July 29, 1993

Docket Nos. 50-282
and 50-306

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

Dear Mr. Anderson:

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NOS. 1 AND 2 - AMENDMENT NOS. 107 AND 100 TO FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60 (TAC NOS. M83379 AND M83380)

The Commission has issued the enclosed Amendment No. 107 to Facility Operating License No. DPR-42 and Amendment No. 100 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 in response to your application dated May 7, 1992, as revised June 24, 1993.

The amendments (1) relocate the Containment Penetration List from Section 4.4 of the Technical Specifications into plant procedures in accordance with the guidance of Generic Letter 91-08, (2) changes Section 3.6.C of the Technical Specifications to clarify when non-automatic containment isolation valves are required to be operable and what actions are to be taken in response to inoperability of a non-automatic containment isolation valve, and (3) deletes condensate cross-connect valve C-41-1 from Section 3.4.B.1.g of the Technical Specifications.

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

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



Enclosures:
1. Amendment No. 107 to DPR-42
2. Amendment No. 100 to DPR-60
3. Safety Evaluation



cc w/enclosures:
See next page

| OFFICE | LA:PD31 | PM:PD31 | BC:SCSB | OGC | AD:PDIII-1 |
|---|---|---|---|---|---|
| NAME | CJamerson | MGamberoni | RBarrett | EHoller | WDean |
| DATE | 7/9/93 | 6/28/93 | 7/6/93 | 7/16/93 | 7/29/93 |

OFFICIAL RECORD COPY
FILENAME: G:\WPDOCS\PRAIRIE\PI83379.AMD

DF01

Mr. Roger O. Anderson, Director
Northern States Power Company

Prairie Island Nuclear Generating
Plant

cc:

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

Mr. E. L. Watzl, Site General Manager
Prairie Island Nuclear Generating
Plant
Northern States Power Company
Route 2
Welch, Minnesota 55089

Lisa R. Tiegel
Assistant Attorney General
Environmental Protection Division
Suite 200
520 Lafayette Road
St. Paul, Minnesota 55155

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

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

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

DATED: July 29, 1993

AMENDMENT NO. 107 TO FACILITY OPERATING LICENSE NO. DPR-42-PRAIRIE ISLAND UNIT 1
AMENDMENT NO. 100 TO FACILITY OPERATING LICENSE NO. DPR-60-PRAIRIE ISLAND UNIT 2

Docket File
NRC & Local PDRs
PDIII-1 Reading
J. Roe, 13/E/4
J. Zwolinski, 13/H/24
L. Marsh
C. Jamerson
M. Gamberoni
W. Long
OGC-WF
D. Hagan, 3206 MNBB
G. Hill (4), P-137
Wanda Jones, MNBB-3701
C. Grimes, 11/F/23
C. McCracken 8/D/1
ACRS (10)
OPA
OC/LFDCB
W. Shafer, R-III

cc: Plant Service list

020054

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. 107
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 May 7, 1992, as revised June 24, 1993, 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. 107 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 its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Hall for

William M. Dean, Acting 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: July 29, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 107

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-xii | TS-xii |
| TS.1-2 | TS.1-2 |
| TS.3.4-2 | TS.3.4-2 |
| TS.3.6-1 | TS.3.6-1 |
| TS.4.4-2 | TS.4.4-2 |
| TABLE TS.4.4-1 (Pgs 1-5) | ------ |
| B.3.6-1 | B.3.6-1 |
| B.3.6-2 | B.3.6-2 |
| ----- | B.3.6-3 |
| B.4.4-1 | B.4.4-1 |
| B.4.4-2 | B.4.4-2 |

## TECHNICAL SPECIFICATIONS

## LIST OF TABLES


| TS TABLE | TITLE |
|---|---|
| 3.5-1 | Engineered Safety Features Initiation Instrument Limiting Set Points |
| 3.5-2 | Instrument Operating Conditions for Reactor Trip |
| 3.5-3 | Instrument Operating Conditions for Emergency Cooling System |
| 3.5-4 | Instrument Operating Conditions for Isolation Functions |
| 3.5-5 | Instrument Operating Conditions for Ventilation Systems |
| 3.5-6 | Instrument Operating Conditions for Auxiliary Electrical System |
| 3.9-1 | Radioactive Liquid Effluent Monitoring Instrumentation |
| 3.9-2 | Radioactive Gaseous Effluent Monitoring instrumentation |
| 3.14-1 | Safety Related Fire Detection Instruments |
| 3.15-1 | Event Monitoring instrumentation - Process & Containment |
| 3.15-2 | Event Monitoring instrumentation - Radiation |
| 4.1-1 | Minimum Frequencies for Checks, Calibrations and Test of Instrument Channels |
| 4.1-2A | Minimum Frequencies for Equipment Tests |
| 4.1-2B | Minimum Frequencies for Sampling Tests |
| 4.2-1 | Special Inservice Inspection Requirements |
| 4.10-1 | Radiation Environmental Monitoring Program (REMP) Sample Collection and Analysis |
| 4.10-2 | RFMP - Maximum Values for the Lower Limits of Detection |
| 4.10-3 | RFMP - Reporting Levels for Radioactivity Concentrations in Environmental Samples |
| 4.12-1 | Steam Generator Tube Inspection |
| 4.13-1 | Snubber Visual Inspection Interval |
| 4.17-1 | Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements |
| 4.17-2 | Radioactive Gaseous Effluent Monitoring instrumentation Surveillance Requirements |
| 4.17-3 | Radioactive Liquid Waste Sampling and Analysis Program |
| 4.17-4 | Radioactive Gaseous Waste Sampling and Analysis Program |
| 5.5-1 | Anticipated Annual Release of Radioactive Material in Liquid Effluents From Prairie Island Nuclear Generating Plant (Per Unit) |
| 5.5-2 | Anticipated Annual Release of Radioactive Nuclides in Gaseous Effluent From Prairie Island Nuclear Generating Plant (Per Unit) |
| 6.1-1 | Minimum Shift Crew Composition |

CONTAINMENT INTEGRITY

CONTAINMENT INTEGRITY shall exist when:

1. Penetrations required to be isolated during accident conditions are either:

    a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

    b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Specifications 3.6.C and 3.6.D.

2. The equipment hatch is closed and sealed.

3. Each air lock is in compliance with the requirements of Specification 3.6.M.

4. The containment leakage rates are within their required limits.

COLD SHUTDOWN

A reactor is in the COLD SHUTDOWN condition when the reactor is subcritical by at least 1% $\Delta k/k$ and the reactor coolant average temperature is less than 200° F.

CORE ALTERATION

CORE ALTERATION is the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel, which may affect core reactivity. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.7.A.6. Plant operation within these operating limits is addressed in individual specifications.

3.4.B.1.d. A minimum of 100,000 gallons of water is available in the condensate storage tanks and a backup supply of river water is available through the cooling water system.

e. Motor operated valves MV-32242 and MV-32243 (Unit 2 valves MV-32248 and MV-32249) shall have valve position monitor lights OPERABLE and shall be locked in the open position by having the motor control center supply breakers physically locked in the off position.

f. Manual valves in the above systems that could (if one is improperly positioned) reduce flow below that assumed for accident analysis shall be locked in the proper position for emergency use. During POWER OPERATION, changes in valve position will be under direct administrative control.

g. The condensate supply cross connect valve C-41-2, to the auxiliary feedwater pumps shall be blocked and tagged open. Any changes in position of this valve shall be under direct administrative control.

2. During STARTUP OPERATION or POWER OPERATION, any one of the following conditions of inoperability may exist for each unit provided STARTUP OPERATION is discontinued until OPERABILITY is restored. If OPERABILITY is not restored within the time specified, place the affected unit (or either unit in the case of a motor driven AFW pump inoperability) in at least HOT SHUTDOWN within the next 6 hours and reduce reactor coolant system average temperature below 350°F within the following 6 hours.

a. A turbine driven AFW pump, system valves and piping may be inoperable for 72 hours.

b. A motor driven AFW pump, system valves and piping may be inoperable for 72 hours.

c. The condensate storage tanks may be inoperable for 48 hours provided the cooling water system is available as a backup supply of water to the auxiliary feedwater pumps.

d. The backup supply of river water provided by the cooling water system may be inoperable for 48 hours provided a minimum of 100,000 gallons of water is available in the condensate storage tanks.

e. The valve position monitor lights for motor operated valves MV-32242 and MV-32243 (Unit 2 valves MV-32248 and MV-32249) may be inoperable for 72 hours provided the associated valves' positions are verified to be open once each shift.

## 3.6 CONTAINMENT SYSTEM

### Applicability

Applies to the integrity of the containment system.

### Objective

To define the operating status of the containment system for plant operation.

### Specification

A. Containment Integrity

1. A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 200°F unless CONTAINMENT INTEGRITY is maintained.

2. If these conditions cannot be satisfied, within one hour initiate the action necessary to place the unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

B. Vacuum Breaker System

1. Both valves in each of two vacuum breaker systems, including actuating and power circuits, shall be OPERABLE when CONTAINMENT INTEGRITY is required (except as specified in 3.6.B.2 and 3.6.B.3 below).

2. With one vacuum breaker inoperable with respect to its containment isolation function, apply the requirements of Specification 3.6.C.3, to the isolation valves associated with the inoperable vacuum breaker.

3. One vacuum breaker may be inoperable with respect to its vacuum relief function for 7 days.

C. Containment Isolation Valves

1. Non-automatic containment isolation valves shall be locked closed or shall be under direct administrative control and capable of being closed within one minute following an accident when CONTAINMENT INTEGRITY is required (except as specified in 3.6.C.3 below).

2. Automatic containment isolation valves, including actuation circuits, shall be OPERABLE when CONTAINMENT INTEGRITY is required (except as specified in 3.6.C.3 below).

3. With one or more of the containment isolation valve(s) inoperable, within four hours:

   (a) restore the inoperable valve(s) to operable status or,

   (b) deactivate the operable valve in the closed position or,

   (c) lock closed at least one valve in each penetration having one inoperable valve.

2. Initial and periodic type B (except airlocks) and type C tests of penetrations shall be performed at a pressure of 46 psig ($P_a$) in accordance with the provisions of Appendix J, Section III.B and Section III.C, and Specification 4.4.A.5. The airlocks shall be tested initially and at six-month intervals at 46 psig by pressurizing the inner volume. In addition, when CONTAINMENT INTEGRITY is required, each airlock shall be tested every 3 days if it is in use by pressurizing the intergasket space to 10 psig.

3. Type A tests will be considered to be satisfactory if the acceptance criteria delineated in Appendix J, Section III.A are met.

4. Type B and C tests will be considered to be satisfactory if the combined leakage rate of all components subjected to Type B and C tests does not exceed 60% of the $L_a$ and if the following conditions are met.

   a. For pipes connected to systems that are in the auxiliary building special ventilation zone, the total leakage past isolation valves shall be less than 0.1 weight percent per 24 hours at pressure $P_a$.

   b. For pipes connected to systems that are exterior to both the shield building and the auxiliary building special ventilation zone, the total leakage past isolation valves shall be less than 0.01 weight percent per 24 hours at pressure $P_a$.

   c. For airlocks, the leakage shall be less than 1% of the $L_a$ at 10 psig for door intergasket tests and 5% of the $L_a$ at 46 psig for overall airlock tests.

5. The retest schedules for Type A, B, and C tests will be in accordance with Section III.D of Appendix J. Each shield building shall be retested in accordance with the Type A test schedule for its containment. The auxiliary building special ventilation zone shall be retested in accordance with the Type A test schedule for Unit 1 containment.

6. Type A, B and C tests will be in accordance with Section V of Appendix J. Inspection and reporting requirements of each shield building test shall be the same for Type A tests. The auxiliary building special ventilation zone shall have the same inspection and reporting requirements as for the Type A tests of Unit 1.

3.6 CONTAINMENT SYSTEM

Bases

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment.

The opening of normally closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) designation of an operator who is in constant communication with the control room and capable of closing the affected valve(s) within one minute, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

Proper functioning of the Shield Building vent system is essential to the performance of the containment system. Therefore, except for reasonable periods of maintenance outage for one redundant chain of equipment, the system should be wholly in readiness whenever above 200°F. Proper functioning of the auxiliary building special vent system and isolation of the auxiliary building normal vent system are similarly necessary to preclude possible unfiltered leakage through penetrations that enter the special ventilation zone.

The auxiliary building special ventilation zone and its associated ventilation system have been designed to serve as secondary containment following a loss of coolant accident (Reference 2). Special care was taken to design the access doors in the boundary and isolation valves in normal ventilation systems so that AUXILIARY BUILDING SPECIAL VENTILATION ZONE INTEGRITY can be intact during reactor operation. The zone can perform its accident function with openings if they can be closed within 6 minutes, since the accident analysis assumed direct leakage of primary containment atmosphere to the environs when the shield building is at positive pressure (6 minutes). As noted in Reference 2, part of the Shield Building is part of the Auxiliary Building Special Ventilation Zone Integrity. The part of the Shield Building which is part of the Auxiliary Building Special Ventilation Zone is subject to the Technical Specifications of the Shield Building Integrity and not those associated with Auxiliary Building Special Ventilation Zone Integrity.

The action statement which allows Shield Building Integrity to be lost for 24 hours will allow for minor modifications to be made to the Shield Building during power operations.

The COLD SHUTDOWN condition precludes any energy release or buildup of containment pressure from flashing of reactor coolant in the event of a system break.

The shutdown margin for the COLD SHUTDOWN condition assures sub-criticality with the vessel closed, even if the most reactive rod control cluster assembly were inadvertently withdrawn.

3.6 CONTAINMENT SYSTEM

Bases continued

The 2 psig limit on internal pressure provides adequate margin between the maximum internal pressure of 46 psig and the peak accident pressure resulting from the postulated Design Basis Accident (Reference 1).

The containment vessel is designed for 0.8 psi internal vacuum, the occurrence of which will be prevented by redundant vacuum breaker systems.

The containment has a nil ductility transition temperature of 0°F. Specifying a minimum temperature of 30°F will provide adequate margin above NDTT during power operation when containment is required.

The conservative calculation of off-site doses for the loss of coolant accident (References 2, 4) is based on an initial shield building annulus air temperature of 60°F and an initial containment vessel air temperature of 104°F. The calculated period following LOCA for which the shield building annulus pressure is positive, and the calculated off-site doses are sensitive to this initial air temperature difference. The specified 44°F temperature difference is consistent with the LOCA accident analysis (Reference 4).

The initial testing of inleakage into the shield building and the auxiliary building special ventilation zone (ABSVZ) has resulted in greater specified inleakage (Figure TS.4.4-1, change No. 1) and the necessity to deenergize the turbine building exhaust fans in order to achieve a negative pressure in the ABSVZ (TS.3.6.E.2). The staff's conservative calculation of doses for these conditions indicated that changing allowable containment leak rate from 0.5% to 0.25%/day would offset the increased leakage (Reference 3).

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers for all emergency air treatment systems. The Charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment.

The operability of the equipment and systems required for the control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA", March 1971.

Air locks are provided with two doors, each of which is designed to seal against the maximum containment pressure resulting from the limiting DBA. Should an air lock become inoperable as a result of an inoperable air lock door or an inoperable door interlock, power operation may continue provided that at least one OPERABLE air lock door is closed. With an air lock door inoperable, access through the closed or locked OPERABLE door is only permitted for repair of inoperable air lock equipment.

## 3.6 CONTAINMENT SYSTEM

Bases continued

OPERABILITY of air locks is required to ensure that CONTAINMENT INTEGRITY is maintained. Should an air lock become inoperable for reasons other than an inoperable air lock door, the air lock leak tight integrity must be restored within 24 hours or actions must be taken to place the unit in a condition for which CONTAINMENT INTEGRITY is not required.

References

1. USAR, Section 5
2. USAR, Section 10.3.4 and FSAR Appendix G
3. Letter to NSP dated November 29, 1973
4. Letter to NSP dated September 16, 1974

## 4.4 CONTAINMENT SYSTEM TESTS

### Bases

The Containment System consists of a steel containment vessel, a concrete shield building, the Auxiliary Building Special Ventilation Zone (ABSVZ), a Shield Building Ventilation System, and an Auxiliary Building Special Ventilation System. In the event of a loss-of-coolant accident, a vacuum in the shield building annulus will cause most leakage from the containment vessel to be mixed in the annulus volume and recirculated through a filter system before its deferred release to the environment through the exhaust fan that maintains vacuum. Some of the leakage goes to the ABSVZ from which it is exhausted through a filter. A small fraction bypasses both filter systems.

The freestanding containment vessel is designed to accommodate the maximum internal pressure that would result from the Design Basis Accident (Reference 1). For initial conditions typical of normal operation, 120°F and 15 psia, an instantaneous double-ended break with minimum safeguards results in a peak pressure of less than 46 psig at 268°F.

The containment will be strength-tested at 51.8 psig and leak-tested at 46.0 psig to meet acceptance specifications.

License Amendment Nos. 62 and 56 dated February 23, 1983 revised the Prairie Island Technical Specifications to conform to the requirements of Appendix J to 10 CFR Part 50. That License Amendment approved several clarifications and exemptions to the Type B and C testing requirements of Appendix J to 10 CFR Part 50. Those clarifications and exemptions were incorporated into the Prairie Island Technical Specifications in the form of Notes 1, 2 and 5 of Table TS.4.4-1. Table TS.4.4-1 was subsequently relocated from the Prairie Island Technical Specifications in response to Generic Letter 91-08, "Removal of Component Lists From Technical Specifications". While the reference of these notes to specific containment penetrations was relocated out of the Technical Specifications with Table TS.4.4-1, the specific clarifications and exemptions approved by License Amendment Nos. 62 and 56 are still binding. The applicability of the Type B and C testing clarifications and exemptions contained in Notes 1, 2 and 5 of relocated Table TS.4.4-1, to specific containment penetrations, is maintained in the Prairie Island Updated Safety Analysis Report.

The safety analysis (References 2, 3) is based on a conservatively chosen reference set of assumptions regarding the sequence of events relating to activity release and attainment and maintenance of vacuum in the shield building annulus and the Auxiliary Building Special Ventilation Zone, the effectiveness of filtering, and the leak rate of the containment vessel as a function of time. The effects of variation in these assumptions, including that for leak rate, has been investigated thoroughly. A summary of the items of conservatism involved in the reference calculation and the magnitude of their effect upon off-site dose demonstrates the collective effectiveness of conservatism in these assumptions.

## 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, achieved either by normal system operation or hydrostatically testing 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. 100
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 May 7, 1992, as revised June 24, 1993, 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. 100 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 its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Hall for

William M. Dean, Acting 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: July 29, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 100

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 marginal lines indicating the area of change.

| REMOVE | INSERT |
|---|---|
| TS-xii | TS-xii |
| TS.1-2 | TS.1-2 |
| TS.3.4-2 | TS.3.4-2 |
| TS.3.6-1 | TS.3.6-1 |
| TS.4.4-2 | TS.4.4-2 |
| TABLE TS.4.4-1 (Pgs 1-5) | ------ |
| B.3.6-1 | B.3.6-1 |
| B.3.6-2 | B.3.6-2 |
| ----- | B.3.6-3 |
| B.4.4-1 | B.4.4-1 |
| B.4.4-2 | B.4.4-2 |

TECHNICAL SPECIFICATIONS

LIST OF TABLES


| TS TABLE | TITLE |
|---|---|
| 3.5-1 | Engineered Safety Features Initiation Instrument Limiting Set Points |
| 3.5-2 | Instrument Operating Conditions for Reactor Trip |
| 3.5-3 | Instrument Operating Conditions for Emergency Cooling System |
| 3.5-4 | Instrument Operating Conditions for Isolation Functions |
| 3.5-5 | Instrument Operating Conditions for Ventilation Systems |
| 3.5-6 | Instrument Operating Conditions for Auxiliary Electrical System |
| 3.9-1 | Radioactive Liquid Effluent Monitoring Instrumentation |
| 3.9-2 | Radioactive Gaseous Effluent Monitoring instrumentation |
| 3.14-1 | Safety Related Fire Detection Instruments |
| 3.15-1 | Event Monitoring instrumentation - Process & Containment |
| 3.15-2 | Event Monitoring instrumentation - Radiation |
| 4.1-1 | Minimum Frequencies for Checks, Calibrations and Test of Instrument Channels |
| 4.1-2A | Minimum Frequencies for Equipment Tests |
| 4.1-2B | Minimum Frequencies for Sampling Tests |
| 4.2-1 | Special Inservice Inspection Requirements |
| 4.10-1 | Radiation Environmental Monitoring Program (REMP) Sample Collection and Analysis |
| 4.10-2 | RFMP - Maximum Values for the Lower Limits of Detection |
| 4.10-3 | RFMP - Reporting Levels for Radioactivity Concentrations in Environmental Samples |
| 4.12-1 | Steam Generator Tube Inspection |
| 4.13-1 | Snubber Visual Inspection Interval |
| 4.17-1 | Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements |
| 4.17-2 | Radioactive Gaseous Effluent Monitoring instrumentation Surveillance Requirements |
| 4.17-3 | Radioactive Liquid Waste Sampling and Analysis Program |
| 4.17-4 | Radioactive Gaseous Waste Sampling and Analysis Program |
| 5.5-1 | Anticipated Annual Release of Radioactive Material in Liquid Effluents From Prairie Island Nuclear Generating Plant (Per Unit) |
| 5.5-2 | Anticipated Annual Release of Radioactive Nuclides in Gaseous Effluent From Prairie Island Nuclear Generating Plant (Per Unit) |
| 6.1-1 | Minimum Shift Crew Composition |

CONTAINMENT INTEGRITY

CONTAINMENT INTEGRITY shall exist when:

1. Penetrations required to be isolated during accident conditions are either:

    a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

    b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Specifications 3.6.C and 3.6.D.

2. The equipment hatch is closed and sealed.

3. Each air lock is in compliance with the requirements of Specification 3.6.M.

4. The containment leakage rates are within their required limits.

COLD SHUTDOWN

A reactor is in the COLD SHUTDOWN condition when the reactor is subcritical by at least 1% $\Delta k/k$ and the reactor coolant average temperature is less than 200°F.

CORE ALTERATION

CORE ALTERATION is the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel, which may affect core reactivity. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.7.A.6. Plant operation within these operating limits is addressed in individual specifications.

3.4.B.1.d. A minimum of 100,000 gallons of water is available in the condensate storage tanks and a backup supply of river water is available through the cooling water system.

e. Motor operated valves MV-32242 and MV-32243 (Unit 2 valves MV-32248 and MV-32249) shall have valve position monitor lights OPERABLE and shall be locked in the open position by having the motor control center supply breakers physically locked in the off position.

f. Manual valves in the above systems that could (if one is improperly positioned) reduce flow below that assumed for accident analysis shall be locked in the proper position for emergency use. During POWER OPERATION, changes in valve position will be under direct administrative control.

g. The condensate supply cross connect valve C-41-2, to the auxiliary feedwater pumps shall be blocked and tagged open. Any changes in position of this valve shall be under direct administrative control.

2. During STARTUP OPERATION or POWER OPERATION, any one of the following conditions of inoperability may exist for each unit provided STARTUP OPERATION is discontinued until OPERABILITY is restored. If OPERABILITY is not restored within the time specified, place the affected unit (or either unit in the case of a motor driven AFW pump inoperability) in at least HOT SHUTDOWN within the next 6 hours and reduce reactor coolant system average temperature below 350°F within the following 6 hours.

a. A turbine driven AFW pump, system valves and piping may be inoperable for 72 hours.

b. A motor driven AFW pump, system valves and piping may be inoperable for 72 hours.

c. The condensate storage tanks may be inoperable for 48 hours provided the cooling water system is available as a backup supply of water to the auxiliary feedwater pumps.

d. The backup supply of river water provided by the cooling water system may be inoperable for 48 hours provided a minimum of 100,000 gallons of water is available in the condensate storage tanks.

e. The valve position monitor lights for motor operated valves MV-32242 and MV-32243 (Unit 2 valves MV-32248 and MV-32249) may be inoperable for 72 hours provided the associated valves' positions are verified to be open once each shift.

## 3.6 CONTAINMENT SYSTEM

Applicability

Applies to the integrity of the containment system.

Objective

To define the operating status of the containment system for plant operation.

Specification

A. Containment Integrity

1. A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 200°F unless CONTAINMENT INTEGRITY is maintained.

2. If these conditions cannot be satisfied, within one hour initiate the action necessary to place the unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

B. Vacuum Breaker System

1. Both valves in each of two vacuum breaker systems, including actuating and power circuits, shall be OPERABLE when CONTAINMENT INTEGRITY is required (except as specified in 3.6.B.2 and 3.6.B.3 below).

2. With one vacuum breaker inoperable with respect to its containment isolation function, apply the requirements of Specification 3.6.C.3, to the isolation valves associated with the inoperable vacuum breaker.

3. One vacuum breaker may be inoperable with respect to its vacuum relief function for 7 days.

C. Containment Isolation Valves

1. Non-automatic containment isolation valves shall be locked closed or shall be under direct administrative control and capable of being closed within one minute following an accident when CONTAINMENT INTEGRITY is required (except as specified in 3.6.C.3 below).

2. Automatic containment isolation valves, including actuation circuits, shall be OPERABLE when CONTAINMENT INTEGRITY is required (except as specified in 3.6.C.3 below).

3. With one or more of the containment isolation valve(s) inoperable, within four hours:

   (a) restore the inoperable valve(s) to operable status or,

   (b) deactivate the operable valve in the closed position or,

   (c) lock closed at least one valve in each penetration having one inoperable valve.

2. Initial and periodic type B (except airlocks) and type C tests of penetrations shall be performed at a pressure of 46 psig ($P_a$) in accordance with the provisions of Appendix J, Section III.B and Section III.C, and Specification 4.4.A.5. The airlocks shall be tested initially and at six-month intervals at 46 psig by pressurizing the inner volume. In addition, when CONTAINMENT INTEGRITY is required, each airlock shall be tested every 3 days if it is in use by pressurizing the intergasket space to 10 psig.

3. Type A tests will be considered to be satisfactory if the acceptance criteria delineated in Appendix J, Section III.A are met.

4. Type B and C tests will be considered to be satisfactory if the combined leakage rate of all components subjected to Type B and C tests does not exceed 60% of the $L_a$ and if the following conditions are met.

   a. For pipes connected to systems that are in the auxiliary building special ventilation zone, the total leakage past isolation valves shall be less than 0.1 weight percent per 24 hours at pressure $P_a$.

   b. For pipes connected to systems that are exterior to both the shield building and the auxiliary building special ventilation zone, the total leakage past isolation valves shall be less than 0.01 weight percent per 24 hours at pressure $P_a$.

   c. For airlocks, the leakage shall be less than 1% of the $L_a$ at 10 psig for door intergasket tests and 5% of the $L_a$ at 46 psig for overall airlock tests.

5. The retest schedules for Type A, B, and C tests will be in accordance with Section III.D of Appendix J. Each shield building shall be retested in accordance with the Type A test schedule for its containment. The auxiliary building special ventilation zone shall be retested in accordance with the Type A test schedule for Unit 1 containment.

6. Type A, B and C tests will be in accordance with Section V of Appendix J. Inspection and reporting requirements of each shield building test shall be the same for Type A tests. The auxiliary building special ventilation zone shall have the same inspection and reporting requirements as for the Type A tests of Unit 1.

## 3.6 CONTAINMENT SYSTEM

### Bases

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment.

The opening of normally closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) designation of an operator who is in constant communication with the control room and capable of closing the affected valve(s) within one minute, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

Proper functioning of the Shield Building vent system is essential to the performance of the containment system. Therefore, except for reasonable periods of maintenance outage for one redundant chain of equipment, the system should be wholly in readiness whenever above 200°F. Proper functioning of the auxiliary building special vent system and isolation of the auxiliary building normal vent system are similarly necessary to preclude possible unfiltered leakage through penetrations that enter the special ventilation zone.

The auxiliary building special ventilation zone and its associated ventilation system have been designed to serve as secondary containment following a loss of coolant accident (Reference 2). Special care was taken to design the access doors in the boundary and isolation valves in normal ventilation systems so that AUXILIARY BUILDING SPECIAL VENTILATION ZONE INTEGRITY can be intact during reactor operation. The zone can perform its accident function with openings if they can be closed within 6 minutes, since the accident analysis assumed direct leakage of primary containment atmosphere to the environs when the shield building is at positive pressure (6 minutes). As noted in Reference 2, part of the Shield Building is part of the Auxiliary Building Special Ventilation Zone Integrity. The part of the Shield Building which is part of the Auxiliary Building Special Ventilation Zone is subject to the Technical Specifications of the Shield Building Integrity and not those associated with Auxiliary Building Special Ventilation Zone Integrity.

The action statement which allows Shield Building Integrity to be lost for 24 hours will allow for minor modifications to be made to the Shield Building during power operations.

The COLD SHUTDOWN condition precludes any energy release or buildup of containment pressure from flashing of reactor coolant in the event of a system break.

The shutdown margin for the COLD SHUTDOWN condition assures sub-criticality with the vessel closed, even if the most reactive rod control cluster assembly were inadvertently withdrawn.

3.6 CONTAINMENT SYSTEM

Bases continued

The 2 psig limit on internal pressure provides adequate margin between the maximum internal pressure of 46 psig and the peak accident pressure resulting from the postulated Design Basis Accident (Reference 1).

The containment vessel is designed for 0.8 psi internal vacuum, the occurrence of which will be prevented by redundant vacuum breaker systems.

The containment has a nil ductility transition temperature of 0°F. Specifying a minimum temperature of 30°F will provide adequate margin above NDTT during power operation when containment is required.

The conservative calculation of off-site doses for the loss of coolant accident (References 2, 4) is based on an initial shield building annulus air temperature of 60°F and an initial containment vessel air temperature of 104°F. The calculated period following LOCA for which the shield building annulus pressure is positive, and the calculated off-site doses are sensitive to this initial air temperature difference. The specified 44°F temperature difference is consistent with the LOCA accident analysis (Reference 4).

The initial testing of inleakage into the shield building and the auxiliary building special ventilation zone (ABSVZ) has resulted in greater specified inleakage (Figure TS.4.4-1, change No. 1) and the necessity to deenergize the turbine building exhaust fans in order to achieve a negative pressure in the ABSVZ (TS.3.6.E.2). The staff's conservative calculation of doses for these conditions indicated that changing allowable containment leak rate from 0.5% to 0.25%/day would offset the increased leakage (Reference 3).

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers for all emergency air treatment systems. The Charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment.

The operability of the equipment and systems required for the control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA", March 1971.

Air locks are provided with two doors, each of which is designed to seal against the maximum containment pressure resulting from the limiting DBA. Should an air lock become inoperable as a result of an inoperable air lock door or an inoperable door interlock, power operation may continue provided that at least one OPERABLE air lock door is closed. With an air lock door inoperable, access through the closed or locked OPERABLE door is only permitted for repair of inoperable air lock equipment.

## 3.6 CONTAINMENT SYSTEM

Bases continued

OPERABILITY of air locks is required to ensure that CONTAINMENT INTEGRITY is maintained. Should an air lock become inoperable for reasons other than an inoperable air lock door, the air lock leak tight integrity must be restored within 24 hours or actions must be taken to place the unit in a condition for which CONTAINMENT INTEGRITY is not required.

References

1. USAR, Section 5
2. USAR, Section 10.3.4 and FSAR Appendix G
3. Letter to NSP dated November 29, 1973
4. Letter to NSP dated September 16, 1974

## 4.4 CONTAINMENT SYSTEM TESTS

### Bases

The Containment System consists of a steel containment vessel, a concrete shield building, the Auxiliary Building Special Ventilation Zone (ABSVZ), a Shield Building Ventilation System, and an Auxiliary Building Special Ventilation System. In the event of a loss-of-coolant accident, a vacuum in the shield building annulus will cause most leakage from the containment vessel to be mixed in the annulus volume and recirculated through a filter system before its deferred release to the environment through the exhaust fan that maintains vacuum. Some of the leakage goes to the ABSVZ from which it is exhausted through a filter. A small fraction bypasses both filter systems.

The freestanding containment vessel is designed to accommodate the maximum internal pressure that would result from the Design Basis Accident (Reference 1). For initial conditions typical of normal operation, 120°F and 15 psia, an instantaneous double-ended break with minimum safeguards results in a peak pressure of less than 46 psig at 268°F.

The containment will be strength-tested at 51.8 psig and leak-tested at 46.0 psig to meet acceptance specifications.

License Amendment Nos. 62 and 56 dated February 23, 1983 revised the Prairie Island Technical Specifications to conform to the requirements of Appendix J to 10 CFR Part 50. That License Amendment approved several clarifications and exemptions to the Type B and C testing requirements of Appendix J to 10 CFR Part 50. Those clarifications and exemptions were incorporated into the Prairie Island Technical Specifications in the form of Notes 1, 2 and 5 of Table TS.4.4-1. Table TS.4.4-1 was subsequently relocated from the Prairie Island Technical Specifications in response to Generic Letter 91-08, "Removal of Component Lists From Technical Specifications". While the reference of these notes to specific containment penetrations was relocated out of the Technical Specifications with Table TS.4.4-1, the specific clarifications and exemptions approved by License Amendment Nos. 62 and 56 are still binding. The applicability of the Type B and C testing clarifications and exemptions contained in Notes 1, 2 and 5 of relocated Table TS.4.4-1, to specific containment penetrations, is maintained in the Prairie Island Updated Safety Analysis Report.

The safety analysis (References 2, 3) is based on a conservatively chosen reference set of assumptions regarding the sequence of events relating to activity release and attainment and maintenance of vacuum in the shield building annulus and the Auxiliary Building Special Ventilation Zone, the effectiveness of filtering, and the leak rate of the containment vessel as a function of time. The effects of variation in these assumptions, including that for leak rate, has been investigated thoroughly. A summary of the items of conservatism involved in the reference calculation and the magnitude of their effect upon off-site dose demonstrates the collective effectiveness of conservatism in these assumptions.

## 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, achieved either by normal system operation or hydrostatically testing 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. 107 AND 100 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 May 7, 1992, as revised June 24, 1993, 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 supplemental letter dated June 24, 1993, provided clarifying information that did not change the initial proposed no significant hazards determination published in the Federal Register (57 FR 24674). The proposed amendments would (1) relocate the Containment Penetration List from Section 4.4 of the Technical Specifications into plant procedures in accordance with the guidance of Generic Letter 91-08, (2) change Section 3.6.C of the Technical Specifications to clarify when non-automatic containment isolation valves are required to be operable and what actions are to be taken in response to inoperability of a non-automatic containment isolation valve, and (3) deletes condensate cross-connect valve C-41-1 from Section 3.4.B.1.g of the Technical Specifications.

## 2.0 EVALUATION

### 2.1 RELOCATION OF CONTAINMENT PENETRATION LIST

The proposed changes to the Prairie Island Technical Specifications being proposed in response to Generic Letter 91-08 are described below.

A. The reference to Table TS.4.4-1, "Unit 1 and Unit 2 Penetration Designation for Leakage Tests," in the Table of Contents "List of Tables" would be deleted consistent with deletion of Table 4.4-1 from Section 4 of the Technical Specifications.

B. Item 2 of the definition of "Containment Integrity" in Section 1.0 would be deleted. This statement references usage of blind flanges as required by Table 4.4-1. Table 4.4-1 is proposed to be deleted as indicated below. A separate statement in Section 1.0 for the installation of blind flanges, is unnecessary as it is redundant to another existing requirement in Item 1.b of the Containment Integrity definition which states that all penetrations are either closed by manual valves, blind flanges or deactivated automatic valves.

C. References to Table TS.4.4-1 would be deleted from Sections 3.6.C.2 and 3.6.C.3 for consistency with the deletion of the table from Section 4.

D. References to Table TS.4.4-1 would be deleted from Sections 4.4.A.2, 4.4.A.4.a and 4.4.A.4.b reflecting the deletion of the table from Section 4. The term "containment system integrity" would be changed to "CONTAINMENT INTEGRITY" in Section 4.4.A.2 to be consistent with the current terminology in Section 1.0 and the policy for capitalizing all defined terms. The acronym "ABSVZ" (Auxiliary Building Special Ventilation Zone) would be spelled out in Sections 4.4.A.4.a and 4.4.A.4.b for clarity and consistency with Sections 4.4.A.5 and 4.4.A.6.

E. In Section 4 of the Technical Specifications Table TS.4.4-1, "Unit 1 and Unit 2 Penetration Designation for Leakage Tests," would be relocated into plant procedures that are subject to the change control provisions for plant procedures in the Administrative Controls Section of the Prairie Island Technical Specifications. Notes that are part of Table 4.4-1 would be affected as follows:

   1. License Amendment No. 62, dated February 23, 1983 revised the Prairie Island Technical Specifications to conform to the requirements of Appendix J to 10 CFR Part 50. Notes 1, 2 and 5 of Table TS.4.4-1 were incorporated into the Prairie Island Technical Specifications by License Amendment No. 62 to provide clarifications and exemptions to the Type B and C testing requirements of Appendix J to 10 CFR Part 50. Discussion of notes 1, 2 and 5 of Table TS.4.4-1 would be incorporated into Bases Section 4.4 so that the applicability of the related Appendix J testing requirements remains clearly defined. While the reference of these notes to specific containment penetrations would be relocated out of the Technical Specifications with Table TS.4.4-1, the specific clarifications and exemptions incorporated into Table TS.4.4-1 by License Amendment 62 would remain in effect.

   2. Note 3 of Table TS.4.4-1, which defines terms utilized in Table TS.4.4-1, would not be retained because it is an integral part of the Table and serves no useful purpose in the Technical Specifications once the table is relocated.

   3. Note 4 of Table TS.4.4-1, which describes which penetrations have blank flanges, would not be retained in the Technical Specifications because of its reference to specific penetration numbers. The information provided by Note 4 will be relocated with Table TS.4.4-1 to the plant procedures and the Prairie Island Updated Safety Analysis Report.

   4. Note 6 of Table TS.4.4-1 would be deleted, on the basis that it duplicates information which is also provided by Section 3.6.D.2.b of the Technical Specifications.

The staff has reviewed the proposed changes and verified that the changes are consistent with Generic Letter 91-08 and do not unintentionally delete or modify any Limiting Conditions of Operation or Surveillance Requirements using the guidance of the Generic Letter. As stated in the Generic Letter, plant administrative procedures provide adequate change control provisions and the operability requirements of the Technical Specifications will remain effective and enforceable. The proposed changes relating to relocation of the List of Penetrations are, therefore, acceptable.

2.2 NON-AUTOMATIC CONTAINMENT ISOLATION VALVES

This license amendment application requests changes to Prairie Island Technical Specification Section 3.6.C which will clarify when the non-automatic containment isolation valves are required to be operable and what actions are to be taken in response to the inoperability of a non-automatic containment isolation valve.

The existing wording in Technical Specification Section 3.6.C.1 does not specify when non-automatic containment isolation valves are required to be operable and does not specifically refer to the containment isolation valve action statements in Section 3.6.C.3. It is not clear from reading the existing wording in Sections 3.6.C.1 and 3.6.C.3 that the action statements in Section 3.6.C.3 are applicable to the non-automatic containment isolation valves. To correct this, Section 3.6.C.1 would be revised to (a) specify that non-automatic containment isolation valves be operable whenever containment integrity is required and (b) refer to the action statements in Section 3.6.C.3. The proposed changes would make it clear that the specified action statements apply to all containment isolation valves, both automatic and non-automatic.

With the exception of automatic instrumentation operability and testing requirements, non-automatic containment isolation valves are subject to operability and surveillance testing requirements similar to those for automatic isolation valves. Accordingly, the proposed changes are acceptable.

2.3 DELETION OF CONDENSATE SUPPLY CROSS-CONNECT VALVE

The two Prairie Island units share three condensate storage tanks which are cross-connected with a suction header serving the four Auxiliary Feed Water Pumps. Technical Specification 3.4.B.1.g currently specifies that condensate cross connect valves C-41-1 and C-41-2 be blocked and tagged open. A reliability study of the Prairie Island auxiliary feedwater system was completed in April 1986 (Ref: NSPNAD-8606P). That reliability study concluded that the reliability of the auxiliary feedwater system could be improved if valve C-41-1 was removed from the condensate supply to the auxiliary feedwater pumps and replaced with a spool piece. Valve C-41-1 was subsequently removed and replaced with a spool piece. However, due to an oversight, the valve was removed and replaced with a spool piece before it was removed from the Technical Specifications. Valve C-41-1 was originally included in the Technical Specifications to protect against inadvertent closure of the valve which would adversely affect the condensate supply to the auxiliary feedwater pumps. When the licensee identified that the valve had been removed without modifying the Technical Specifications, it was concluded that the spool piece performed the same function as a blocked and tagged open valve and that the use of the spool piece met the intent of Technical Specification 3.4.B.1.g. The replacement of valve C-41-1 with a spool piece reduces the possibility of a human or administrative error that could adversely affect the reliability of the Auxiliary Feed Water System condensate supply. Remaining valves provide capability to isolate the condensate storage tanks when necessary. The proposed amendment is, therefore, acceptable.

## 3.0 STATE CONSULTATION

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

## 4.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding (57 FR 24674, June 10, 1992). Accordingly, this amendment meets 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 this amendment.

## 5.0 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: W. Long

Date: July 29, 1993