Docket Nos. 50-282/306

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

Gentlemen:

In response to your request dated July 8, 1977, the Commission has issued the enclosed Amendment Nos. 25 and 19 to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2, respectively.

The amendments consist of miscellaneous changes in the Technical Specifications to (1) revise the diesel generator testing, (2) revise the sampling tests for boron and iodine, (3) clarify the dual role of the Residual Heat Removal system and (4) make miscellaneous administrative changes to correct typographical errors, clarify the intent of the Technical Specifications and relocate the Spent Fuel Pool Special Ventilation System limiting conditions for operation and surveillance requirements. In addition, we have changed the Technical Specifications relating to the use of respiratory protection equipment according to our letter of August 31, 1977, and your response dated September 13, 1977.

During our review of your proposed request we found that certain changes were necessary to meet NRC requirements. Your staff has agreed to these changes and they have been incorporated.

Copies of the related Safety Evaluation and the Notice of Issuance also are enclosed.

Sincerely,

Original signed by
M. Grotenhuis

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment Nos. 25 and 19 to License Nos. DPR-42 and DPR-60
2. Safety Evaluation
3. Notice

Const. 1
60

*SEE ATTACHED YELLOW FOR PREVIOUS CONCURRENCES.

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DATE	1/16/78	1/14/78	/ /*	/ /*	1/18/78	1/18/78

NRC FORM 318 (9-76) NRCM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

Docket Nos. 50-282/306

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

Gentlemen:

In response to your request dated July 8, 1977, the Commission has issued the enclosed Amendment Nos. and to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2, respectively.

The amendments consist of miscellaneous changes in the Technical Specifications to (1) revise the diesel generator testing, (2) revise the sampling tests for boron and iodine, (3) clarify the dual role of the Residual Heat Removal system and (4) make miscellaneous administrative changes to correct typographical errors, clarify the intent of the Technical Specifications and relocate the Spent Fuel Pool Special Ventilation System limiting conditions for operation and surveillance requirements.

During our review of your proposed request we found that certain changes were necessary to meet NRC requirements. Your staff has agreed to these changes and they have been incorporated.

Copies of the related Safety Evaluation and the Notice of Issuance also are enclosed.

Sincerely,

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment Nos. and to License Nos. DPR-42 and DPR-60
2. Safety Evaluation
3. Notice

11/16/77

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 25
License No. DPR-42

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

   A. The application for amendment by the Northern States Power Company (the licensee) dated July 8, 1977, 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 applications, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (1) 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 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. 25 , 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

Marshall Grotenhuis

for Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 18, 1978

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. 19
License No. DPR-60

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

   A. The application for amendment by the Northern States Power Company (the licensee) dated July 8, 1977, 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 applications, 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 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. 19 , 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

Marshall Grotenhuis

for Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 18, 1978

ATTACHMENT TO LICENSE AMENDMENT NOS. 25 AND 19

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

DOCKET NOS. 50-282 AND 50-306

Replace the following pages of the Technical Specifications contained in Appendix A of the above-indicated licenses with the attached pages bearing the same numbers, except as otherwise indicated. The changed areas on the revised pages are reflected by a marginal line.

Remove	Insert
TS-i and iv	TS-i and iv
3.2-2	3.2-2
3.3-1	3.3-1
———	3.3-1A (new page)
3.3-2	3.3-2
3.3-7	3.3-7
3.6-3A	3.6-3A
3.6-3B	———
3.6-4	3.6-4
———	3.8-2A (new page)
3.8-3	3.8-3
3.8-4	3.8-4
3.13-2	3.13-2
Table 4.1-1 (1 of 4)	Table 4.1-1 (1 of 4)
Table 4.1-2A	Table 4.1-2A
Table 4.1-2B	Table 4.1-2B
4.4-4	4.4-4
4.4-8	4.4-8
4.6-1	4.6-1
———	4.6-1A (new page)
4.6-2	4.6-2
———	4.15-1 (new page)
———	4.15-2 (new page)
6.5-2	6.5-2
6.5-3	6.5-3
6.5-4	———
6.5-5	———
Table 6.5-1 (3 pages)	———

TECHNICAL SPECIFICATIONS

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7. Motor-operated valve Number 8809C for that unit shall be open, shall have its valve position monitor light operable, and shall have its motor control center supply breaker physically locked in the open position.

8. Manual valves in the boric acid system shall be physically locked in the position required for automatic boric acid injection following a steam line break accident.

C. Except as specified in 3.2 D.below, the reactor in the second unit shall not be made or maintained critical nor shall it be heated or maintained above 200°F with the reactor in the other unit already critical unless the following conditions are satisfied.

1. A minimum of two charging pumps for each unit shall be operable.

2. At least three of the four boric acid transfer pumps shall be operable.

3. At least two boric acid tanks shall each contain a minimum of 2000 gallons of 11.5% to 13% by weight boric acid solution at a temperature of at least 145°F.

4. System piping and valves shall be operable to the extent of establishing for each unit two independent flow paths for boric acid injection: one flow path from its associated boric acid tank to the core, and one flow path from its refueling water storage tank to its core. The flow paths shall be arranged so that each boric acid tank can supply only its associated unit.

5. Two channels of heat tracing shall be operable for the flow paths from the boric acid tanks.

6. Automatic valves, piping, and interlocks associated with the above components which are required to operate for the steam line break accident are operable.

7. The motor-operated valve in each unit numbered 8809C shall be open, shall have its valve position monitor light operable, and shall have its motor control center supply breaker physically locked in the open position.

3.3 ENGINEERED SAFETY FEATURES

Applicability

Applies to the operating status of the engineered safety features.

Objective

To define those limiting conditions that are necessary for operation of engineered safety features: (1) to remove decay heat from the core in an emergency or normal shutdown situations, and (2) to remove heat from containment in normal operating and emergency situations.

Specification

A. Safety Injection and Residual Heat Removal Systems

1. A reactor shall not be made or maintained critical nor shall it be heated or maintained above 200°F unless the following conditions are satisfied except as permitted in Specification 3.3 A.2.

   a. The refueling water tank contains not less than 200,000 gallons of water with a boron concentration of at least 1950 ppm.

   b. Each reactor coolant system accumulator shall be operable except that each may be isolated below a pressurizer pressure of 1000 psig. Operability requires:

      (1) The isolation valve open
      (2) Between 1250 and 1282.9 cubic feet of borated water
      (3) A minimum boron concentration of 1900 ppm
      (4) A nitrogen cover pressure of at least 700 psig

   c. Two safety injection pumps are operable except that pump control switches in the control room may be in the "pullout" position whenever the steam bubble is not established in the Pressurizer.

   d. Two residual heat removal pumps are operable.

   e. Two residual heat exchangers are operable.

   f. Automatic valves, interlocks and piping associated with the above components and required to function during accident conditions, are operable.

   g. Manual valves in the above systems that could (if one is improperly positioned) reduce injection flow below that assumed for accident analyses, shall be blocked and tagged in the proper position for injection. RHR system valves, however, may be positioned as necessary to regulate plant heatup or cooldown rates when the reactor is subcritical. All changes in valve position shall be under direct administrative control.

h. For Unit 1 operation, the following valve conditions shall exist:

(1) Safety injection system motor-operated valves 8801A, 8801B, 8806A 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 open.

(2) Safety injection system motor-operated valves 8816A and 8816B shall be closed, shall have valve position monitor lights operable, and shall have the motor control center supply breakers physically locked.

(3) Accumulator discharge valves 8800A and 8800B shall have position monitor lights and alarms operable.

(4) Residual Heat Removal System valves 8701A and 8701B shall have normal valve position indication operable.

i. For Unit 2 operation, the valve conditions corresponding to these stated in Specification 3.3.A.1.h for Unit 1 shall exist.

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 during power operation operability is not restored within the time specified, the reactor shall be placed in the hot shutdown condition. If the requirements of TS 3.3 A.1 are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition.

a. One safety injection pump may be out of service, provided the pump is restored to operable status within 24 hours. The other safety injection pump shall be tested to demonstrate operability prior to initiating repair of the inoperable pump.

b. One residual heat removal pump may be out of service, provided the pump is restored to operable status within 24 hours. The other residual heat removal pump shall be tested to demonstrate operability prior to initiating repair of the inoperable pump.

c. One residual heat exchanger may be out of service for a period of no more than 48 hours.

d. Any redundant valve in the system required for safety injection, may be inoperable provided repairs are completed within 24 hours. Prior to initiating repairs, all valves in the system that provide redundancy shall be tested to demonstrate operability.

e. One accumulator may be inoperable for up to one hour.

f. One safety injection system and one residual heat system may be inoperable for a time interval not to exceed 24 hours provided the redundant safety injection system and heat removal system required for functioning during accident conditions is operable.

B. Containment Cooling Systems

1. A reactor shall not be made or maintained critical nor shall it be heated above 200°F unless the following conditions are satisfied except as permitted by Specification 3.3.B.2.

   a. Two containment spray pumps are operable.

   b. Four fan cooler units are operable.

limited time in hot shutdown, the reactor will be placed in the cold shutdown condition, utilizing normal shutdown and cooldown procedures. In the cold shutdown condition there is no possibility of a LOCA that would release fission products or damage the fuel elements.

The specified intervals for equipment inoperability are based on:

(1) Assuring with high reliability that the safety system will function properly if required to do so.

(2) Allowance of sufficient time to complete required repairs and testing using safe and proper procedures.

Assuming the reactor has been operating at full rated power for at least 100 days, the magnitude of the decay heat decreases as follows after initiating hot shutdown.

Time After Shutdown	Decay Heat, % of Rated Power
1 min.	4.5
30 min.	2.0
1 hour	1.62
8 hours	0.96
48 hours	0.62

Thus, the requirement for core cooling in case of a postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition significantly reduced the potential consequences of a loss-of-coolant accident, and also allows more free access to some of the engineered safeguards components in order to effect repairs. Failure to complete repairs within 48 hours after placing the reactor in the hot shutdown condition is considered indicative of need for major maintenance, and in such case the reactor would therefore be placed in the cold shutdown condition.

The accumulator and refueling water tank conditions specified are consistent with those assumed in the LOCA analysis.[2]

The containment cooling function is provided by two independent systems: fan-coolers and containment sprays. During normal operation, only three of the four fan-coolers are required to remove heat lost from equipment and piping within the containment.[3] In the event of the Design Basis Accident, any one of the following combinations will provide sufficient cooling to reduce containment pressure: four fan-coolers, two containment spray pumps, or two fan-coolers plus one containment spray pump.[4] One of the four fan-coolers is permitted to be inoperable during power operation. This is an abnormal operating situation, in that plant operating procedures require that an inoperable fan-cooler be repaired as soon as practical. However, because of the difficulty of access

E. Emergency Air Treatment Systems

1. Except as specified in Specification 3.6.E.3 below, all trains of the Shield Building Ventilation System, the Auxiliary Building Special Ventilation System, and the diesel generation required for their operation shall be operable at all times.

2. a. The results of in-place DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks respectively shall show $\geq$99% DOP removal for particles having a mean diameter of 0.7 microns and $\geq$99% halogenated hydrocarbon removal.

   b. The results of laboratory carbon sample analysis shall show $\geq$90% radioactive methyl iodide removal efficiency (130°C, 95% RH).

3. From and after the date that one train of the Shield Building Ventilation System or one train of the Auxiliary Building Special Ventilation System is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days (unless such train is made operable), provided that during such seven days the redundant train is verified to be operable daily.

4. If the conditions for operability of the Shield Building Ventilation System cannot be met, procedures shall be initiated immediately to establish reactor conditions for which containment integrity is not required for the affected unit.

5. If the conditions for operability of the Auxiliary Building Special Ventilation System cannot be met, procedures shall be initiated immediately to establish reactor conditions for which containment integrity is not required in either unit.

Basis

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 containment integrity is required. 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 vent zone and its associated ventilation system have been designed to serve as secondary containment following a loss of coolant accident.[3] Special care was taken to design the access doors in the boundary and isolation valves in normal ventilation systems so that containment integrity can be intact during reactor operation. During construction of Unit 2, it may be necessary to provide temporary openings in the boundary. The zone can perform its accident function if they are 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).

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.

When the reactor vessel head is removed with containment integrity violated, the reactor must not only be in the cold shutdown condition, but also in the refueling shutdown condition. This ten percent shutdown margin prevents the occurrence of criticality under any circumstances, even when fuel is being moved during refueling operations.

Containment integrity is not required when new fuel is in the reactor since radioactivity is negligibly small. This condition in the specification expedites initial testing of the reactor.

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. Therefore, the two parts of Specification 3.6.A.1 allow containment integrity to be violated when a fission product inventory is present only under circumstances that preclude both criticality and release of stored energy.

D. Spent Fuel Pool Special Ventilation System

1. Except as specified in Specification 3.8.D.3 below, both trains of the Spent Fuel Pool Special Ventilation System and the diesel generators required for their operation shall be operable at all times.

2. a. The results of in-place DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks respectively shall show $\geq$ 99% DOP removal for particles having a mean diameter of 0.7 microns and $\geq$99% halogenated hydrocarbon removal.

   b. The results of laboratory carbon sample analysis shall show $\geq$90% radioactive methyl iodide removal efficiency (130°C, 95% RH).

   c. The Spent Fuel Pool Special Ventilation System fans shall operate within $\pm$10% of 5200 cfm per train.

3. From and after the date that one train of the Spent Fuel Pool Special Ventilation System is made or found inoperable for any reason, fuel handling operations are permissible only during the succeeding seven days (unless such train is made operable) provided that the redundant train is verified to be operable daily.

4. If the conditions for operability of the Spent Fuel Pool Special Ventilation System cannot be met, fuel handling operations in the Auxiliary Building shall be terminated immediately.

Basis

The equipment and general procedures to be utilized during refueling are discussed in the FSAR. Detailed instructions, the precautions specified above, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety.[1] Whenever changes are not being made in core geometry, one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels (B. above) and neutron flux provides immediate indication of an unsafe condition. The residual heat removal pump is used to maintain a uniform boron concentration.

The shutdown margin indicated in A.5. above will keep the core subcritical, even if all control rods were withdrawn from the core. During refueling, the reactor refueling cavity is filled with approximately 275,000 gallons of borated water. The boron concentration of this water is sufficient to maintain the reactor subcritical by approximately 10% $\Delta k/k$ in the cold condition with all rods inserted, and will also maintain the core subcritical even if no control rods were inserted into the reactor.[2] Periodic checks of refueling water boron concentration insure that proper shutdown margin is maintained. A.6. above allows the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

No movement of fuel in the reactor is permitted until the reactor has been subcritical for at least 100 hours to permit decay of the fission products in the fuel. The delay time is consistent with the fuel handling accident analysis.[3]

The spent fuel assemblies will be loaded into the spent fuel cask for shipment to a reprocessing plant after sufficient decay of fission products. In loading the cask into a carrier, there is a potential drop of 66 feet[5]. The cask will not be loaded onto the carrier for shipment prior to a 3-month storage period. At this time, the radioactivity has decayed so that a release of fission products from all fuel assemblies in the cask would result in off-site doses less than 10 CFR Part 100. It is assumed, for this dose analysis that 12 assemblies rupture after storage for 90 days. Other assumptions are the same as those used in the dropped fuel assembly accident in the SER, Section 15. The resultant doses at the site boundary are 94 Rems to the thyroid and 1 Rem whole body.

The Spent Fuel Pool Special Ventilation System [4] is a safeguards system which maintains a negative pressure in the spent fuel enclosure upon detection of high area radiation. The Spent Fuel Pool Normal Ventilation System is automatically isolated and exhaust air is drawn through filter modules containing a roughing filter, particulate filter, and a charcoal filter before discharge to the environment via one of the Shield Building exhaust stacks. Two completely redundant trains are provided. The exhaust fan and filter of each train are shared with the corresponding train of the Containment In-service Purge System. High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers in each SFPSVS filter train. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a HEPA filter leakage of less than 1% through DOP testing and a charcoal adsorber leakage of less than 1% through halogenated hydrocarbon testing. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90% under test conditions which are more severe than accident conditions. The satisfactory completion of these periodic tests combined with the qualification testing conducted on new filters and adsorber provide a high level of assurance that the emergency air treatment systems will perform as predicted in the accident analyses.

References

(1) FSAR Section 9.5.2
(2) FSAR Table 3.2.1-1
(3) FSAR Section 14.2.1
(4) FSAR Section 9.6
(5) FSAR Page 9.5-20a

3.13 CONTROL ROOM AIR TREATMENT SYSTEM

Basis

The Control Room Special Ventilation System is designed to filter the Control Room atmosphere during accident conditions. The system is designed to automatically start on a high radiation signal in the ventilation air or when a Safety Injection signal is received from either unit. Two completely redundant trains are provided.

Each train has a filter unit consisting of a prefilter, HEPA filters, and charcoal adsorbers. The HEPA filters remove particulates from the Control Room atmosphere and prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to remove any radioiodines from the Control Room atmosphere. The in-place test results should indicate a HEPA filter leakage of less than 1% through DOP testing and a charcoal adsorber leakage of less than 1% through halogenated hydrocarbon testing. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90% under test conditions more severe than expected accident conditions. System flows should be near their design values. The verification of these performance parameters combined with the qualification testing conducted on new filters and adsorber provide a high level of assurance that the Control Room Special Ventilation System will perform as predicted in reducing potential doses to plant personnel below those levels stated in Criterion 19 of Appendix A to 10 CFR 50.

In-place testing procedures will be established utilizing applicable sections of ANSI N510 - 1975 standard as a procedural guideline only.

TABLE TS.4.1-1
(Page 1 of 4)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

Channel Description	Check	Calibrate	Functional Test	Response Test	Remarks
1. Nuclear Power Range	S(1) M(4)	D(2) Q(4)	M(3) M(5) M(6)	R	1) Once/shift when in service 2) Heat balance 3) Signal to ΔT; bistable action (permissive, rod stop, trips) 4) Upper and lower chambers for axial off-set using in-core detectors 5) Simulated signal for testing positive and negative rate bistable action 6) Quadrant Power Tilt Monitor
2. Nuclear Intermediate Range	*S(1)	NA	P(2)	R	1) Once/shift when in service 2) Log level; bistable action (permissive, rod stop, trips)
3. Nuclear Source Range	*S(1)	NA	P(2)	R	1) Once/shift when in service 2) Bistable action (alarm, trips)
4. Reactor Coolant Temperature	S(1,2)	R(1,2,3)	M(1) M(2) T(3)	R(1) R(2)	1) Overtemperature ΔT 2) Overpower ΔT 3) Control Rod Bank Insertion Limit Monitor
5. Reactor Coolant Flow	S	R	M	NA	
6. Pressurizer Water Level	S	R	M	NA	
7. Pressurizer Pressure	S	R	M	NA	
8. 4KV Voltage & Frequency	NA	R	M	NA	Reactor protection circuits only
8a. RCP Breakers	NA	R	T	NA	
9. Analog Rod Position	S(1) M(2)	R	T(2)	NA	1) With step counters 2) Rod Position Deviation Monitor Tested by updating computer bank count and comparing with analog rod position test signal.

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 maintainance 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.	(Deleted)			
6.	(Deleted)			
7.	(Deleted)			
8.	Fire Protection Pump & Power Supply	Functional	Monthly	9.6.1
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 (Note 1)	10
12.	(Deleted)			

NOTES:

1. Performance of the turbine stop valve, governor valve, and intercept valve functional test may be omitted, on a one-time basis, during the month of February, 1976 on Unit 1.

* See Specification 4.1.D.

TABLE TS.4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

	Test	Frequency	FSAR Section Reference
1. Reactor Coolant Samples	Radiochemistry (a) Gross Beta-gamma activity (excluding tritium)	Monthly 5/week	
	Tritium Activity *Chemistry (Cl, F, and O_2)	Weekly 5/week	
2. Reactor Coolant Boron (b)	*Boron concentration	2/week (c)	9.2
3. Refueling Water Storage Tank Water Sample	Boron concentration	Weekly	
4. Boric Acid Tanks	Boron concentration	Twice/week	
5. Chemical Additive Tank	NaOH concentration	Monthly	6.4
6. Accumulator	Boron concentration	Monthly	6
7. Spent Fuel Pit	Boron concentration	Monthly	9.5.5
8. Secondary Coolant	Gross Beta-gamma activity	Weekly	
	I-131 concentration in water and in steam (e)	Weekly (d)	

NOTES:

a. To determine activity of corrosion products having a half-life greater than 30 minutes. (See Specification 3.1 D.)

b. See Specification 3.8 for requirements during refueling.

c. The maximum interval between analyses shall not exceed 5 days.

d. If activity of the samples is greater than 10% of the limit in Specification 3.4.A.9, the frequency shall be daily.

e. I-131 analysis in steam is not required if I-131 is undetectable in water.

* See Specification 4.1.D.

B. Emergency Charcoal Filter Systems

1. Periodic tests of the shield building ventilation system shall be performed at quarterly intervals to demonstrate operability. Each redundant train shall be determined to be operable at the time of its periodic test if it meets drawdown performance computed for the test conditions with 75% of the shield building inleakage specified in Figure TS 4.4-1 after initiation of a simulated signal of safety injection or high containment building pressure signal.

2. Periodic tests of the auxiliary building special ventilation system shall be performed at approximately quarterly intervals to demonstrate its operability. Each redundant train shall be determined to be operable at the time of periodic test if it isolates the normal ventilation system and produces a measureable negative pressure in the ABSVZ within 6 minutes after actuation by a simulated safety injection signal or high radioactivity signal in the auxiliary building stack.

3. At least once per operating cycle, or once each 18 months, which ever comes first, tests of the filter units in the Shield Building Ventilation System and the Auxiliary Building Special Ventilation System shall be performed as indicated below:

   a. The pressure drop across the combined HEPA filters and the charcoal adsorbers shall be demonstrated to be less than 6 inches of water at system design flow rate ($\pm$10%).

   b. The inlet heaters and associated controls for each train shall be determined to be operable.

   c. Automatic initiation of each train of each ventilation system.

4. a. The tests of Specification 3.6.E.2 shall be performed at least once per operating cycle, or once every 18 months whichever occurs first, or after every 720 hours of system operation or following painting, fire or chemical release in any ventilation zone communicating with the system that could contaminate the HEPA filters or charcoal adsorbers.

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.

The shield building ventilation system consists of two independent systems that have only a discharge point in common, the shield building vent. Both systems are normally activated and one alone must be capable of accomplishing the design function of the system. During the first operating cycle, tests will be performed to demonstrate the capability of both the separate and combined systems under different wind conditions up to 45 mph if possible.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per operating cycle to verify operability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. A charcoal adsorber tray which can accommodate a sufficient number of representative adsorber sample modules for estimating the amount of penetration of the system adsorbent through its life is currently under development. When this tray is available, sample modules will be installed with the same batch characteristics as the system adsorbent and will be withdrawn for the methyl iodide removel efficiency tests. Each module withdrawn will be replaced or

4.6 PERIODIC TESTING OF EMERGENCY POWER SYSTEM

Applicability

Applies to periodic testing and surveillance requirements of the emergency power system.

Objective

To verify that the emergency power sources and equipment are operable.

Specification

The following tests and surveillance shall be performed:

A. **Diesel Generators**

1. At least once each month, for each diesel generator:

   a. Verify the fuel level in the day and engine-mounted tank.
   b. Verify the fuel level in the fuel storage tank.
   c. Verify that a sample of diesel fuel from the fuel storage tank is within the acceptable limits specified in Table 1 of ASTM D975-68 when checked for viscosity, water, and sediment.
   d. Verify the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.
   e. Verify the diesel starts from the normal standby condition.
   f. Verify the generator synchronizes, is loaded to at least 1375 kw, and operated for at least one hour.

2. At least once each 18 months:

   a. Subject each diesel generator to a thorough inspection in accordance with procedures prepared in conjunction with the manufacturer's recommendations for this class of standby service.
   b. For each unit, simulate a loss of offsite power in conjunction with a safety injection signal, and:

      1. Verify de-energization of the emergency busses and load shedding from the emergency busses.
      2. Verify the diesels start from the normal standby condition on the auto-start signal and energize the emergency busses in one minute.
      3. Verify that the diesel generator system trips, except those for engine overspeed and the generator differential current, are automatically bypassed.
      4. Verify that the auto-connected loads do not exceed 3000 kw.

   c. **Verify the capability of each generator to operate at least one hour while loaded to 3000 kw.**
   d. **Verify the capability of each generator to reject a load of at least 650 kw without tripping.**
   e. During this test, operation of the emergency lighting system shall be ascertained.

B. <u>Station Batteries</u>

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.

Basis

The monthly tests specified for the diesel generators will demonstrate their continued capability to start and to carry rated load. The fuel supplies and starting circuits and controls are continuously monitored, and abnormal conditions in these systems would be alarm-indicated without need for test startup.

The less frequent overall system test will demonstrate that the emergency power system and the control systems for the engineered safeguards equipment will function automatically in the event of loss of all other sources of a-c power, and that the diesel generators will start automatically in the event of a loss-of-coolant accident. This test will demonstrate proper tripping of motor feeder breakers, main supply and tie breakers on the affected bus, and sequential starting of essential equipment, as well as the operability of the diesel generators.

The specified test frequencies provide reasonable assurance that any mechanical or electrical deficiency will be detected and corrected before it can result in failure of one emergency power supply to respond when called upon to function. Its possible failure to respond is, of course, anticipated by providing two diesel generators, each supplying, through an independent bus, a complete and adequate set of engineered safeguards equipment. Further, both diesel generators are provided as backup to multiple sources of external power, and this multiplicty of sources should be considered with regard to adequacy of test frequency.

Each diesel generator can start and be ready to accept full load within 10 seconds, and will sequentially start and supply the power requirements for one complete set of safeguards equipment in approximately one minute.[1]

An internal fault in the generator could damage the generator severely. Moreover, this change complies with BTP EICSB 17. Auto-connected loads should not exceed the overload rating of the diesel generator for the 2000 hour maintenance interval, as prescribed in Regulatory Guide 1.9.

Station batteries will deteriorate with time, but precipitous failure is extremely unlikely. The surveillance specified is that which has been demonstrated over the years to provide indication of a cell becoming unserviceable long before it fails.

If a battery cell has deteriorated, or if a connection is loose, the voltage under load will drop excessively, indicating need for replacement or maintenance.

Reference

(1) FSAR, Section 8.2.3

4.15 SPENT FUEL POOL SPECIAL VENTILATION SYSTEM

<u>Applicability</u>

Applies to the periodic testing requirements for the Spent Fuel Pool Special Ventilation System (SFPSVS).

<u>Objective</u>

To specify tests for assuring the operability of the Spent Fuel Pool Special Ventilation System.

<u>Specification</u>

A. At least once per operating cycle or once every 18 months, whichever occurs first, the following shall be demonstrated:

   1. The pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at system design flow rate ($\pm$10%).

   2. Automatic initiation of each train shall be demonstrated with a simulated high radiation signal.

B. 1. The tests of Specification 3.8.D.2 shall be performed at least once per operating cycle, or once every 18 months whichever occurs first, or after every 720 hours of system operation or following painting, fire or chemical release in any ventilation zone communicating with the system that could contaminate the HEPA filters or charcoal adsorbers.

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

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

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

Basis

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter.

The frequency of tests and sample analysis is necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. A charcoal adsorber tray which can accommodate a sufficient number of representative adsorber sample modules for estimating the amount of penetration of the system adsorbent through its life is currently under development. When this tray is available, sample modules will be installed with the same batch characteristics as the system adsorbent and will be withdrawn for the methyl iodide removal efficiency tests. Each module withdrawn will be replaced or blocked off. Until these trays can be installed, to guarantee a representative adsorbent sample, procedures should allow for the removal of a tray containing the oldest batch of adsorbent in each train, emptying of one bed from the tray, mixing the adsorbent thoroughly, and obtaining at least two samples. One sample will be submitted for laboratory analysis and the other held as a backup. If test results are unacceptable, all adsorbent in the train will be replaced. Adsorbent in the tray removed for sampling will be renewed. Any HEPA filters found defective will be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52 - Rev. 1 June 1976.

If significant painting, fire, or chemical release occurs such that the HEPA filters or charcoal adsorbers could become contaminated from the fumes, chemicals, or foreign material, the same tests and sample analysis will be performed as required for operational use. The determination of significant will be made by the shift supervisor after consulting knowledgeable staff members.

Operation of each train of the system for 10 hours every month will demonstrate operability of the system and remove excessive moisture which may build up on the adsorber.

Demonstrating automatic initiation of the system using simulated high area radiation signals will assure that the system will start when required.

In-place testing procedures will be established utilizing applicable sections of ANSI N510 - 1975 standard as a procedural guideline only.

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.

C. Maintenance and Test

The following maintenance and test procedures will be developed to satisfy routine inspection, preventive maintenance programs, and operating license requirements.

1. Routine testing of Engineered Safeguards and equipment as required by the facility License and the Technical Specifications.

2. Routine testing of standby and redundant equipment.

3. Preventive or corrective maintenance of plant equipment and systems that could have an effect on nuclear safety.

4. Calibration and preventive maintenance of instrumentation that could affect the nuclear safety of the plant.

5. Special testing of equipment for proposed changes to operational procedures or proposed system design changes.

D. Temporary Changes to Procedures

Temporary changes to procedures described in A, B, and C above, which do not change the intent of the original procedure may be made with the concurrence of two individuals holding senior operator licenses. Such changes shall be documented, reviewed by the Operations Committee and approved by a member of plant management designated by the Plant Manager within one month.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 25 AND 19 TO FACILITY 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

INTRODUCTION

By letter dated July 8, 1977, Northern States Power Company (NSP) requested amendments to Facility License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2 (PINGP). The proposed amendments consist of miscellaneous changes in the Technical Specifications to (1) Revise the diesel generator testing requirements to conform to current NRC guidance, (2) Revise the frequency for Boron analysis and I-131 analysis, (3) Clarification of the dual role of the RHR system, (4) Make miscellaneous administrative changes to correct typographical errors, to clarify the intent of the Technical Specifications and to relocate the limiting conditions of operator and surveillance requirements on the Spent Fuel Pool Special Ventilation System. In addition we have amended the Technical Specifications relating to the use of respiratory protection equipment according to our letter of August 31, 1977, and the NSP response dated September 13, 1977.

DISCUSSION

Item 1 above, diesel generator testing, proposes to change the test program for the emergency diesel generators. The need for an update became apparent during an inspection of the facility on January 18-20, 1977 (See inspection report 50-282/77-1, 50-306/77-1 dated February 8, 1977), when the requirements of a load rejection test were interpreted by the inspector in a manner different than that of the licensee. The interpretation problem is resolved in this proposed amendment by including the tests outlined in IEEE-387 dated March 25, 1972, explicitly in the Technical Specifications rather than by reference. These changes also revise the test program for the emergency diesel generators at Prairie Island to conform to the current NRC guidance in this area. Minor departures from current guidance are required due to unique design features of the facility. The changes have been submitted to satisfy the Region III Office of Inspection and Enforcement comments regarding inconsistencies in the current test program.

Item 2 above proposes changes in Table 4.1-2B, Minimum Frequencies for Sampling Tests. One change revises the boron analysis frequency to be consistent with the requirements at other PWR facilities. The reference to the boron concentration monitoring system (BCMS) has been removed. The BCMS is utilized as an advisory system only and is not related in any way to plant control or protective systems. Refer to page 9.2-15 in the Prairie Island Final Safety Analysis Report. The BCMS has been inoperable for long periods of time and the problems with this instrument have not been corrected. The proposed wording establishes a boron analysis frequency identical to that currently in effect at facilities not equipped with a BCMS.

Another change deletes the requirement to perform an I-131 analysis of secondary steam if I-131 is not detectable in secondary water. There is no reason to perform an I-131 analysis of steam in that instance.

The final change corrects the notes at the bottom of the table to reflect the changes above. In addition, the reference to specification 3.4.A.5 has been corrected to refer to specification 3.4.A.9.

Item 3 above clarifies the dual role of the RHR system. Like the auxiliary feedwater system, it is used during startup and shutdown operation and it also performs a safeguard function (low head safety injection). The current wording of TS.3.3.A.1g could be interpreted to preclude operation of the RHR system to regulate heatup and cooldown rates or to stabilize temperature during plant heatup while reactor coolant chemistry is being brought into specifications. The proposed change would clarify the wording of TS.3.3.A.1.g to recognize the dual role of the RHR system.

Item 4 above proposes changes which are administrative and clerical in nature. This includes; correction of typographical errors, clarifications regarding accumulator operability, Section 3.3 basis, and Table 4.1-1; and relocation of the limiting conditions and surveillance requirements for the Spent Fuel Pool Special Ventilation system to more appropriate sections that will eliminate a possible confusion for plant operations. These are administrative changes and need not be evaluated further.

The portion of the amendments relating to the use of respiratory protection equipment would eliminate any reference to respiratory protection equipment in the technical specifications since it is now specifically addressed by Section 20.103 of 10 CFR Part 20. This is in accord with the revocation provision of subsection 6.5.B.4 of the Technical Specifications, which would also be deleted by these amendments. Section 20.103 references Regulatory Guide 8.15. Although it is not identified as the October 1976 version of Regulatory Guide 8.15, the Commission, in its Statement of

Consideration that accompanied the revised regulation 41 F.R. 230, Page 52300, November 29, 1976, stated that "Changes to the guide would result in a redating or renumbering of the guide with appropriate changes to Section 20.103(c) including prior public notice and procedures thereof in the Federal Register." This is an administrative change and will not be evaluated further.

During our review of the proposed changes, we found that certain modifications to the proposal were necessary to meet NRC requirements. These changes were discussed with the licensee's staff. The licensee has agreed to these changes and the changes will be incorporated into the amendment.

EVALUATION

Item 1, diesel generator testing, NSP proposes to change the test program as indicated in the amendment request. We have reviewed the new test program and find that it includes all the tests currently required. It conforms to the current NRC guidance and complies with recommendations made by the Region III Office of Inspection and Enforcement. We did modify the diesel generator tests from the request to include recognition of verification of the generator differential current trip bypass to prevent damage to the generator because of an internal fault. This is consistent with the Branch Technical Positions, EICSB-17. We conclude that the proposed amendment relating to deisel generator testing would improve the safety of operation of the plant by updating and clarifying the Technical Specification and therefore find them acceptable.

Item 2-NSP proposes changes in Table 4.1.2B involving boron concentration tests and I-131 tests. The Boron Concentration Monitoring System (BCMS) has not been operable and the proposed changes establish a boron analysis consistent with the Branch Technical Positions, EICSB-17. We conclude that the proposed amendment relating to diesel generator testing would improve the safety of operation of the plant by updating and clarifying the Technical Specifications and therefore find them acceptable.

Item 2-NSP proposes changes in Table 4.1.2B involving boron concentration tests and I-131 tests. The Boron Concentration Monitoring System (BCMS) has not been operable and the proposed changes establish a boron analysis frequency that is consistent with a plant that does not have a BCMS. This frequency is twice per week with a maximum interval of five days between tests. The previous requirement was also twice per week; however, a daily check was required when the BCMS was not operable. Boron concentration in the reactor coolant water changes slowly and in a predictable manner throughout the lifetime of the core. Should there be an unexpected change in the boron concentrations, the corresponding reactivity change would be noticed by the reactor operators. Among the ways they would notice the changes are the control rod positions which are monitored at least once per shift. On the basis of the above and the fact that during reactor operation the coolant pumps keep the boron uniformly distributed, the measurement of boron concentration twice a week during reactor operation rather than on a daily basis or by the BCMS does not result in a decrease in safety.

The I-131 test requirement now includes tests for both steam and water. The proposed change would not require a test for I-131 in steam if I-131 was undectable in water. No sampling in the steam should be required if sampling in the water shows undectable levels. This is based on the fact that iodine is more soluble in water than in steam and is not as likely to be detected in the steam phase if not detectable in water. The staff therefore concludes that the proposed change in the I-131 test requirement does not result in a decrease in safety.

We have reviewed the proposed changes in Table 4.1-2B. We find that they have no effect on plant safety and are acceptable to the staff.

Item 3, NSP proposes changes which clarify the dual role of the RHR system. When the RHR system is used to remove heat from the reactor coolant system during plant startup or shutdown operations, the heat exchanger flow control valves are manually adjusted to provide the required heat exchanger flow. Total RHR system flow is maintained constant by the automatically controlled heat exchanger bypass valve. When aligned for safeguards operation, the heat exchanger flow control valves are opened and the flow control valve is manually closed.

During plant heatup, RHR is aligned for safeguards operation at a reactor coolant temperature of about 250°F after chemistry is within limits and reactor coolant pumps are started. During cooldown, the RHR system is aligned for heat removal when reactor coolant temperature is reduced to 350°F. In the event of a loss-of-coolant accident when reactor coolant temperature is above 200°F and RHR is aligned for heat removal, operator action would be required to initiate low head safety injection.

Operator action for initiation of low head safety injection is permissible in this condition because:

a) There is a reduced amount of stored energy in the reactor coolant system at this time compared to design bases accident conditions.

b) There is a reduced decay heat production rate at this time compared to design basis accident conditions.

c) The time period the plant is subjected to this condition is a small fraction of the total.

We have reviewed the proposed changes and find that they are acceptable.

ENVIRONMENTAL CONSIDERATION

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 these amendments.

CONCLUSION

We have concluded, based on the considerations discussed above, 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.

Date: January 18, 1978

UNITED STATES NUCLEAR REGULATORY COMMISSION 7590-01

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 Nos. 25 and 19 to Facility Operating License Nos. DPR-42 and DPR-60, issued to the Northern States Power Company (the licensee), which revised the licenses and their appended Technical Specifications for operation of Unit Nos. 1 and 2 of the Prairie Island Nuclear Generating Plant (the facilities) located in Goodhue County, Minnesota. The amendments are effective as of their date of issuance.

The amendments consisted of miscellaneous changes in the Technical Specifications to (1) revise the diesel generator testing, (2) revise the sampling tests for Boron and I-131, (3) clarify the dual role of the Residual Heat Removal system, and (4) make miscellaneous administrative changes to correct typographical errors, clarify the intent of the Technical Specifications and relocate the Spent Fuel Pool Special Ventilation System limiting conditions for operation and surveillance requirements. In addition, we have deleted from the Technical Specifications any reference to respiratory protection equipment since it is now specifically addressed by Section 20.103 of 10 CFR Part 20 of the Commission's regulations.

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 the amendments.

For further details with respect to this action, see (1) the application for amendments dated July 8, 1977, (2) Amendment Nos. 25 and 19 to License Nos. DPR-42 and DPR-60, respectively, and (3) the Commission's related Safety Evaluation. 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 of the Minneapolis Public Library, 300 Nicollet Mall, Minneapolis, Minnesota 55401. A single 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 Operating Reactors.

Dated at Bethesda, Maryland, this 18th day of January, 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Marshall Grotenhuis

Marshall Grotenhuis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors