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U.S. NUCLEAR REGULATORY COMMISSION

Docket Nos. 50-282
and 50-306

Mr. L. O. Mayer, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

Dear Mr. Mayer:

The Commission has issued the enclosed Amendment No. 46 to Facility Operating License No. DPR-42 and Amendment No. 40 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 in response to your application dated December 19, 1980.

The amendment incorporates new requirements in the form of license conditions, limiting conditions for operation, and surveillance requirements for instruments and equipment as well as shift manning requirements resulting from the NRC's assessment of the accident at Three Mile Island Unit 2.

By our letter of September 13, 1979, we issued new requirements to all operating nuclear power plants established as a result of our review of the TMI-2 accident. Certain of these requirements, designated Lessons Learned Category "A" requirements, were to have been completed prior to any operation subsequent to January 1, 1980. Our evaluation and acceptance of your actions to comply with these Category "A" items was contained in our letter to you of April 18, 1980.

In order to provide reasonable assurance that operating reactor facilities are maintained within the limits determined acceptable following the implementation of these TMI-2 Lessons Learned Category "A" items, we requested that licensees include certain of these items in the operating license as license conditions and additional Technical Specifications. These requirements were contained in our letter to you of July 2, 1980, which contained model requirements that we had determined to be acceptable.

Your request for amendments dated December 19, 1980, is responsive to this request. Certain changes have been made to conform to our requirements. These have been discussed with, and concurred in by, members of your staff. The issuance of this amendment acceptably resolves our requirements for license conditions and Technical Specifications for TMI-2 Lessons Learned Category "A" items for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2. Our acceptance of these new requirements is documented in

OFFICE							
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MAR 02 1981

our evaluation letter of April 18, 1980 and our letter to you of July 2, 1980, which, together with this letter, constitute our Safety Evaluation of this matter.

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendments.

We have concluded that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

A copy of the Notice of Issuance is also enclosed.

Sincerely,

Original signed by
Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 46 to DPR-42
2. Amendment No. 40 to DPR-60
3. Notice of Issuance

cc: w/enclosures
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Docket No. 50-282 and 50-306

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Office of the Secretary of the Commission

SUBJECT: NORTHERN STATES POWER COMPANY, PRAIRIE ISLAND NUCLEAR GENERATING PLATN, UNIT NOS. 1 AND 2

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (12) of the Notice are enclosed for your use.

☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).

☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.

☐ Notice of Availability of Applicant's Environmental Report.

☐ Notice of Proposed Issuance of Amendment to Facility Operating License.

☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.

☐ Notice of Availability of NRC Draft/Final Environmental Statement.

☐ Notice of Limited Work Authorization.

☐ Notice of Availability of Safety Evaluation Report.

☐ Notice of Issuance of Construction Permit(s).

☐ Notice of Issuance of Facility Operating License(s) or Amendment(s).

XX Other: Amendments Nos. 46 and 40.
Referenced documents provided PDR.

Division of Licensing, ORB#3
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

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NRC FORM 102 (1-76)

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

March 2, 1981

Docket Nos. 50-282
and 50-306

Mr. L. O. Mayer, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

Dear Mr. Mayer:

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The amendment incorporates new requirements in the form of license conditions, limiting conditions for operation, and surveillance requirements for instruments and equipment as well as shift manning requirements resulting from the NRC's assessment of the accident at Three Mile Island Unit 2.

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We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendments.

We have concluded that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

A copy of the Notice of Issuance is also enclosed.

Sincerely,

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 46 to DPR-42
2. Amendment No. 40 to DPR-60
3. Notice of Issuance

cc: w/enclosures
See next page

Northern States Power Company

cc:

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U. S. Environmental Protection Agency
Federal Activities Branch
Region V Office
ATTN: EIS COORDINATOR
230 South Dearborn Street
Chicago, Illinois 60604

cc w/enclosure(s) and incoming
dated: December 19, 1980

Chairman, Public Service Commission
of Wisconsin
Hill Farms State Office Building
Madison Wisconsin 53702

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 46
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 December 19, 1980, 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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 46 , 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

R. A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 2, 1981

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. 40
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 December 19, 1980, 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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 40 , 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

R. A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 2, 1981

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 46 TO FACILITY OPERATING LICENSE NO. DPR-42

AMENDMENT NO. 40 TO FACILITY OPERATING LICENSE NO. DPR-60

DOCKET NOS. 50-282 AND 50-306

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by Amendment Number and contain vertical lines indicating the area of change.

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TS-iii
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TS.3.1-3
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TS.3.4-1
TS.3.4-2
TS.3.4-3
TS.3.5-3
TS.3.5-4
TS.3.5-5 (new)
TABLE TS.3.5-1
TABLE TS.3.5-3 (Page 1 of 2)
TABLE TS.3.5-3 (Page 2 of 2) (new)
TABLE TS.3.5-4
TS.3.15-1 (new)
TABLE TS.3.15.1 (new)
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TS.4.6-1A
TS.4.6-3 (new)
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APPENDIX A TECHNICAL SPECIFICATIONS

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APPENDIX A TECHNICAL SPECIFICATIONS

LIST OF TABLES

3. Pressurizer

a. Whenever average reactor coolant system temperature is above 350°F or the reactor is critical, the pressurizer shall be operable with:

1. Steam bubble

2. Pressurizer heater groups "A" and "B" and their associated safeguards power supplies operable

3. At least one operable spray

b. With the pressurizer inoperable due to an inoperable heater group restore the equipment to operable status within 72 hours or place the reactor in at least Hot Shutdown within the following 18 hours.

c. With the pressurizer inoperable for any other reason than (b) above, the reactor shall be placed in at least Hot Shutdown within the following 12 hours.

d. At least one pressurizer safety valve shall be operable whenever the head is on the reactor vessel, except during hydrostatic tests. Both pressurizer safety valves shall be operable whenever average reactor coolant system temperature is above 350°F or the reactor is critical. Pressurizer safety valve lift setting shall be 2485 psig ± 1%.

e. Except as specified in (f) and (g) below, two power operated relief valves (PORV's) and their associated block valves shall be operable whenever average reactor coolant system temperature is above 350°F or the reactor is critical.

f. With one or more PORV's inoperable, within one hour either restore the PORV(s) to operable status or close the associated block valve(s). If this cannot be done, place the reactor in the Cold Shutdown condition within the following 36 hours.

g. With one or more block valves inoperable, within one hour either restore the block valve(s) to operable status or close the valve. If this cannot be done, place the reactor in the Cold Shutdown condition within the following 36 hours.

Basis

When the boron concentration of the reactor coolant system is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the equivalent of the primary system volume in approximately one-half hour.

"Steam Generator Tube Surveillance", Technical Specification 4.12, identifies steam generator tube imperfections having a depth $\geq$50% of the 0.050-inch tube wall thickness as being unacceptable for power operation. The results of steam generator burst and tube collapse tests submitted to the staff have demonstrated that tubes having a wall thickness greater than 0.025-inch have adequate margins of safety against failure due to loads imposed by normal plant operation and design basis accidents.[2]

Part A of the specification requires that both reactor coolant pumps be operating when the reactor is critical to provide core cooling in the event that a loss of flow occurs. In the event of the worst credible coolant flow loss (loss of both pumps from 100% power) the minimum calculated DNBR remains well above 1.30. Therefore, cladding damage and release of fission products to the reactor coolant will not occur. Critical operation, except for low power physics tests, with less than two pumps is not planned. Above 10% power, an automatic reactor trip will occur if flow from either pump is lost. Below 10% power, a shutdown under administrative control will be made if flow from either pump is lost.

The pressurizer is needed to maintain acceptable system pressure during normal plant operation, including surges that may result following anticipated transients. Each of the pressurizer safety valves is designed to relieve 325,000 lbs per hour of saturated steam at the valve set point. Below 350°F and 450 psig in the reactor coolant system, the residual heat removal system can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve therefore provides adequate defense against over-pressurization of the reactor coolant system for reactor coolant temperatures less than 350°F. The combined.capacity of both safety valves is greater than the maiximum surge rate resulting from complete loss of load.[1]

Basis (continued)

The requirement that two groups of pressurizer heaters be operable provides assurance that at least one group will be available during a loss of offsite power to maintain natural circulation. Backup heater group "A" is normally supplied by one safeguards bus. Backup heater group "B" can be manually transferred within minutes to the redundant safeguards bus. Tests have confirmed the ability of either group to maintain natural circulation conditions.

The pressurizer power operated relief valves (PORV's) operate to relieve reactor coolant system pressure below the setting of the pressurizer Code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The PORV's are pneumatic valves operated by instrument air. They fail closed on loss of air or loss of power to their DC solenoid valves. The PORV block valves are motor operated valves supplied by the 480 volt safeguards buses.

References

[1]FSAR, Section 14.1.9

[2]Testimony by J Knight in the Prairie Island Public Hearing on January 28, 1975.

3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the operating status of the steam and power conversion system.

Objective

To specify minimum conditions of steam-relieving capacity and auxiliary feed-water supply necessary to assure the capability of removing decay heat from the reactor, and to limit the concentration of activity that might be released by steam relief to the atmosphere.

Specification

A. A reactor shall not be heated above 350°F unless the following conditions are satisfied:

1. Safety and Relief Valves

   a. Rated relief capacity of ten steam system safety valves is available for that reactor, except during testing.

   b. Both steam generator power-operated relief valves for that reactor are operable.

2. Auxiliary Feed System

   a. For single unit operation, the turbine-driven pump associated with that reactor plus one motor-driven pump are operable.

   b. For two-unit operation, all four auxiliary feedwater pumps are operable.

   c. Valves and piping associated with the above components are operable except that during Startup Operation necessary changes may be made in motor-operated valve position. All such changes shall be under direct administrative control.

   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. For Unit 1 operation motor operated valves MV32242 and MV32243 shall have valve position monitor lights operable and shall be locked in the open position by having the motor control center supply breakers manually locked open. For Unit 2, corresponding valve conditions shall exist.

f. Essential features including system piping, valves, and interlocks directly associated with the above components are operable.

g. 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.

3. Steam Exclusion System

Both isolation dampers in each ventilation duct that penetrates rooms containing equipment required for a high energy line rupture outside of containment shall be operable or at least one damper in each duct shall be closed.

4. Radiochemistry

The iodine-131 activity of the water on the secondary side of either steam generator for that reactor does not exceed 0.30 uCi/cc.

B. If, during startup operation or power operation, any of the conditions of Specification 3.4.A., except as noted below for 2.a or 2.b cannot be met, startup operations shall be discontinued and if operability cannot be restored within 48 hours, the affected reactor shall be placed in the cold shutdown condition using normal operating procedures.

With regard to Specifications 2a or 2b, if a turbine driven AFW pump is not operable, that AFW pump shall be restored to operable status within 72 hours or the affected reactor shall be cooled to less than 350°F within the next 12 hours. If a motor driven AFW pump is not operable, that AFW pump shall be restored to operable status within 72 hours or one unit shall be cooled to less than 350°F within the next 12 hours.

Basis

A reactor shutdown from power requires removal of decay heat. Decay heat removal requirements are normally satisfied by the steam bypass to the condenser and by continued feedwater flow to the steam generators. Normal feedwater flow to the steam generators is provided by operation of the turbine-cycle feedwater system.

The ten main steam safety valves have a total combined rated capability of 7,745,000 lbs/hr. The total full power steam flow is 7,094,000 lbs/hr; therefore, the ten main steam safety valves will be able to relieve the total steam flow if necessary.(1)

In the unlikely event of complete loss of offsite electrical power to either or both reactors, continued removal of decay heat would be assured by availability of either the steam-driven auxiliary feedwater pump or the motor-driven auxiliary feedwater pump associated with each reactor, and by steam discharge to the atmosphere through the main steam safety valves. One auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from one reactor. The motor-driven auxiliary feedwater pump for each reactor can be made available to the other reactor.

The minimum amount of water specified for the condensate storage tanks is sufficient to remove the decay heat generated by one reactor in the first 24 hours of shutdown. Essentially unlimited replenishment of the condensate storage supply is available from the intake structures through the cooling water system.

The two power-operated relief valves located upstream of the main steam isolation valves are required to remove decay heat and cool the reactor down following a high energy line rupture outside containment (2). Isolation dampers are required in ventilation ducts that penetrate those rooms containing equipment needed for the accident.

The secondary coolant activity is based on a postulated release of the contents of one steam generator to the atmosphere.(3) This could happen, for example, as a result of a steam break accident combined with failure of a steam line isolation valve. The limiting dose for this case results from iodine-131 because of its low MPC, and because its long half-life relative to the other iodine isotopes results in its greater concentration in leakage fluid. The accident is assumed to occur at zero load when the steam generators contain maximum water. With allowance for plate-out retention of iodine in water droplets, one-tenth of the contained iodine is assumed to reach the site boundary. The maximum inhalation dose at the site boundary is then as follows:

$$\text{Dose (rem)} = \frac{C \cdot V}{10} \cdot B(t) \cdot X/Q \cdot DCF$$

Where: C = secondary coolant activity, 0.30 uCi/cc

V = water volume in one steam generator = 3510 ft^3 = 99 M^3

$B(t)$ = breathing rate, 3.47×10^{-4} M^3 /sec

$X/Q = 9.8 \times 10^{-4}$ sec/m^3

$DCF = 1.50 \times 10^{6}$ rem/Ci I^{131} inhaled

The resulting dose is 1.5 rem.

References

(1) FSAR, Section 10.4
(2) FSAR, Appendix I
(3) FSAR, Section 14

TABLE TS.3.5-1

ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT LIMITING SET POINTS

	FUNCTIONAL UNIT	CHANNEL	LIMITING SET POINTS*
1	High Containment Pressure (Hi)	Safety Injection*	$\leq$4 psig
2	High Containment Pressure (Hi-Hi)	a. Containment Spray	$\leq$23 psig
		b. Steam Line Isolation of Both Lines	$\leq$17 psig
3	Pressurizer Low Pressure	Safety Injection*	$\geq$1815 psig
4	Low Steam Line Pressure	Safety Injection* Lead Time Constant Lag Time Constant	$\geq$500 psig $\geq$12 seconds $\leq$2 seconds
5	High Steam Flow in a Steam Line Coincident with Safety Injection and Low T_{avg}	Steam Line Isolation of Affected Line	d/p corresponding to $\leq 0.745 \times 10^6$ lb/hr at 1005 psig $\geq 540^\circ F$
6	High-high Steam Flow in a Steam Line Coincident with Safety Injection	Steam Line Isolation of Affected Line	$\leq$d/p corresponding to 4.5×10^6 lb/hr at 735 psig
7	High Pressure Difference Between Shield Building and Containment	Containment Vacuum Breakers	$\leq$0.5 psi
8	High Temperature in Ventilation Ducts	Ventilation System Isolation Dampers	$\leq 120^\circ F$
9	High Radiation in Containment Exhaust Air	Containment Ventilation Isolation	$\leq$count rate corresponding to 500 mrem/year whole body and 3000 mrem/year skin due to noble gases at the site boundary

*Initiates also containment isolation, feedwater line isolation and starting of all containment fans.
d/p means differential pressure

Steam Line Isolation

In the event of a steam line break, the steam line stop valve of the affected line is automatically isolated to prevent continuous, uncontrolled steam release from more than one steam generator. The steam lines are isolated on high containment pressure (Hi-Hi) or high steam line flow in coincidence with low T_{avg} and safety injection or high steam flow (Hi-Hi) in coincidence with safety injection. Adequate protection is afforded for breaks inside or outside the containment even when it is assumed that the steam line check valves do not function properly.

Containment Ventilation Isolation

Valves in the containment purge and inservice purge systems automatically close on receipt of a Safety Injection signal or a high radiation signal. Gaseous and particulate monitors in the exhaust stream or a gaseous monitor in the exhaust stack provide the high radiation signal.

Ventilation System Isolation

In the event of a high energy line rupture outside of containment,[(4)] redundant isolation dampers in certain ventilation ducts are closed.

Auxiliary Feedwater System Actuation

The following signals automatically start the pumps and open the steam admission control valve to the turbine driven pump of the affected unit:

1. Low-low water level in either steam generator
2. Trip of both main feedwater pumps
3. Safety Injection signal
4. Undervoltage on both 4.16 KV normal buses (turbine driven pump only)

Manual control from both the control room and the Hot Shutdown Panel are also available. The design provides assurance that water can be supplied to the steam generators for decay heat removal when the normal feedwater system is not available.

Limiting Instrument Setpoints

1. The high containment pressure limit is set at about 10% of the maximum internal pressure. Initiation of Safety Injection protects against loss of coolant [2] or steam line break accidents as discussed in the safety analysis.

2. The Hi-Hi containment pressure limit is set at about 50% of the maximum internal pressure for initiation of containment spray and at about 30% for initiation of steam line isolation. Initiation of Containment Spray and Steam Line Isolation protects against large loss of coolant [2] or steam line break accidents [3] as discussed in the safety analysis.

3. The pressurizer low pressure limit is set substantially below system operating pressure limits. However, it is sufficiently high to protect against a loss of coolant accident as shown in the safety analysis.[2]

4. The steam line low pressure signal is lead/lag compensated and its setpoint is set well above the pressure expected in the event of a large steam line break accident as shown in the safety analysis.[3]

5. The high steam line flow limit is set at approximately 20% of nominal full-load flow at the no-load pressure and the high-high steam line flow limit is set at approximately 120% of nominal full-load flow at the full load pressure in order to protect against large steam break accidents. The coincident low T_{avg} setting limit for steam line isolation initiation is set below its hot shutdown value. The safety analysis shows that these settings provide protection in the event of a large steam break.[3]

6. Steam generator low-low water level and 4.16 KV Bus 11 and 12 (21 and 22 in Unit 2) low bus voltage provide initiation signals for the Auxiliary Feedwater System. Selection of these setpoints is discussed in Section 2.3 of the Technical Specifications.

7. High radiation signals providing input to the Containment Ventilation Isolation circuity are set in accordance with the Radioactive Effluent Technical Specifications. The setpoints are established to prevent exceeding the limits of 10 CFR Part 20 at the site boundary.

Instrument Operating Conditions

During plant operations, the complete instrumentation systems will normally be in service. Reactor safety is provided by the Reactor Protection System, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the plant design. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the Reactor Control and Protection System when any one or more of the channels is out of service.

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel on process control equipment is accomplished by placing that channel bistable in a tripped mode; e.g., a two-out-of-three circuit becomes a one-out-of-two circuit. The source and intermediate range nuclear instrumentation system channels are not intentionally placed in a tripped mode since these are one-out-of-two trips, and the trips are therefore bypassed during testing. Testing does not trip the system unless a trip condition exists in a concurrent channel.

References

(1) FSAR - Section 7.5

(2) FSAR - Section 14.3

(3) FSAR - Seciton 14.2.5

(4) FSAR - Appendix I

TABLE TS.3.5-3

INSTRUMENT OPERATING CONDITIONS FOR EMERGENCY COOLING SYSTEM

FUNCTIONAL UNIT	1 MINIMUM OPERABLE CHANNELS	2 MINIMUM DEGREE OF REDUNDANCY	3 PERMISSIBLE BYPASS CONDITIONS	4 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 or 2 CANNOT BE MET
1. SAFETY INJECTION				
a. Manual	2	1		Hot shutdown **
b. High Containment Pressure	2	1		Hot shutdown **
c. Steam Generator Low Steam Pressure/Loop	2	1	primary pressure less than 2000 psig	Hot shutdown **
d. Pressurizer Low Pressure	2	1	primary pressure less than 2000 psig	Hot shutdown **
2. CONTAINMENT SPRAY				
a. Manual	2	--*		Hot shutdown **
b. Hi-Hi Containment Pressure (Containment Spray)				Hot shutdown **
Channel a	2	1		
Channel b	2	1		
Channel c	2	1		
Logic	2	1		

TABLE TS.3.5-3 (continued)

INSTRUMENT OPERATING CONDITIONS FOR EMERGENCY COOLING SYSTEMS

FUNCTIONAL UNIT	1 MINIMUM OPERATING CHANNELS	2 MINIMUM DEGREE OF REDUNDANCY	3 PERMISSIBLE BYPASS CONDITIONS	4 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 CANNOT BE MET
3. AUXILIARY FEEDWATER				
a. Steam Generator Low-Low Water Level	2	1		Hot shutdown
b. Undervoltage on 4.16 KV Buses 11 and 12 (21 and 22 Unit 2) (Start Turbine Driven Pump only)	2/bus	1/bus		Hot shutdown
c. Trip of Main Feedwater Pumps	2/pump	1/pump		Hot shutdown
d. Safety Injection	(See Item No. 1)			Hot shutdown
e. Manual	2	1		Hot shutdown

* - Must actuate two switches simultaneously.

** - If minimum conditions are not met within 24 hours, steps shall be taken on the affected unit to place the unit in cold shutdown conditions.

TABLE TS.3.5-4

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

FUNCTIONAL UNIT	1 MINIMUM OPERABLE CHANNELS	2 MINIMUM DEGREE OF REDUNDANCY	3 PERMISSIBLE BYPASS CONDITIONS	4 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 CANNOT BE MET
1. CONTAINMENT ISOLATION				
a. Safety Injection	(See Item No. 1 of Table TS.3.5-3)			Hot shutdown**
b. Manual	2	1		Hot shutdown
2. CONTAINMENT VENTILATION ISOLATION				
a. Safety Injection	(See Item No. 1 of Table TS.3.5-3)			Maintain Purge and Inservice Purge Valves closed if conditions of (a), (b), or (c) cannot be met
b. High Radiation in Exhaust Air	2	1		
c. Manual	2	1		
3. STEAM LINE ISOLATION				
a. Hi-Hi Steam Flow with Safety Injection	2	1		Hot shutdown**
b. Hi Steam Flow and 2 of 4 Low Tavg with Safety Injection	2	1		Hot shutdown**
c. Hi Containment Pressure	1/loop	1		Hot shutdown**
d. Manual	1/loop	-		Hot shutdown**
4. EMERGENCY COOLDOWN EQUIPMENT ROOM ISOLATION				
a. High temperature in ventilation system ducts	2	1		Hot shutdown**

** - If minimum conditions are not met within 24 hours, steps shall be taken on the affected unit to place the unit in cold shutdown conditions.

3.15 EVENT MONITORING INSTRUMENTATION

Applicability

Applies to plant instrumentation which does not perform a protective function, but which provides information to monitor and assess important parameters during the following an accident.

Objectives

To ensure that sufficient information is available to operators to determine the effects of and determine the course of an accident to the extent required to carry out required manual actions.

Specification

A. The event monitoring instrumentation channels specified in Table TS.3.15-1 shall be Operable.

B. With the number of Operable event monitoring instrumentation channels less than the Required Total Number of Channels shown on Table TS.3.15-1, either restore the inoperable channels to Operable status within seven days, or be in at least Hot Shutdown within the next 12 hours.

C. With the number of Operable event monitoring instrumentation channels less than the Minimum Channels Operable requirements of Table TS.3.15-1, either restore the minimum number of channels to Operable status within 48 hours or be in at least Hot Shutdown within the next 12 hours.

Basis

The operability of the event monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations."

TABLE TS.3.15-1
EVENT MONITORING INSTRUMENTATION

Instrument	Required Total No. of Channels	Minimum Channels Operable
1. Pressurizer Water Level	2	1
2. Auxiliary Feedwater Flow to Steam Generators (One Channel Flow and One Channel Wide Range Level for Each Steam Generator)	2/steam gen	1/steam gen
3. Reactor Coolant System Subcooling Margin***	2	1
4. Pressurizer Power Operated Relief Valve Position (One Common Channel Temperature, One Channel Limit Switch per Valve, and One Channel Acoustic Sensor per Valve*)	2/valve	1/valve
5. Pressurizer Power Operated Relief Block Valve Position (One Common Channel Temperature, One Channel Limit Switch per Valve, and One Channel Acoustic Sensor per Valve*)	2/valve	1/valve
6. Pressurizer Safety Valve Position (One Channel Temperature per Valve and Common Acoustic Sensor**)	2/valve	1/valve

* - A common acoustic sensor provides backup position indication for each pressurizer power operated relief valve and its associated block valve.

** - The acoustic sensor channel is common to both valves. When operable, the acoustic sensor may be considered as an operable channel for each valve.

*** - Fully qualified input instrumentation is being installed in accordance with the NRC's TMI Action Plan. Until installation is completed, this function will be satisfied using the plant process computer.

TABLE TS.4.1-1
(Page 5 of 5)

	Channel Description	Check	Calibrate	Functional Test	Response Test	Remarks
35.	Event Monitoring Instrumentation	M	R	NA	NA	Includes all those in FSAR Table 7.7-2 and Table TS.3.15-1 not included elsewhere in this Table
36.	Steam Exclusion Actuation System	W	R	M	NA	See FSAR Appendix I, Section 1.14.6
37.	Pressurizer PORV Control	NA	R	M	NA	Instrument Channels for PORV Control Including Overpressure Mitigation System

S - Each Shift

D - Daily

W - Weekly

M - Monthly

Q - Quarterly

R - Each refueling shutdown

P - Prior to each startup if not done previous week

T - Prior to each startup following shutdown in excess of 2 days if not done in the previous 30 days

NA - Not Applicable

* - See Specification 4.1.D

TABLE TS.4.1-2A

MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

	Test	Frequency	FSAR Section Reference
1. Control Rod Assemblies	Rod drop times of full length rods	All rods during each refueling shutdown or following each removal of the reactor vessel head; affected rods following maintenance on or modification to the control rod drive system which could affect performance of those specific rods	7
1a. Reactor Trip Breakers	Open trip	Monthly	-
2. Control Rod Assemblies	Partial movement of all rods	Every 2 weeks	7
3. Pressurizer Safety Valves	Set point	Each refueling shutdown	4
4. Main Steam Safety Valves	Set point	Each refueling shutdown	10
5. Pressurizer PORV Block Valves	Functional	Quarterly	-
6. Pressurizer PORV's	Functional	Every 18 months	-
7. (Deleted)			
8. (Deleted)			
9. Primary System Leakage	Evaluate	Daily	4
10. (Deleted)			
11. Turbine stop valves, governor valves, and intercept valves. (Part of turbine overspeed protection)	Functional	Monthly	10
12. (Deleted)			

NOTES:

* See Specification 4.1.D.

DPR-42 - Amendment No. 17, 25, 26, 46
DPR-60 - Amendment No. 11, 19, 20, 40

B. Station Batteries

1. Each battery shall be tested each month. Tests shall include measuring voltage of each cell to the nearest hundredth volt, and measuring the temperature and density of a pilot cell in each battery.

2. The following additional measurements shall be made every three months: the density and height of electrolyte in every cell, the amount of water added to each cell, and the temperature of each fifth cell.

3. All measurements shall be recorded and compared with previous data to detect signs of deterioration or need of equalization charge according to the manufacturer's recommendation.

4. The batteries shall be subjected to a performance test discharge during the first refueling and once every five years thereafter. Battery voltage shall be monitored as a function of time to establish that the battery performs as expected during heavy discharge and that all electrical connections are tight.

5. Integrity of Station Battery fuses shall be checked once each day when the battery charger is running.

C. Pressurizer Heater Emergency Power Supply

The emergency pressurizer heater supply shall be demonstrated operable at least once every 18 months by transferring Backup Heater Group "B" from its normal bus to its safeguards bus and energizing the heaters.

The surveillance specified for the pressurizer heater power source provides assurance that Backup Heater Group "B" can be transferred to its emergency bus. Normally, this group of heaters is supplied from a normal plant 480 volt bus. In an emergency, a manual transfer switch can be used to supply the heater group from a safeguards supply bus.

4.8 STEAM AND POWER CONVERSION SYSTEMS

Applicability

Applies to periodic testing requirements of the auxiliary feedwater, steam generator power operated relief valves, and steam exclusion systems.

Objective

To verify the operability of the steam and power conversion systems required for emergency shutdown cooling of the plant.

Specification

A. Auxiliary Feedwater System

1. Each motor-driven auxiliary feedwater pump shall be started at intervals of one month and full flow to the steam generators shall be demonstrated once every refueling shutdown.

2. The steam turbine-driven auxiliary feedwater pump shall be started at intervals of one month and full flow to the steam generators shall be demonstrated once each year when steam from the steam generators is available.

3. The auxiliary feedwater pumps discharge valves shall be tested by operator action at intervals of one month.

4. Motor-operated valves required to function during accident conditions shall be tested at intervals of one month.

5. These tests shall be considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly.

6. During power operation, for the manual valves outside containment, that could reduce AFW flow, if improperly positioned, to less than assumed in the accident analysis, monthly inspection are required to verify the valves are locked in the proper position required for emerency use.

7. After each cold shutdown and prior to exceeding 10% power, a test is required to verify the normal flow path from the primary AFW source to the steam generators. This test may consist of maintaining steam generator level during startup with the auxiliary feed pumps.

8. At least once every 18 months during shutdown verify that each pump starts as designed automatically and each automatic valve in the flow path actuates to its correct position upon receipt of each auxiliary feedwater actuation test signal.

B. Power Operated Relief Valves

Each power operated main steam relief valve shall be isolated and tested monthly.

C. Steam Exclusion System

Isolation dampers in each duct that penetrates rooms containing equipment required for a high energy line rupture outside of containment shall be tested for operability once each month.

In addition, damper mating surfaces will be examined visually at each reactor refueling shutdown to assure that no physical change has occurred that could affect leakage.

Basis

Monthly testing of the auxiliary feedwater pumps, monthly valve inspections, and startup flow verification provide assurance that the AFW system will meet emergency demand requirements. The discharge valves of the pumps are normally open, as are the suction valves from the condensate storage tanks. Proper opening of the steam admission valve on each turbine-driven pump will be demonstrated each time a turbine-driven pump is tested. Ventilation system isolation dampers required to function for the postulated rupture of a high energy line will also be tested.

At 18-month intervals, pump starting and valve positioning is verified using test signals to simulate each of the automatic actuation parameters.

Reference

FSAR, Sections 6.6, 14, and Appendix I.

D. Each member of the plant staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Superintendent Radiation Protection who shall meet or exceed the qualifications of Regulatory Guide 1.8, September, 1975, and (2) the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents. The training program shall be under the direction of a designated member of Northern States Power management.

E. A training program for the fire brigade shall be maintained under the direction of a designated member of Northern States Power management. This program shall meet the requirements of Section 27 of the NFPA Code - 1976 with the exception of training scheduling. Fire brigade training shall be scheduled as set forth in the plant training program.

TABLE TS.6.1-1

MINIMUM SHIFT CREW COMPOSITION (Note 1 and 3)

CATEGORY	BOTH UNITS IN COLD SHUTDOWN OR REFUELING SHUTDOWN	ONE UNIT IN COLD SHUTDOWN OR REFUELING SHUTDOWN AND ONE UNIT ABOVE COLD SHUTDOWN	BOTH UNITS ABOVE COLD SHUTDOWN
No. Licensed Senior Operators (LSO)	2 (Note 2)	2 (Note 2)	2
Total No. Licensed Operators (LSO & LO)	4	4	5
Total No. Licensed & Unlicensed Operators	6	7	8
Shift Technical Advisor	0	1	1

NOTES:

1. Shift crew composition may be one less than the minimum requirements for a period of time not to exceed two hours in order to accomodate an unexpected absence of one duty shift crew member provided immediate action is taken to restore the shift crew composition to within the minimum requirements specified.

2. Does not include the licensed Senior Reactor Operator, or Senior Reactor Operator Limited to Fuel Handling, supervising refueling operatons.

3. Each LSO and LO shall be licensed on each unit.

1. a. Paragraph 20.203 "Caution signs, labels, signals and controls". In lieu of the "Control device" or alarm signal required by paragraph 20.203(c)(2), each high radiation area in which the intensity of radiation is 1000 mRem/hr or less shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (or continuous escort by a qualified person for the purpose of making a radiation survey) and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

   b. The above procedure shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mRem/hr, except that locked doors shall be provided to prevent unauthorized entry into these areas and the keys to these locked doors shall be maintained under the administrative control of the Plant Manager.

2. A program shall be implemented to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

   a. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

   b. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

   A program acceptable to the Commission was described in letters from L O Mayer, NSP, to Director of Nuclear Reactor Regulation, dated December 31, 1979 "Lessons Learned Implementation" and March 13, 1980, "1/1/80 Lessons Learned Implementation Additional Information".

3. A program shall be implemented which will ensure the capability to accurately determine the airborne iodine concentration in essential plant areas under accident conditions. This program shall include the following:

   a. Training of personnel,

   b. Procedures for monitoring, and

   d. Provisions for maintenance of sampling and analysis equipment.

   A program acceptable to the Commission was described in letters from L O Mayer, NSP, to Director of Nuclear Reactor Regulation, dated December 31, 1979 "Lessons Learned Implementation" and March 13, 1979, "1/1/80 Lessons Learned Implementation Additional Information".

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-282 AND 50-306

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 46 to Facility Operating License No. DPR-42, and Amendment No. 40 to Facility Operating License No. DPR-60 issued to Northern States Power Company (the licensee), which revised Technical Specifications for operation of Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2 (the facilities) located in Goodhue County, Minnesota. The amendments are effective as of the date of issuance.

The amendments revise the Technical Specifications to incorporate additional requirements as a result of reviews and modifications accomplished to satisfy the Category "A", TMI-2 Lessons Learned recommendations.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated December 19, 1980, (2) Amendment Nos. 46 and 40 to License Nos. DPR-42 and DPR-60, and (3) the Commission's related letter dated March 2, 1981 . All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Environmental Conservation Library, 300 Nicollet Mall, Minneapolis, Minnesota 55401. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 2nd day of March, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

R. A. Clark

R. A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing