

The amendments change the TS Section 3.10.G and its associated Bases to allow continued operation for 72 hours for diagnosis and repair, with one or more control rods immovable due to an electrical problem in the rod control system, provided all affected control rods remain trippable.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/s/

Armando Masciantonio, Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 94 to License No. DPR-42
2. Amendment No. 87 to License No. DPR-60
3. Safety Evaluation

cc w/enclosures:
See next page

NRC FILE CENTER COPY

LA/PD31:DRP345	PM/PD31:DRP345	SRXB	D/PD31:DRP345	OGC
PShuttleworth	AMasciantonio	RJones for	LMarsh	
02/13/91	02/14/91	02/14/91	02/19/91	03/18/91

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/s/

Armando Masciantonio, Project Manager
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			3/19/91	

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 20, 1991

Docket Nos. 50-282
and 50-306

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Nuclear Support Services
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

Dear Mr. Parker:

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NOS. 1 AND 2 - AMENDMENT NOS. 94 AND 87 TO FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60 (TAC NOS. 79307 AND 79308)

The Commission has issued the enclosed Amendment No. 94 to Facility Operating License No. DPR-42 and Amendment No. 87 to the Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated November 14, 1990.

The amendments change the TS Section 3.10.G and its associated Bases to allow continued operation for 72 hours for diagnosis and repair, with one or more control rods immovable due to an electrical problem in the rod control system, provided all affected control rods remain trippable.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

A. S. Masciantonio

Armando Masciantonio, Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
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cc w/enclosures:
See next page

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Goodhue County Courthouse
Red Wing, Minnesota 55066

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 94
License No. DPR-42

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northern States Power Company (the licensee) dated November 14, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-42 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 94, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

William D. Long

for L. B. Marsh, Director
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 20, 1991

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 87
License No. DPR-60

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northern States Power Company (the licensee) dated November 14, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-60 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 87, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

William O. Long for

L. B. Marsh, Director
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 20, 1991

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (continued)

3.10.G. Control Rod Operability Limitations

1. An inoperable rod is a rod which (a) does not trip, (b) cannot be moved as a result of excessive friction or mechanical interference, or (c) is declared inoperable under specification 3.10.E or 3.10.H.

2. The reactor shall be brought to the HOT SHUTDOWN condition within 6 hours should more than one inoperable rod be discovered during POWER OPERATION.

3. If the inoperable rod is located below the 200 step level and is capable of being tripped, or if the rod is located below the 30 step level whether or not it is capable of being tripped, then the insertion limits specified in the CORE OPERATING LIMITS REPORT apply.

4. If the inoperable rod cannot be located, or if the inoperable rod is located above the 30 step level and cannot be tripped, then the insertion limits specified in the CORE OPERATING LIMITS REPORT apply.

5. If POWER OPERATION is continued with one inoperable rod, the potential ejected rod worth and associated transient power distribution peaking factors shall be determined by analysis within 30 days unless the rod is earlier made OPERABLE. The analysis shall include due allowance for nonuniform fuel depletion in the neighborhood of the inoperable rod. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, THERMAL POWER shall be reduced to a level consistent with the safety analysis.

6. With one or more rod(s) trippable, but immovable due to an electrical problem in the rod control system, within one hour verify that control rod position is within the rod insertion limits specified in section 3.10.D. Restore the Rod Control System to OPERABLE status within 72 hours or declare the affected rod(s) inoperable and apply the limitations specified in sections 3.10.G.2 through 3.10.G.5.

H. Rod Drop Time

At operating temperature and full flow, the drop time of each RCCA shall be no greater than 1.8 seconds from loss of stationary gripper coil voltage to dashpot entry. If the time is greater than 1.8 seconds, the rod shall be declared inoperable.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases continued

D. Rod Insertion Limits (continued)

as stated above. Therefore, this specification has been written to further minimize the likelihood of any hypothesized event during the performance of these tests later in life. This is accomplished by limiting to two hours per year the time the reactor can be in this type of configuration, and requiring that a rod drop test is performed on the rod to be measured prior to performance of test.

Operation with abnormal rod configuration during low power and zero power testing is permitted because of the brief period of the test and because special precautions are taken during the test.

E. Rod Misalignment Limitation

Rod misalignment requirements are specified to ensure that power distributions more severe than those assumed in the safety analyses do not occur.

F. Inoperable Rod Position Indicator Channels

The rod position indicator channel is sufficiently accurate to detect a rod ± 7 inches away from its demand position. A misalignment less than 15 inches does not lead to over-limit power peaking factors. If the rod position indicator channel is not operable, the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors, and/or core thermocouples, and/or movable incore detectors, will be used to verify power distribution symmetry. These indirect measurements do not have the same resolution if the bank is near either end of the core, because a 15-inch misalignment would have no effect on power distributions. Therefore, it is necessary to apply the indirect checks following significant rod motion.

G. Control Rod Operability Limitations

One inoperable control rod is acceptable provided that the power distribution limits are met, trip shutdown capability is available, and provided the potential hypothetical ejection of the inoperable rod is not worse than the cases analyzed in the safety analysis report. The rod ejection accident for an isolated fully-inserted rod will be worse if the residence time of the rod is long enough to cause significant non-uniform fuel depletion. The four-week period is short compared with the time interval required to achieve a significant non-uniform fuel depletion.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Bases continued

In most cases, when more than one rod is found to be trippable but immovable, the malfunction can be traced to the Rod Control System. Since the majority of Rod Control System malfunctions can be repaired without reactor shutdown and since the unit conditions are not outside any accident analysis assumptions, there is time available to locate the malfunction and restore the rods to an OPERABLE status. The rod insertion and power distribution limitations in specifications 3.10.D, 3.10.G.3, 3.10.G.4 and 3.10.G.5 ensure that core design limits are not exceeded.

H. Rod Drop Time

The required drop time to dashpot entry is consistent with the safety analysis.

I. Monitor Inoperability Requirements

If either the rod bank insertion limit monitor or rod position deviation monitor are inoperable, additional surveillance is required to ensure adequate shutdown margin is maintained.

If the rod position deviation monitor and quadrant power tilt monitor(s) are inoperable, the overpower reactor trip setpoint is reduced (and also power) to ensure that adequate core protection is provided in the event that unsatisfactory conditions arise that could affect radial power distribution.

Increased surveillance is required, if the quadrant power tilt monitors are inoperable and a load change occurs, in order to confirm satisfactory power distribution behavior. The automatic alarm functions related to QUADRANT POWER TILT must be considered incapable of alerting the operator to unsatisfactory power distribution conditions.

J. DNB Parameters

The RCS flow rate, T_{avg}, and Pressurizer Pressure requirements are based on transient analyses assumptions. The flow rate shall be verified by calorimetric flow data and/or elbow taps. Elbow taps are used in the reactor coolant system as an instrument device that indicates the status of the reactor coolant flow. The basic function of this device is to provide information as to whether or not a reduction in flow rate has occurred. If a reduction in flow rate is indicated below the value specified in the CORE OPERATING LIMITS REPORT, shutdown is required to investigate adequacy of core cooling during operation.

ATTACHMENT TO LICENSE AMENDMENT NO. 94

FACILITY OPERATING LICENSE NO. DPR-42

DOCKET NO. 50-282

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change

REMOVE	INSERT
TS-iv	TS-iv
TS-x	TS-x
TS.3.10-7	TS.3.10-7
B.3.10-9	B.3.10-9
B.3.10-10	B.3.10-10

ATTACHMENT TO LICENSE AMENDMENT NO. 87

FACILITY OPERATING LICENSE NO. AND DPR-60

DOCKET NO. 50-306

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change

REMOVE	INSERT
TS-iv	TS-iv
TS-x	TS-x
TS.3.10-7	TS.3.10-7
B.3.10-9	B.3.10-9
B.3.10-10	B.3.10-10

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 94 AND 87 TO

FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By letter dated November 14, 1990, Northern States Power Company (NSP or the licensee) requested amendments to the Technical Specifications (TS) appended to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2. The proposed amendments would revise TS Section 3.10.G and its associated Bases to allow continued operation for 72 hours for diagnosis and repair, with one or more control rods immovable due to an electrical problem in the rod control system, provided all affected control rods remain trippable.

As described in the Updated Safety Analysis Report, the control rod drive system provides rod cluster control assembly insertion and withdrawal rates consistent with the required reactivity changes for reactor operational load changes. It also provides a fast insertion rate during a "trip" of the rod cluster control assemblies which results in a rapid shutdown of the reactor.

Current specification 3.10.G.1 specifies that a control rod which cannot be moved by its drive mechanism and cannot be corrected within 8 hours be declared inoperable. Specification 3.10.G.2 specifies that the reactor shall be brought to the hot shutdown condition within 6 hours should more than one inoperable rod be discovered during power operations. These current specifications fail to distinguish the greater significance of rods being immovable due to mechanical interference. Acknowledgement of the lesser significance of electrically immovable rods would allow additional time for diagnosis and repair of malfunctioning equipment while maintaining safe operation of the plant.

The proposed changes provide distinct action statements for immovable rods that are more consistent with the significance of the malfunctions. A rod that is immovable due to excessive friction of mechanical interference is more significant than a rod that cannot be stepped due to an electrical malfunction, but remains trippable. Distinguishing between these types of malfunctions will allow an appropriate time period to complete corrective action commensurate with the significance of the malfunction. The proposed changes to the TS would allow continued operation for 72 hours for diagnosis and repair, with one or more control rods immovable due to an electrical problem in the rod control system .

2.0 EVALUATION

The proposed TS change would allow operation to continue for 72 hours for diagnosis and repair for the case where one or more control rod assemblies are electrically inoperable. The existing Prairie Island TS require the Plants to be in HOT SHUTDOWN within 6 hours. A rod that is inoperable due to being untrippable is a more significant failure than a rod that cannot be moved due to an electrical failure but is still trippable. The change distinguishes between these failures and allows more time for repair of the rod(s) that cannot be moved due to an electrical failure. At the end of the 72 hours allowed out-of-service time, the proposed specifications declare any remaining electrically immovable rods, inoperable, and applies the requirements of specification 3.10.G.2 which requires that unit to be taken to HOT SHUTDOWN within 6 hours if more than one control rod is inoperable.

Based on our review, we find the proposed changes to the Priaire Island TS, as requested by Northern States Power Company, to be acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes in requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

4.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public. The staff, therefore, concludes that the proposed changes are acceptable.

Principal Contributor: A. Masciantonio

Dated: March 20, 1991