Docket Nos. 50-282
50-306

OCT 14 1976

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 an additional portion of your request dated May 7, 1975, as modified by your letter dated May 26, 1976, the Commission has issued the enclosed Amendment Nos. 17 and 11 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 revise existing and add new limiting conditions for operation, surveillance requirements and bases for the Control Room Special Ventilation and Emergency Charcoal Filter Systems to the Technical Specifications for the facilities. The amendments also revise several sections of the Technical Specifications to include the miscellaneous clarifications and corrections requested in the two submittals.

During our review of the proposed changes, we found that certain modifications to your proposed Technical Specifications were necessary to meet our requirements. These changes were discussed with your staff and with their agreement were included in these amendments.

This completes our action on your May 7, 1975 request. Our May 10, 1976 issuance of Amendments 13 and 7 to Licenses DPR-42 and DPR-60 authorized items 14, 15 and 16 of your May 7, 1975 request.

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

Sincerely,

Original Signed by:
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 17 to License DPR-42
2. Amendment No. 11 to License DPR-60
3. Safety Evaluation
4. Notice

cc w/enclosures:
See next page

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Recalled NSP 4 NE transmittal letter signed by [illegible] 10/[illegible]/76

Amend 17 - DPR-42
11 - DPR-60

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DATE	9/30/76	9/30/76	10/1/76	10/7/76	10/14/76

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 17
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 May 7, 1975, as modified by filing dated May 26, 1976, 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.

OFFICE ➔						
SURNAME ➔						
DATE ➔						

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by:
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: OCT 1 . 1973

OFFICE ➤						
SURNAME ➤						
DATE ➤						

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 11
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 May 7, 1975, as modified by filing dated May 26, 1976, 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.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: OCT 2 - 1976

OFFICE ➤						
SURNAME ➤						
DATE ➤						

ATTACHMENT TO LICENSE AMENDMENT NOS. 17 AND 11

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	TS-i
TS.3.3-1	TS.3.3-1
TS.3.3-5	TS.3.3-5
TS.3.3-5A	TS.3.3-5A
TS.3.4-1	TS.3.4-1
TS.3.4-2	TS.3.4-2
TS.3.6-1	TS.3.6-1
TS.3.6-3	TS.3.6-3
	TS.3.6-3A (New)
	TS.3.6-3B (New)
TS.3.6-5	TS.3.6-5
	TS.3.6-6 (New)
TS.3.8-2	TS.3.8-2
TS.3.8-3	TS.3.8-3
	TS.3.13-1 (New)
	TS.3.13-2 (New)
TS.4.1-1	TS.4.1-1
Table 4.1-1 page 1	Table 4.1-1 page 1
Table 4.1-2A	Table 4.1-2A
TS.4.4-4	TS.4.4-4
TS.4.4-5	TS.4.4-5
TS.4.4-8	TS.4.4-8
TS.4.4-9	TS.4.4-9
	TS.4.14-1 (New)
	TS.4.14-2 (New)

TECHNICAL SPECIFICATIONS

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2.2	Safety Limit, Reactor Coolant System Pressure	TS.2.2-1
2.3	Limiting Safety System Settings, Protective Instrumentation	TS.2.3-1
3.0	Limiting Conditions for Operation	TS.3.1-1
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3.13	Control Room Air Treatment System	TS.3.13-1
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4.7	Main Steam Stop Valves	TS.4.7-1
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4.9	Reactivity Anomalies	TS.4.9-1
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4.14	Control Room Air Treatment System	TS.4.14-1

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 analysis, shall be blocked and tagged in the proper position for injection. During power operation, changes in valve position will 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.

(1) Three component cooling pumps are operable.

(2) Four component cooling heat exchangers are operable.

(3) All valves, interlocks and piping associated with the above components, and required for the functioning of the system during accident conditions, are operable.

b. During startup operations or power operation either one of the following conditions of inoperability may exist provided startup operation is discontinued until operability is restored. The reactor shall be placed in the hot shutdown condition if during power operation operability is not restored within the time specified. The reactor shall be placed in the cold shutdown condition if operability is not restored within an additional 48 hours.

(1) One of the three component cooling pumps may be out of service for a period not to exceed 24 hours.

(2) One of the two component cooling heat exchangers associated with either unit may be out of service for a period not to exceed 48 hours.

D. Cooling Water System

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 provided by Specification 3.3 D.2. below.

a. Two diesel-driven cooling water pumps and two motor-driven cooling water pumps are operable.

b. All valves, interlocks, instrumentation, piping and fuel oil supply required for the functioning of the cooling water system during accident conditions are operable.

2. During startup operation or power operation, the following conditions of inoperability may exist provided startup operation is discontinued until operability is restored. The reactors shall be placed in the cold shutdown condition if operability is not restored within the stated time interval.

a. One diesel-driven cooling water pump may be inoperable for a period not to exceed seven days (total for both diesel-driven cooling water pumps during any consecutive 30 day period) provided:

(1) the operability of the other diesel-driven pump and its associated diesel generator are demonstrated immediately and at least once every 24 hours thereafter,

(2) the engineered safety features associated with that pump are operable; and

(3) both off-site power supply paths from the grid to the 4Kv emergency buses are operable.

(4) two motor-driven cooling water pumps shall be operable.

b. One of the two required motor driven cooling water pumps may be inoperable for a period not to exceed seven days provided:

(1) the operability of both diesel-driven cooling water pumps is demonstrated immediately and at least once every 24 hours thereafter.

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

2. The following auxiliary feedwater pump conditions exist.

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

3. Both power-operated relief valves for that reactor are operable.

4. 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.

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

6. Manual valves in the above systems that could (if one is improperly positioned) reduce flow below that assumed for accident analysis shall be blocked and tagged in the proper position for emergency use. During power operation, changes in valve position will be under direct administrative control.

7. 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.

8. 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.

9. 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 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. If 2.b. is not met within 7 days, one unit shall be placed in the cold shutdown condition.

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 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 (3). Isolation dampers are required in ventilation ducts that penetrate those rooms containing equipment needed for the accident.

3.6 CONTAINMENT SYSTEM

Applicability

Applies to the integrity of the containment system.

Objective

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

Specification

A. Containment System Integrity

1. Containment system integrity as defined in Specification TS.1 shall not be violated except when one of the following conditions exist: (a) the reactor is in the cold shutdown condition with the reactor vessel head installed, (b) the reactor is in the refueling shutdown condition with the vessel head removed, or (c) the fuel inside containment has not been used for power operation.

2. Deleted

3. Positive reactivity changes shall not be made by rod drive motion when containment system integrity is not intact except that rod drop tests and rod disconnecting and reconnecting may be done if the reactor is initially subcritical by at least 10% $\Delta k/k$.

9. The valves and actuation circuits that isolate the auxiliary building normal ventilation system following an accident shall be considered operable for containment integrity if the ventilation system can be manually isolated within 6 minutes following an accident and the inoperable components are repaired within 7 days.

10. The Auxiliary Building Special Ventilation System shall be considered operable only if the Turbine Building roof exhauster fans can be deenergized within 30 minutes of a loss of coolant accident.

B. Containment Internal Pressure

During power operation, if the internal pressure of the containment vessel exceeds 2 psig, the condition shall be corrected within eight hours or the reactor shall be placed in the cold shutdown condition.

C. Containment and Shield Building Air Temperature

During power operation, if the average temperature of the air in the containment vessel exceeds 44°F above the average temperature of the air in the shield building, the condition shall be corrected within 8 hours or the reactor shall be placed in the cold shutdown condition.

During the first fuel cycle of Unit 1, the average air temperatures will be determined from air temperature measurements at a number of locations in the shield building and the containment. For subsequent power operation of either unit, the average temperatures may be determined from other instrumentation determined to be representative of average air temperature as established and reported in accordance with item 3 of Table TS.6.7-1.

D. Containment Shell Temperature

During power operation, if the containment shell temperature becomes less than 30°F, the condition shall be corrected within 8 hours or the reactor shall be placed in the cold shutdown. During the first fuel cycle of Unit 1, the shell temperature will be measured at several locations. For subsequent power operation of either unit the minimum shell temperature may be determined as established and reported in accordance with item 3 of Table TS 6.7-1.

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, the Spent Fuel Pool 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 ≥99% DOP removal for particles having a mean diameter of 0.7 microns and ≥99% halogenated hydrocarbon removal.

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

   c. The Spent Fuel Pool Special Ventilation System fans only shall operate within ±10% of 4000 cfm per train.

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.

6. 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.

7. 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

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

This specification also prevents positive insertion of reactivity whenever containment integrity is not maintained if such addition would violate the respective shutdown margins. Effectively, the boron concentration must be maintained at a predicted concentration of 2100 ppm[1] during initial refueling and 2000 ppm during subsequent refueling, or more if the containment system is to be disabled with the vessel open.

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

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

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

The conservative calculation of off-site doses for the loss of coolant accident[3] [6] is based on an initial shield building annulus air temperature of 60°F and an initial containment vessel air temperature of 104°F The calculated period following LOCA for which the shield building annulus pressure is positive, and the calculated off-site doses are sensitive to this initial air temperature difference. The specified 44°F ture difference is consistent with the LOCA accident analysis[6].

The initial testing of inleakage into the shield building and the auxiliary building special ventilation zone (ABSVZ) has resulted in greater specified inleakage (Figure TS 4.4-1, change No. 1) and the necessity to deenergize the turbine building exhaust fans in order to achieve a negative pressure in the ABSVZ (TS 3.6.A.10). The staff's conservative calculation of doses for these conditions indicated that changing allowable containment leak rate from 0.5%/day to 0.25%/day would offset the increased leakage.[5]

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers for all emergency air treatment systems. The Charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The 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 adsrober provide a high level of assurance that the emergency air treatment systems will perform as predicted in the accident analyses.

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

References

(1) FSAR, Table 3.2.1-1
(2) FSAR, Section 5
(3) FSAR, Section 9.6.5 and Appendix G
(4) SAfety Evaluation Report, dated September 28, 1972, Section 15 and Supplement No. 2 dated April 30, 1973
(5) Letter to NSP dated November 29, 1973
(6) Letter to NSP dated September 16, 1974

6. Direct communication between the control room and the operating floor of the containment shall be available whenever changes in core geometry are taking place.

7. No movement of irradiated fuel in the reactor shall be made until the reactor has been subcritical for at least 100 hours.

8. The radiation monitors which initiate isolation of the Containment Purge System shall be tested and verified to be operable immediately prior to a refueling operation.

B. During fuel handling operations, the following conditions shall be satisfied:

1. No heavy loads will be transported over or placed in either part of the spent fuel pool when irradiated fuel is stored in that part.

2. Prior to spent fuel handling in the auxiliary building, tests shall be made to determine the operability of the spent fuel pool special ventilation system including the radiation monitors in the normal ventilation system that actuate the special system and isolate the normal systems.

3. Prior to fuel handling operations, fuel-handling cranes shall be load-tested for operability of limit switches, interlocks, and alarms.

4. When the spent fuel cask contains one or more fuel assemblies, it will not be suspended more than 30 feet above any surface until the fuel has decayed more than 90 days.

C. If any of the specified conditions in 3.8 A. or 3.8 B. above are not met, refueling or fuel-handling operations shall cease. Work shall be initiated to correct the violated conditions so that the specifications are met, and no operations which may increase the reactivity of the core shall be performed.

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 Pool Special Ventilation System provides ventilation of the spent fuel pool area in the event that high radiation is detected. The system shares exhaust fans and filters with the Containment In-Service Purge System.(4)

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

3.13 CONTROL ROOM AIR TREATMENT SYSTEM

Applicability

Applies to the operability of the Control Room Special Ventilation System.

Objective

To specify operability requirements for the Control Room Special Ventilation System.

Specification

A. Except as specified in Specification 3.13.C below, both trains of the Control Room Special Ventilation System and the diesel-generator required for their operation shall be operable at all times when containment integrity is required.

B. Each Control Room Special Ventilation System train shall satisfy the following operability requirements:

1. 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.

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

3. Fans shall be shown to operate within $\pm$10% of 4000 cfm.

C. From and after the date that one train of the Control Room Special Ventilation System is made or found to be inoperable for any reason, reactor operation or refueling operations are 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.

D. If these conditions cannot be met, reactor shutdown shall be initiated and the reactor shall be in cold shutdown within 36 hours and refueling operations shall be terminated within two hours.

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 applicalbe sections of ANSI N510 - 1975 standard as a procedural guideline only.

4.0 SURVEILLANCE REQUIREMENTS

Specific time intervals between tests may be adjusted plus or minus 25% to accommodate normal test schedules with the exception that, the intervals between tests scheduled for refueling shutdowns shall not exceed two years.(1)

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the minimum frequency and type of surveillance to be applied to plant equipment and conditions.

Specification

A. Calibration, testing, and checking of instrumentation channels and testing of logic channels shall be performed as specified in Table TS.4.1-1.

B. Equipment tests shall be conducted as specified in Table TS.4.1-2A.

C. Sampling tests shall be conducted as specified in Table TS.4.1-2B.

D. Whenever the plant condition is such that a system or component is not required to be operable the surveillance testing associated with that system or component may be discontinued. The asterisked items in Tables 4.1-1, 4.1-2A, and 4.1-2B are required at all times, however. Discontinued surveillance tests shall be resumed less than one test interval before establishing plant conditions requiring operability of the associated system or component, unless such testing is not practicable (i.e. nuclear power range calibration cannot be done prior to reaching power operation) in which case the testing will be resumed within 48 hours of attaining the plant condition which permits testing to be accomplished.

Basis

Channel Check

Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action, and a check supplements this type of built-in surveillance.

Based on experience in operation of both conventional and nuclear plant systems, when the plant is in operation, the minimum checking frequencies set forth are deemed adequate for reactor and steam system instrumentation.

(1) The interval between tests scheduled for refueling shutdowns may be 25 months for the first operating cycle of Unit 2.

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.	Diesel Fuel Supply	*Check Fuel Inventory	Daily	8.4.1
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.

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 Systems, Auxiliary Building Special Ventilation System, and the Spent Fuel Pool 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 (±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.

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

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

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

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

C. Containment Vacuum Breakers

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

D. Residual Heat Removal System

1. Those portions of the residual heat removal systems external to the isolation valves at the containment, shall be hydrostatically tested for leakage at 12-month intervals.

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

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

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

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

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.

The Spent Fuel Pool Special Ventilation System is a safeguards system which maintains a negative pressure in the spent fuel enclosure upon detection of high radiation in the area. 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.

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

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. Replacement charcoal adsorber and HEPA filters will be qualified in accordance with the intent 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.

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.

Periodic checking of the inlet heaters and associated controls for each train will provide assurance that the system has the capability of reducing inlet air humidity so that charcoal adsorber efficiency is enhanced.

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

References

(1) FSAR, Section5, and Appendix 14-C
(2) FSAR, Section 14, and Appendix G
(3) Safety Evaluation Report, Sections 6.2 and 15.0
(4) FSAR, Section 14
(5) FSAR, Section 14.3.6
(6) Letter to NSP from AEC dated November 29, 1973

4.14 CONTROL ROOM AIR TREATMENT SYSTEM TESTS

Applicability

Applies to the periodic testing requirements for the Control Room Special Ventilation System.

Objective

To specify tests for assuring the operability of the Control Room Special Ventilation System.

Specification

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 (±10%).

2. Automatic initiation of the Control Room Special Ventilation System shall be demonstrated with a simulated high radiation or Safety Injection signal.

B. 1. The tests of Specification 3.13.B. 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 effect 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 15 minutes 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 shall be replaced. Adsorbent in the tray removed for sampling shall be renewed. Any HEPA filters found defective shall 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 shall be performed as required for operational use. The determination of significant shall be made by the shift supervisor after consulting knowledgeable staff members.

Operation of each train of the system for 15 minutes 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 accident signals will assure that the system will start when require.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 17 AND 11 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 May 7, 1975, Northern States Power Company (NSP) requested an amendment to Facility License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2 (PINGP). This request was in response to our letter of January 8, 1975 regarding Limiting Conditions for Operation, Surveillance Requirements and Bases for the Control Room Special Ventilation and Emergency Charcoal Filter Systems. Also included in the request of May 7, 1975, were some miscellaneous items, identified as items 2, 3, 4, 8, 9 and 10 which are included in the evaluation, and some administrative changes, items 14, 15 and 16, which were completed by License Amendments 13 and 7 dated May 10, 1976.

DISCUSSION

The NRC letter to NSP dated January 8, 1975, indicated the need for additional Limiting Conditions for Operation and Surveillance Requirements to assure high confidence that the control room special ventilation and emergency charcoal filter systems would function reliably, when needed, at a degree of efficiency equal to or better than that assumed in the accident analysis. Model technical specifications based on Regulatory Guide 1.52 (June 1973) were provided for guidance. The model technical specifications were revised in the Spring of 1975 to include certain tests described in ANSI-N510 (1975). The revisions to the model technical specifications were discussed with the licensee, as were certain other changes in the May 7, 1975 request, which were required to meet the Regulatory requirements. The licensee has agreed with these changes and they will be incorporated in the amendments.

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The amendments to the licenses included changes in the technical specifications and bases for the emergency air treatment systems, as listed below:

1. The Shield Building Ventilation System,

2. The Auxiliary Building Special Ventilation System,

3. The Spent Fuel Pool Special Ventilation System, and

4. The Control Room Air Treatment Systems.

In addition, the miscellaneous corrections and clarifications, item 2 relating to the operating status of the reactor coolant system accumulators, item 3 relating to coolant water pump operability requirements, item 4 relating to the auxiliary feedwater pump valves, item 8 relating to miscellaneous testing program requirements for certain equipment, item 9 relating to minimum frequencies for nuclear power range tests and item 10 relating to minimum frequencies for equipment tests are included in this evaluation.

EVALUATION

These proposed amendments contain four major changes to the Technical Specifications related to installed filter systems. The first change consists of replacing paragraph 3.6.A.2. with a more detailed section (3.6.E). Paragraph 3.6.A.2. briefly defines the operability requirements for the Shield Building Ventilation Systems and the Auxiliary Building Special Ventilation System. Section 3.6.E defines the operability requirements for these systems in greater detail. Similar requirements appropriate for the Spent Fuel Pool Ventilation System have been added. This change details the range of test results required for each of the systems and also gives precise conditions of operability of these systems under which reactor operation is permissible.

Proper functioning of these systems would be assured by the completion of the specified tests. This in turn would provide a high level of assurance that the emergency charcoal filter systems would perform their safety functions in the event of an accident.

The second change incorporates a new section 3.13 which defines the operability requirements of the Control Room Air Treatment System. This new section includes the test results required to indicate proper operability of the system and also states the precise conditions of operability of this system under which reactor operation is permissible. The technical specifications did not previously have such limiting conditions.

The Control Room Special Ventilation System is designed to filter the control room atmosphere during accident conditions. The verification of the performance parameters combined with the qualification testing on new filters provide a high level of assurance that the Control Room Special Ventilation System will perform its safety function in the event of an accident.

The third change revises section 4.4.B.3 to include more detailed test requirements of the Emergency Charcoal Filter Systems. It also includes more detail concerning the acceptable test results.

These additional surveillance requirements would provide added assurance that the Emergency Charcoal Filter Systems would perform their safety functions in case of an accident.

The fourth change consists of the addition of a new section 4.14. This section includes specifications for testing frequency and the requirements for the test results for the Control Room Air Treatment System. Such specifications did not previously exist in the technical specifications.

Addition of these surveillance requirements would test the operability of the Control Room Special Ventilation System and would provide added assurance that it would perform its safety function in case of an accident.

We have reviewed the Limiting Conditions of Operation and Surveillance Requirements proposed by the licensee as modified by the staff and have concluded that the specifications are consistent with the Regulatory requirements for safety related filter systems. These specifications, as modified, provide reasonable assurance that the systems will function, when needed, as described in the Final Safety Analysis Report, as amended.

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On the basis of the above considerations the proposed changes to the technical specifications relating to installed filter systems are acceptable.

The evaluation of the miscellaneous items (items 2, 3, 4, 8, 9 and 10 of the May 7, 1975 application) is as follows:

Item 2, relating to the operating status of the reactor coolant system accumulators, was also a subject of Amendments 8 and 3 to the licenses dated June 11, 1975 and Amendment 6 to the License No. DPR-42 dated October 25, 1974. Amendments 8 and 3, the most recent amendments concerning the operating status of the accumulator, permitted an increase in the volume of borated water in the accumulator tanks but did not include provision for isolation of each accumulator when the pressurizer pressure is less than 1000 psi. The accumulator cannot be placed in service until a system pressure of 1000 psig is reached. The proposed wording of the requested change provides for such isolation and is generally consistent with the standard technical specifications. We have determined that the revised wording is consistent with correct method of operation of the accumulators and will not affect the operability of the accumulators under accident conditions. We find this change acceptable and it would not adversely affect the safe operation of the plant.

Item 3 requests a change which would provide further detail regarding the operability status of the cooling water pumps. The proposed amendment would add the requirement that two of the three motor driven pumps be operable if one diesel-driven cooling motor pump is out of service. In addition, the proposed specification would permit continual operation with only one motor driven pump operable for as long as seven days providing the operability of both diesel driven pumps is demonstrated. This proposed change is generally consistent with the Standard Technical Specifications. We conclude that the proposed change provides no compromise to the safety of the plant and is acceptable.

Item 4 requests a change that would add the operability requirement for the valves and piping associated with the auxiliary feedwater pumps which was previously not specified. Permission was also requested for necessary changes in motor operated valve position during Startup Operation. Such changes would be controlled administratively and represent valve movement normally required during startup. We have concluded that the proposed change is actually the way the valves must function during startup operation, represents no compromise in the safety of the plant and is acceptable.

Item 8, relating to surveillance testing program requirements for certain equipment, was also the subject of Amendment 6 to the license dated October 25, 1974. Amendment 6 required the testing of certain safety related systems after the plant was in a shutdown condition at which time systems were not required to be operable. This requires the performance of surveillance tests on systems at a time when they are not required to be operable and may in fact have been made inoperable by maintenance. Many of these tests require the plant to be at normal temperature and pressure or at power before they can be accomplished and therefore it is not possible to perform the tests while the plant is shutdown. Initial surveillance tests would be required for all discontinued tests before containment integrity is re-established. Discontinued surveillance tests would be resumed less than one test interval before establishing plant conditions requiring operability of the system or component, or, in case such testing is not practicable, the test would be performed within 48 hours of attaining the necessary plant conditions. We agree that the proposed change would provide the most rigorous requirements possible for re-establishing the surveillance tests following an extended shutdown. Continuing surveillance tests (asterisked items in tables 4.1-1, 4.1-2A and 4.1-2B) would be performed under all plant conditions. We conclude that the proposed change would not affect the safety of the plant and is acceptable.

Item 9, relating to the minimum frequencies for nuclear power range tests, corrects a typographical error in the original table which omitted the "heat balance" requirement. Item 10, relating to the minimum frequencies for equipment tests, requests that items in table 4.1-2A which are duplicated elsewhere in the Technical Specifications be deleted from the table. By deleting the duplicated items the chance of the operators mistaking the

requirements for additional rather than duplicate requirements is removed. In addition, the tests requirements will be located in a section of the Technical Specifications where they would be more readily located. We conclude that the change requested in items 9 and 10 are administrative in nature and do not affect the safe operation of the plant, therefore, they are acceptable.

ENVIRONMENTAL CONSIDERATIONS

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), than 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

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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 Nos. 17 and 11 to Facility Operating License Nos. DPR-42 and DPR-60, issued to the Northern States Power Company (the licensee), which revised the 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.

These amendments revised existing and added new limiting conditions for operation, surveillance requirements and bases for the Control Room Special Ventilation and Emergency Charcoal Filter Systems to the Technical Specifications for the facilities to enhance the efficiency and reliability of the systems. The amendments also revise several sections of the Technical Specifications to include the miscellaneous clarifications and corrections requested by the licensee.

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.

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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 May 7, 1975 and supplement thereto dated May 26, 1976, (2) Amendment Nos. 17 and 11 to License Nos. DPR-42 and DPR-60, respectively, and (3) the Commission's concurrently issued 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 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 14th day of October, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by:
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

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