March 8, 1995

Mr. Roger O. Anderson, Director
Licensing and Management Issues
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: FREQUENCY OF RESIDUAL HEAT REMOVAL (RHR) SYSTEM LEAKAGE TESTING (TAC NOS. M91292 AND M91293)

Dear Mr. Anderson:

The Commission has issued the enclosed Amendment No. 115 to Facility Operating License No. DPR-42 and Amendment No. 108 to Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated January 13, 1995.

The amendments revise Prairie Island Nuclear Generating Plant TS 4.4-D.1, "Residual Heat Removal System," to extend the interval for RHR system leakage testing from "once every 12 months" to "once every refueling outage." The revised Bases also reflect this change.

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,

ORIGINAL SIGNED BY

Charles R. Thomas, Acting Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosures:
1. Amendment No. 115 to DPR-42
2. Amendment No. 108 to DPR-60
3. Safety Evaluation

cc w/encls: See next page

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cc:

J. E. Silberg, Esquire
Shaw, Pittman, Potts and Trowbridge
2300 N Street, N. W.
Washington DC 20037

Site General Manager
Prairie Island Nuclear Generating
Plant
Northern States Power Company
1717 Wakonade Drive East
Welch, Minnesota 55089

Adonis A. Neblett
Assistant Attorney General
Office of Attorney General
455 Minnesota Street
Suite 900
St. Paul, Minnesota 55101-2127

U.S. Nuclear Regulatory Commission
Resident Inspector Office
1719 Wakonade Drive East
Welch, Minnesota 55089-9642

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, Illinois 60532-4351

Mr. Jeff Cole, Auditor/Treasurer
Goodhue County Courthouse
Box 408
Red Wing, Minnesota 55066-0408

Kris Sanda, Commissioner
Department of Public Service
121 Seventh Place East
Suite 200
St. Paul, Minnesota 55101-2145

Site Licensing
Prairie Island Nuclear Generating
Plant
Northern States Power Company
1717 Wakonade Drive East
Welch, Minnesota 55089

DATED: March 8, 1995

AMENDMENT NO. 115 TO FACILITY OPERATING LICENSE NO. DPR-42-PRAIRIE ISLAND UNIT 1
AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NO. DPR-60-PRAIRIE ISLAND UNIT 2

Docket File
PUBLIC
PDIII-1 Reading
J. Roe
J. Hannon
C. Jamerson
C. Thomas (2)
OGC-WF
G. Hill (4)
C. Grimes, O-11F23
H. Abelson
ACRS (4)
OPA
OC/LFDCB
W. Kropp, RIII
SEDB

cc: Plant Service list

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 115
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 January 13, 1995, 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. 115 , 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 the date of issuance, with full implementation within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Kevin A. Connaughton for

John N. Hannon, Director
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 8, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 115

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 vertical lines indicating the area of change.

REMOVE	INSERT
TS 4.4-4	TS 4.4-4
TS B.4.4-2	TS B.4.4-2

b. Cold DOP testing shall be performed after each complete or partial replacement of a HEPA filter bank or after any structural maintenance on the system housing that could affect the HEPA bank bypass leakage.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of a charcoal adsorber bank or after any structural maintenance on the system housing that could affect the charcoal adsorber bank bypass leakage.

d. Each circuit shall be operated with the heaters on at least 10 hours every month.

5. Perform an air distribution test on the HEPA filter bank after any maintenance or testing that could affect the air distribution within the systems. The test shall be performed at rated flow rate (±10%). The results of the test shall show the air distribution is uniform within ±20%.

C. <u>Containment Vacuum Breakers</u>

The air-operated valve in each vent line shall be tested at quarterly intervals to demonstrate that a simulated containment vacuum of 0.5 psi will open the valve and a simulated accident signal will close the valve. The check valves as well as the butterfly valves will be leak-tested during each refueling shutdown in accordance with the requirements of Specification 4.4.A.2.

D. <u>Residual Heat Removal System</u>

1. Those portions of the residual heat removal system external to the isolation valves at the containment, shall be hydrostatically tested for leakage during each refueling shutdown.

2. Visual inspection shall be made for excessive leakage from components of the system. Any visual leakage that cannot be stopped at test conditions shall be measured by collection and weighing or by another equivalent method.

3. The acceptance criterion is that maximum allowable leakage from either train of the recirculation heat removal system components (which includes valve stems; flanges and pump seals) shall not exceed two gallons per hour when the system is at 350 psig.

4. Repairs shall be made as required to maintain leakage within the acceptance criterion in Specification 4.4.D.3

5. If repairs are not completed within 7 days, the reactor shall be shut down and depressurized until repairs are effected and the acceptance criterion in 3. above is satisfied.

4.4 CONTAINMENT SYSTEM TESTS

Bases continued

Several penetrations of the containment vessel and the shield building could, in the event of leakage past their isolation valves, result in leakage being conveyed across the annulus by the penetrations themselves, thus bypassing the function of the Shield Building Ventilation System (Reference 5). Such leakage is estimated not to exceed .025% per day. A special zone of the auxiliary building has minimum-leakage construction and controlled access, and is designated as a special ventilation zone where such leakage would be collected by either of two redundant trains of the Auxiliary Building Special Ventilation System. This system, when activated, will supplant the normal ventilation and draw a vacuum throughout the zone such that all outleakage will be through particulate and charcoal filters which exhaust to the shield building exhaust stack.

The design basis loss-of-coolant accident was initially evaluated by the AEC staff (Reference 3) assuming primary containment leak rate of 0.5% per day at the peak accident pressure. Another conservative assumption in the calculation is that primary containment leakage directly to the ABSVZ is 0.1% per day and leakage directly to the environs is 0.01% per day. The resulting two-hour doses at the nearest SITE BOUNDARY and 30-day doses at the low population zone radius of 1½ miles are less than guidelines presented in 10CFR100.

Initial leakage testing of the shield building and the ABSV resulted in a greater inleakage than the design basis. The staff has reevaluated doses for these higher inleakage rates and found that for a primary containment leak rate of 0.25% per day at peak accident pressure, the offsite doses are about the same as those initially calculated for higher primary containment leakage and lower secondary containment in-leakage (Reference 6).

The Residual Heat Removal Systems functionally become a part of the containment volume during the post-accident period when their operation is changed over from the injection phase to the recirculation phase. Redundancy and independence of the systems permit a leaking system to be isolated from the containment during this period, and the possible consequences of leakage are minor relative to those of the Design Basis Accident (Reference 4); however, their partial role in containment warrants surveillance of their leak-tightness.

The limiting leakage rates from the recirculation heat removal system are judgment values based primarily on assuring that the components could operate without mechanical failure for a period on the order of 200 days after a design basis accident. The test pressure, 350 psig, gives an adequate margin over the highest pressure within the system after a design basis accident. A recirculation heat removal system leakage of 2 gal/hr will limit off-site exposure due to leakage to insignificant levels relative to those calculated for leakage directly from the containment in the design basis accident.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 108
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 January 13, 1995, 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. 108, 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 the date of issuance, with full implementation within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Kevin A. Connaughton for

John N. Hannon, Director
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 8, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 108

FACILITY OPERATING LICENSE NO. 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 vertical lines indicating the area of change.

REMOVE	INSERT
TS 4.4-4	TS 4.4-4
B.4.4-2	B.4.4-2

b. Cold DOP testing shall be performed after each complete or partial replacement of a HEPA filter bank or after any structural maintenance on the system housing that could affect the HEPA bank bypass leakage.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of a charcoal adsorber bank or after any structural maintenance on the system housing that could affect the charcoal adsorber bank bypass leakage.

d. Each circuit shall be operated with the heaters on at least 10 hours every month.

5. Perform an air distribution test on the HEPA filter bank after any maintenance or testing that could affect the air distribution within the systems. The test shall be performed at rated flow rate (±10%). The results of the test shall show the air distribution is uniform within ±20%.

C. Containment Vacuum Breakers

The air-operated valve in each vent line shall be tested at quarterly intervals to demonstrate that a simulated containment vacuum of 0.5 psi will open the valve and a simulated accident signal will close the valve. The check valves as well as the butterfly valves will be leak-tested during each refueling shutdown in accordance with the requirements of Specification 4.4.A.2.

D. Residual Heat Removal System

1. Those portions of the residual heat removal system external to the isolation valves at the containment, shall be hydrostatically tested for leakage during each refueling shutdown.

2. Visual inspection shall be made for excessive leakage from components of the system. Any visual leakage that cannot be stopped at test conditions shall be measured by collection and weighing or by another equivalent method.

3. The acceptance criterion is that maximum allowable leakage from either train of the recirculation heat removal system components (which includes valve stems; flanges and pump seals) shall not exceed two gallons per hour when the system is at 350 psig.

4. Repairs shall be made as required to maintain leakage within the acceptance criterion in Specification 4.4.D.3

5. If repairs are not completed within 7 days, the reactor shall be shut down and depressurized until repairs are effected and the acceptance criterion in 3. above is satisfied.

4.4 CONTAINMENT SYSTEM TESTS

Bases continued

Several penetrations of the containment vessel and the shield building could, in the event of leakage past their isolation valves, result in leakage being conveyed across the annulus by the penetrations themselves, thus bypassing the function of the Shield Building Ventilation System (Reference 5). Such leakage is estimated not to exceed .025% per day. A special zone of the auxiliary building has minimum-leakage construction and controlled access, and is designated as a special ventilation zone where such leakage would be collected by either of two redundant trains of the Auxiliary Building Special Ventilation System. This system, when activated, will supplant the normal ventilation and draw a vacuum throughout the zone such that all outleakage will be through particulate and charcoal filters which exhaust to the shield building exhaust stack.

The design basis loss-of-coolant accident was initially evaluated by the AEC staff (Reference 3) assuming primary containment leak rate of 0.5% per day at the peak accident pressure. Another conservative assumption in the calculation is that primary containment leakage directly to the ABSVZ is 0.1% per day and leakage directly to the environs is 0.01% per day. The resulting two-hour doses at the nearest SITE BOUNDARY and 30-day doses at the low population zone radius of 1½ miles are less than guidelines presented in 10CFR100.

Initial leakage testing of the shield building and the ABSV resulted in a greater inleakage than the design basis. The staff has reevaluated doses for these higher inleakage rates and found that for a primary containment leak rate of 0.25% per day at peak accident pressure, the offsite doses are about the same as those initially calculated for higher primary containment leakage and lower secondary containment in-leakage (Reference 6).

The Residual Heat Removal Systems functionally become a part of the containment volume during the post-accident period when their operation is changed over from the injection phase to the recirculation phase. Redundancy and independence of the systems permit a leaking system to be isolated from the containment during this period, and the possible consequences of leakage are minor relative to those of the Design Basis Accident (Reference 4); however, their partial role in containment warrants surveillance of their leak-tightness.

The limiting leakage rates from the recirculation heat removal system are judgment values based primarily on assuring that the components could operate without mechanical failure for a period on the order of 200 days after a design basis accident. The test pressure, 350 psig, gives an adequate margin over the highest pressure within the system after a design basis accident. A recirculation heat removal system leakage of 2 gal/hr will limit off-site exposure due to leakage to insignificant levels relative to those calculated for leakage directly from the containment in the design basis accident.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 115 AND 108 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 January 13, 1995, the Northern States Power Company (NSP or the licensee) submitted a proposed revision 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 revision would revise TS 4.4.D.1 to extend the interval for residual heat removal (RHR) system leakage testing from the current "once every 12 months" to "once every refueling outage."

Following a design-basis, loss-of-coolant accident, the RHR system becomes an extension of the containment once a changeover from the safety injection phase to the recirculation phase occurs. To minimize post-accident leakage from the RHR system, TS 4.4.D.3 imposes a limit of 2 gallons per hour from either RHR train, at a system pressure of 350 psig. This limit ensures that the incremental offsite exposure from this source will be insignificant when compared to the exposure resulting from direct containment leakage following the design-basis accident. TS 4.4.D.1 requires that those portions of the RHR system located outboard of the containment isolation valves (which are open during post-accident operation) be hydrostatically tested at regular intervals to ensure that this limit is not exceeded.

2.0 EVALUATION

Under the currently specified 12-month test interval, the RHR leakage test must be performed during power operation, during which time the RHR system is not in operation. The test is conducted by pressurizing the RHR system to the 350 psig test pressure using coolant supplied via a letdown line from the chemical and volume control system (CVCS). Leakage is determined by visual observation. Because the CVCS system is connected to only one RHR train, testing of the other train requires that a cross-connect valve between the two trains be temporarily opened. This valve is normally maintained closed to provide train separation and redundancy.

If the leakage test is conducted during a refueling outage, the RHR system is operating in the shutdown cooling mode and, during the initial stages of cooldown, is pressurized to above the 350 psig test pressure. No change in valve configuration is necessary to test either RHR train. Accordingly, performance of the leakage test during an outage is less complex and requires fewer manual actions than when performed during power operation.

The proposed reduction in the testing frequency increases the potential for leakage to go undetected for a longer period of time (approximately 6 months). However, previous testing on a 12-month interval has not disclosed significant leakage and extending the interval to refueling outages greatly reduces the complexity of the test. It is also likely that during the routine quarterly functional testing and inspection of the RHR system (at a much lower pressure than 350 psig), any significant leakage would be identified. Additionally, leakage testing of the RHR system on a refueling interval is consistent with the Westinghouse Standard Technical Specifications (NUREG-1431, Rev.0) and with the interval for Type B and C containment penetration testing specified in Appendix J of 10 CFR Part 50.

3.0 CONCLUSION

Based on the above evaluation, we find the revision to TS 4.4.D.1 extending the RHR system leakage test interval from "once every 12 months" to "once every refueling outage" for Prairie Island Units 1 and 2 to be acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Minnesota State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change surveillance requirements. The NRC 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 the amendments involve no significant hazards consideration and there has been no public comment on such finding (60 FR 6308). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 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 the amendments.

6.0 CONCLUSION

The Commission 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.

Principal Contributor: H. Abelson

Date: March 8, 1995