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Docket Nos. 50-282
and 50-306

Mr. D. M. Musolf
Nuclear Support Services Department
Northern States Power Company
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

Dear Mr. Musolf:

The Commission has issued the enclosed Amendment Nos. 63 and 57 to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, in response to your application dated October 29, 1982.

The amendments revise the Technical Specifications (TS) concerned with Containment Purge issue B-24 and NUREG-0737 Item II.E.4.2, Addition of new Snubbers, and Event Monitoring Instrumentation, NUREG-0737, Items II.F.1.1 and II.F.1.2.

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

Sincerely,

Original signed by

Dominic C. DiIanni, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 63 to DPR-42
2. Amendment No. 57 to DPR-60
3. Safety Evaluation
4. Notice of Issuance

cc: w/enclosures
See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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Docket No. 50-282/50-306

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: NORTHERN STATES POWER COMPANY, Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2

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

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

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

☐ Notice of Availability of Applicant's Environmental Report.

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

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

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

☐ Notice of Limited Work Authorization.

☐ Notice of Availability of Safety Evaluation Report.

☐ Notice of Issuance of Construction Permit(s).

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

☒ Other: Amendment Nos. 63 and 57
Referenced documents have been provided PDR.

Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

| | | | | | | |
|---|---|---|---|---|---|---|
| OFFICE | ORB#3:DL | | | | | |
| SURNAME | PMKreutzer/pn | | | | | |
| DATE | 3/23/83 | | | | | |

NRC FORM 102 7—79

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cc:

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Federal Activities Branch
Region V Office
ATTN: Regional Radiation
Representative
230 South Dearborn Street
Chicago, Illinois 60604

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. 63
License No. DPR-42

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

   A. The application for amendment by Northern States Power Company (the licensee) dated October 29, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-42 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 63 , 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

Robert A. Clark

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 23, 1983

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. 57
License No. DPR-60

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

   A. The application for amendment by Northern States Power Company (the licensee) dated October 29, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-60 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 57 , 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

Robert A. Clark

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 23, 1983

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NOS. 63 AND 57 TO FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60

DOCKET NOS. 50-282 AND 50-306

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

| Remove | Insert |
|---|---|
| TS.i | TS.i |
| TS.iii | TS.iii |
| TS.3.6-2 | TS.3.6-2 |
| TS.3.8-1 | TS.3.8-1 |
| TS.3.8-4 | TS.3.8-4 |
| Table TS.3.12-1 (pg. 1 of 8) | Table TS.3.12-1 (pg. 1 of 8) |
| Table TS.3.12-1 (pg. 2 of 8) | Table TS.3.12-1 (pg. 2 of 8) |
| Table TS.3.12-1 (pg. 7 of 8) | Table TS 3.12-1 (pg. 7 of 8) |
| Table TS.3.12-1 (pg. 8 of 8) | Table TS.3.12-1 (pg. 8 of 8) |
| TS.3.15-1 | TS.3.15-1 |
| | TS.3.15-2 |
| Table TS.3.15-1 | Table TS.3.15-1 |
| | Table TS.3.15-2 |
| Table TS.4.1-1 (pg. 4 of 5) | Table TS.4.1-1 (pg. 4 of 5) |
| TS.4.5-2 | TS.4.5-2 |

# TECHNICAL SPECIFICATIONS

## TABLE OF CONTENTS



Unit 1 - Amendment No. ~~50~~, ~~59~~, 63
Unit 2 - Amendment No. ~~44~~, ~~53~~, 57

# TECHNICAL SPECIFICATIONS

## LIST OF TABLES

4. Positive reactivity changes shall not be made by boron dilution when containment system integrity is not intact unless the boron concentration in the reactor is maintained $\geq$ 2100 ppm for the initial refueling and $\geq$ 2000 ppm for subsequent refuelings.

5. The vacuum breaker system shall be considered operable for containment system integrity when both valves in each of two vacuum breakers, including actuating and power circuits, are operable or when one vacuum breaker is daily demonstrated as operable and the other has been inoperable for no more than 7 days under conditions for which containment integrity is required.

6. Automatic containment isolation valves listed in Table TS.4.4-1 shall be considered operable for containment system integrity when all automatic isolation valves, including actuation circuits, for each penetration are operable or the inoperable valve is deactivated in the closed position, or at least one valve in each penetration having an inoperable valve is locked closed.

7. a. The 36-inch containment purge system double gasketed blind flanges shall be installed whenever the reactor is above cold shutdown.

   b. The 18-inch containment inservice purge system double gasketed blind flanges shall be installed whenever the reactor is above cold shutdown except as noted below.

   c. The inservice purge system may be operated above cold shutdown when required for safe plant operation if the following conditions are met:

      1. The debris screens are installed on the supply and exhaust ducts in containment.

      2. Both valves shall satisfactorily pass a local leak rate test prior to use.

      3. The two automatic primary containment isolation valves and the automatic shield building ventilation damper in each duct that penetrates containment shall be operable, including instruments and controls associated with them.

      4. The blind flanges (i.e., 42B (53 in Unit 2) and 43A (52 in Unit 2) shall be reinstalled and satisfactorily pass a local leak rate test, each time after the inservice purge system is used.

8. During maintenance, construction and testing activities, containment integrity is considered intact if the auxiliary building special vent zone boundary is opened intermittently, provided such openings are under direct administrative control and can be reduced to less than 10 square feet within 6 minutes following an accident.

## 3.8 REFUELING AND FUEL HANDLING

Applicability

Applies to operating limitations during fuel-handling and refueling operations.

Objectives

To ensure that no incident could occur during fuel handling and refueling operations that would affect public health and safety.

Specification

A. During refueling operations the following conditions shall be satisfied:

1. The equipment hatch and at least one door in each personnel air lock shall be closed. In addition, at least one isolation valve shall be operable or locked closed in each line which penetrates the containment and provides a direct path from containment atmosphere to the outside.

2. Radiation levels in fuel handling areas, the containment and the spent fuel storage pool areas shall be monitored continuously.

3. The core subcritical neutron flux shall be continuously monitored by at least two neutron monitors, each with continuous visual indication in the control room and one with audible indication in the containment, which are in service whenever core geometry is being changed. When core geometry is not being changed, at least one neutron flux monitor shall be in service.

4. During reactor vessel head removal and while loading and unloading fuel from the reactor, the minimum boron concentration of 2000 ppm shall be maintained in the reactor coolant system. The required boron concentration shall be verified by chemical analysis daily.

5. During movement of fuel assemblies or control rods out of the reactor vessel, at least 23 feet of water shall be maintained above the reactor vessel flange. The required water level shall be verified prior to moving fuel assemblies or control rods and at least once every day while the cavity is flooded.

6. At least one residual heat removal pump shall be operable and running. The pump may be shut down for up to one hour to facilitate movement of fuel or core components.

7. If the water level above the top of the reactor vessel flange is less than 20 feet, except for control rod latching and unlatching operations, both residual heat removal loops shall be operable.

8. If Sepcification 3.8.A.6 or 3.8.A.7 cannot be satisfied, all fuel handling operations in containment shall be suspended, the containment integrity requirements of Specification 3.8.A.1 shall be satisfied, and no reduction in reactor coolant boron concentration shall be made.

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.

During movement of irradiated fuel assemblies or control rods, a water level of 23 feet is maintained to provide sufficient shielding.

The water level may be lowered to the top of the RCCA drive shafts for latching and unlatching. The basis for this allowance is (1) the refueling cavity pool has sufficient level to allow time to initiate repairs or emergency procedures to cool the core, (2) during latching and unlatching the level is closely monitored because the activity uses this level as a reference point, (3) the time spent at this level is minimal.

<u>References</u>

(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

TABLE TS.3.12-1 (Page 1 of 8)

## SAFETY RELATED SNUBBERS

| Snubber No. | Location | Elevation | Accessible or Inaccessible (A or I) | Snubbers Especially Difficult to Remove | In High Radiation Area During Shutdown |
|---|---|---|---|---|---|
| | UNIT I | | | | |
| AFSH-22 A&B | Main and Aux- | 773'-4½" | A | | |
| AFSH-36 | iliary Steam | 745'-7½" | A | | |
| AFSH-39 | | 699'-10½" | A | | |
| AFSH-48 | | 699'-6½" | A | | |
| MSDH-25 A&B | | 736'-6-7/16" | A | | |
| MSDH-26 A&B | | 756'-7½" | A | | |
| MSDH-29 | | 756'-7½" | A | | |
| MSDH-30 | | 736'-6-7/16" | A | | |
| MSH-48 A&B | | 739'-1-11/16" | A | | |
| MSH-62 A&B | | 735'-6" | A | | |
| MSH-63 | | 756'-0" | A | | |
| MSH-64 | | 743'-0" | A | | |
| MSH-65 | | 748'-0" | A | | |
| MSH-66 | | 753'-0" | A | | |
| MSH-67 | | 743'-0" | A | | |
| MSH-68 A&B | | 755'-8" | A | | |
| MSH-69 A&B | | 748'-0" | A | | |
| MSH-101 | | 729'-0" | A | | |
| MSH-102 | | 735'-0" | A | | |
| MSH-103 A&B | | 737'-0" | A | | |
| | UNIT II | | | | |
| AFSH-2 | Main and Aux- | 749'-4" | A | | |
| AFSH-19 | iliary Steam | 745'-7½" | A | | |
| AFSH-20 | | 745'-7½" | A | | |
| AFSH-24 | | 745'-6" | A | | |
| AFSH-29 A&B | | 721'-1-9/16" | A | | |
| AFSH-33 | | 707'-5" | A | | |
| AFSH-39 | | 696'-6½" | A | | |
| AFSH-40 | | 696'-6½" | A | | |
| AFSH-44 | | 750'-7½" | A | | |
| AFSH-46 | | 750'-7" | A | | |
| MSDH-17 | | 739'-0" | A | | |
| MSDH-18 | | 759'-0" | A | | |
| MSDH-19 | | 739'-0" | A | | |
| MSDH-20 | | 759'-0" | A | | |

Unit 1 - Amendment No. ~~14~~, ~~49~~, ~~50~~, 63
Unit 2 - Amendment No. ~~8~~, ~~43~~, ~~44~~, 57

SAFETY RELATED SNUBBERS

| Snubber No. | Location | Elevation | Accessible or Inaccessible (A or I) | Snubbers Especially Difficult to Remove | In High Radiation Area During Shutdown |
|---|---|---|---|---|---|
| | UNIT II | | | | |
| MSH-23 A&B | Main and Auxiliary Steam | 739'-1-3/16" | A | | |
| MSH-54 A&B | | 756'-0-1/16" | I | | |
| MSH-75 | | 744'-0" | A | | |
| MSH-76 A&B | | 748'-0" | A | | |
| MSH-77 | | 748'-0" | A | | |
| MSH-78 | | 743'-0" | A | | |
| MSH-79 | | 753'-0" | A | | |
| MSH-80 | | 755'-0" | A | | |
| MSH-81 A&B | | 735'-9" | A | | |
| MSH-82 A&B | | 755'-8" | A | | |
| MSH-83 | | 761'-13/16" | I | | |
| MSH-101 | | 727'-0" | A | | |
| MSH-102 | | 734'-0" | A | | |
| MSH-103A&B | | 736'-0" | A | | |
| | UNIT I | | | | |
| RHRRH-5 | Safety Injection | 723'-4½" | I | | |
| RHRRH-41 | | 698'-11" | I | | |
| RHRRH-58 | | 670'-0" | A | | |
| RHRRH-60 | | 670'-0" | A | | |
| RPCH-160 | | 718'-½" | I | | |
| RSIH-92 | | 714'-11" | I | | |
| RSIH-93 | | 714'-11" | I | | |
| RSIH-95 | | 711'-2" | I | | |
| RSIH-96 | | 711'-2" | I | | |
| RSIH-98 | | 701'-2" | I | | |
| RSIH-163 | | 717'-9" | I | | |
| RSIH-167 | | 717'-9" | I | | |
| RSIH-413 A&B | | 722'-8" | A | | |
| RISH-414 | | 716'-10" | I | | |
| RISH-442 | | 717'-9½" | I | | |
| RSIH-469 | | 707'-6½" | I | | |
| RSIH-476 | | 707'-1-3/4" | I | | |
| SIRH-9 | | 737'-0" | I | | |
| SIRH-11 | | 718'-6" | I | | |
| SIRH-17 | | 730'-0" | I | | |
| SIRH-18 | | 730'-0" | I | | |
| SIRH-22 | | 711'-4" | I | | |
| SIRH-23 A&B | | 711'-4" | I | | |
| SIRH-26 | | 705'-0" | I | | |

## SAFETY RELATED SNUBBERS

| Snubber No. | Location | Elevation | Accessible or Inaccessible (A or I) | Snubbers Especially Difficult to Remove | In High Radiation Area During Shutdown |
|---|---|---|---|---|---|
| | UNIT II | | | | |
| RCVCH-1396 | Chemical & Vol Control | 702'-10" | I | | |
| RCVCH-1505 | | 708'-6" | I | | |
| RCVCH-1513 | | 710'-1" | I | | |
| RCVCH-1524 | | 719'-1" | I | | |
| RCVCH-1574 | | 721'-0" | I | | |
| RCVCH-1668 | | 705'-5" | I | | |
| RCVCH-1373 | | 722'-11" | I | | |
| RCVCH-1389 | | 706'-1" | I | | |
| RRCH-253 | | 704'-4" | I | | |
| RRCH-255 | | 704'-8" | I | | |
| RRCH-261 | | 707'-2" | I | | |
| RRCH-288 | | 707'-2" | I | | |
| RRCH-291 | | 704'-6" | I | | |
| RRCH-292 | | 704'-7" | I | | |
| | UNIT I | | | | |
| CCH-304 | Comp Cooling | 717'-7" | A | | |
| CCH-373 | | 712'-4" | A | | |
| CCH-376 A&B | | 700'-5" | A | | |
| CCH-377 | | 703'-0" | A | | |
| CCH-378 | | 708'-4" | A | | |
| CCH-380 | | 670'-8" | A | | |
| CCH-381 A&B | | 671'-4" | A | | |
| CCH-397 | | 699'-3" | A | | |
| CCH-398 A&B | | 671'-4" | A | | |
| | UNIT II | | | | |
| CCH-161 | Comp Cooling | 717'-7" | A | | |
| CCH-166 | | 719'-11" | A | | |
| CCH-167 | | 720'-0" | A | | |
| CCH-172 | | 720'-0" | A | | |
| CCH-173 | | 708'-5" | A | | |
| CCH-176 | | 705'-3" | A | | |
| CCH-179 A&B | | 671'-4" | A | | |
| CCH-180 | | 670'-8" | A | | |
| CCH-181 | | 708'-4" | A | | |
| CCH-182 | | 704'-2" | A | | |
| CCH-185 A&B | | 671'-4" | A | | |
| CCH-186 | | 670'-10" | A | | |
| | UNIT I | | | | |
| RCSH-81 | Containment Spray | 760'-9" | I | | |
| RCSH-82 | | 760'-8" | I | | |
| RCSH-83 A&B | | 732'-1" | I | | |
| | UNIT II | | | | |
| CSH-75 A&B | Containment Spray | 731'-10" | I | | |
| CSH-76 | | 752'-7" | I | | |
| CSH-79 | | 751'-9" | I | | |
| CSH-82 A&B | | 731'-11" | I | | |
| CSH-83 | | 767'-2" | I | | |
| CSH-84 | | 767'-2" | I | | |
| CSH-210 | | 698'-0" | I | | |
| CSH-215 | | 698'-0" | A | | |
| CSH-224 | | 710'-6" | A | | |

## SAFETY RELATED SNUBBERS

| Snubber No. | Location | Elevation | Accessible or Inaccessible (A or I) | Snubbers Especially Difficult to Remove | In High Radiation Area During Shutdown |
|---|---|---|---|---|---|
| | UNIT I | | | | |
| RRHH-20 | RHR | 704'-3" | A | | |
| RRHH-62 | | 705'-10" | A | | |
| | UNIT II | | | | |
| CVCRH-6 | RHR | 711'-0" | I | | |
| RRHH-21 | | 704'-6" | A | | |
| | UNIT II | | | | |
| ZX-PSCH-127 | ZX | 707'-0" | A | | |

## 3.5 EVENT MONITORING INSTRUMENTATION

Applicability

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

Objectives

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

A. Specification - Process Monitors

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

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

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

B. Specification - Radiation Monitors

1. The event monitoring instrumentation channels specified in Table TS.3.15-2 shall be Operable.

2. With the number of Operable event monitoring instrumentation channels less than the Required Total Number of Channels shown on Table TS.3.15-2, either restore the inoperable channels to Operable status within seven days, or prepare and submit a Special Report to the Commission pursuant to Technical Specification 6.7.B.2 within the next 30 days outlining the action taken, the cause of the inoperability, the plans and the schedule for restoring the system to Operable status.

3. With the number of Operable event monitoring instrumentation channels less than the Minimum Channels Operable requirement of Table TS.3.15-2, initiate the preplanned alternate method of monitoring the appropriate parameters in addition to submitting the report required in (2) above.

Basis

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

# TABLE TS.3.15-1
## EVENT MONITORING INSTRUMENTATION - PROCESS

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

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

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

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

TABLE TS.3.15-2

EVENT MONITORING INSTRUMENTATION - RADIATION

| Instrument | Required Total No. of Channels | Minimum Channels Operable |
|---|---|---|
| 1. Containment Radiation Monitors (Hi Range) | 2 | 1 |
| 2. Steam Relief Activity Monitors | 1/steam line | 1/steam line |
| 3. High Range Shield Building Ventilation Monitors | 1 | 1 |

Table TS.4.1-1
(Page 4 of 5)

Unit 1 - Amendment No. 55, 59, 61, 63
Unit 2 - Amendment No. 49, 53, 55, 57

| | Channel Description | Check | Calibrate | Functional Test | Response Test | Remarks |
|---|---|---|---|---|---|---|
| 27. | Turbine Overspeed Protection Trip Channel | NA | R | M | NA | |
| 28. | Deleted | | | | | |
| 29. | Deleted | | | | | |
| 30. | Deleted | | | | | |
| 31. | Seismic Monitors | R | R | NA | NA | Includes those reported in Item 4 of Table TS.6.7-1 |
| 32. | Coolant Flow - RTD Bypass Flowmeter | S | R | M | NA | |
| 33. | CRDM Cooling Shroud | S | NA | R | NA | FSAR page 3.2-56 |
| 34. | Reactor Gap Exhaust Air Temperature | S | NA | R | NA | FSAR page 5.4-2 |
| 35a. | Post-Accident Monitoring Instruments | M | R | NA | NA | Includes all those in FSAR Table 7.7-2 and Table TS.3.15-1 not included elsewhere in this Table |
| b. | Post-Accident Monitoring Radiation Instruments | D | R | M | NA | Includes all those in Table TS.3.15-2 |
| 36. | Steam Exclusion Actuation System | W | R | M | NA | See FSAR Appendix I, Section I.14.6 |
| 37. | Overpressure Mitigation System | NA | R | R | NA | Instrument Channels for PORV Control Including Overpressure Mitigation System |
| 38. | Degraded Voltage 4 KV Safeguard Busses | NA | R | M | NA | |
| 39. | Loss of Voltage 4 KV Safeguard Busses | NA | R | M | NA | |

3. Containment Fan Coolers

Each fan cooler unit shall be tested during each reactor refueling shutdown to verify proper operation of all essential features including low motor speed, cooling water valves, and normal ventilation system dampers. Individual unit performance will be monitored by observing the terminal temperatures of the fan coil unit and by verifying a cooling water flow rate of greater than or equal to 900 gpm to each fan coil unit.

4. Component Cooling Water System

a. System tests shall be performed during each reactor refueling shutdown. Operation of the system will be initiated by tripping the actuation instrumentation.

b. The test will be considered satisfactory if control board indication and visual observations indicate that all components have operated satisfactorily.

5. Cooling Water System

a. System tests shall be performed at each refueling shutdown. Tests shall consist of an automatic start of each diesel engine and automatic operation of valves required to mitigate accidents including those valves that isolate non-essential equipment from the system. Operation of the system will be initiated by a simulated accident signal to the actuation instrumentation. The tests will be considered satisfactory if control board indication and visual observations indicate that all components have operated satisfactorily and if cooling water flow paths required for accident mitigation have been established.

b. At least once each 18 months, subject each diesel engine to a thorough inspection in accordance with procedures prepared in conjunction with the manufacturer's recommendations for this class of standby service.

B. Component Tests

1. Pumps

a. The safety injection pumps, residual heat removal pumps and containment spray pumps shall be started and operated at intervals of one month. Acceptable levels of performance shall be that the pumps start and reach their required developed head on minimum recirculation flow and the control board indications and visual observations indicate that the pumps are operating properly for at least 15 minutes.

b. A test consisting of a manually-initiated start of each diesel engine, and assumption of load within one minute, shall be conducted monthly.

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

# RELATED TO AMENDMENT NOS. 63 AND 57

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

# NORTHERN STATES POWER COMPANY

# PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NOS. 1 AND 2

# DOCKET NOS. 50-282 AND 50-306

## Introduction

By letter dated October 29, 1982, Northern States Power Company (NSP or the licensee) requested amendments to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2 (PINGP). The requested amendments propose changes in the Technical Specifications (TS) in the following areas.

1. TS 3.6 and Table TS 4.4-1, Containment Ventilation System

2. TS 3.8, Refueling and Fuel Handling

3. TS 3.1.2, Snubbers

4. TS 3.15, Event Monitoring Instrumentation

5. TS 4.5, Engineered Safety Feature

## 1. TS 3.6 and Table TS 4.4-1, Containment Ventilation System

By our letter dated September 9, 1982 related to the containment purge issue B-24 and NUREG-0737 Item II.E.4.2, we requested the licensee to propose changes to the TS as discussed in our safety evaluation. This proposed TS is in response to our request. Proposed changes to Table TS 4.4-1, which are found acceptable, are addressed in our safety evaluation (note part 2 IV) issued by letter dated February 23, 1983 for Amendment Nos. 62 and 56. The proposed TS change TS 3.6 requires the double gasketed blind flanges at the 36 inch and the 18 inch purge systems to be in place whenever the reactor is above cold shutdown. In addition, the flange seals are to be subjected to a Type B leak test as specified in 10 CFR Part 50 Appendix J when the blind flanges are removed during cold shutdown and during each refueling shutdown prior to returning the reactor to power operation. The proposed TS also requires that a Type C leak test, as prescribed by 10 CFR Part 50 Appendix J be performed on the purge valves prior to operating the 18 inch containment inservice purge system for plant safety reasons.

Our technical basis for requiring the blind flanges and testing of the purge valves in the 18 inch purge system is discussed in Sections II.B and II.C.2 of our Safety Evaluation issued to the licensee by letter dated September 9, 1982.

We have reviewed the proposed TS and except for leak testing the blind flanges after purging (3.6.A.7.c.4) we find the licensee has adequately addressed the request in our letter dated September 9, 1982. The leak rate testing requirement of the blind flanges after purging was inadvertently omitted by the licensee. The proposed change was modified to include this omission which was discussed with and agreed to by the licensee. On this basis, we find the proposed TS change acceptable.

2. TS 3.8 - Refueling and Fuel Handling

The existing TS 3.8.A.7 is as follows:

> If the water level above the top of the reactor vessel flange is less than 20 feet, except when the cavity is being drained for head replacement or control rod latching and unlatching operations, both residual heat removal loops shall be operable.

The licensee proposes to delete the phrase "when the cavity is being drained for head replacement". By deleting this phrase, the licensee is required to have both residual heat removal loops operable during periods when the reactor vessel head is being reinstalled and the water level is less than 20 feet above the reactor vessel flange. Having both residual heat removal loops operable when there is less than 20 ft of water above the reactor vessel flange ensures that a single failure of one operating loop will not result in a complete loss of residual heat removal capability. The licensee's purpose for requesting this change is 1) to clarify the requirement of operability of both residual heat removal during the control rod latching and unlatching operation and the reactor head replacement operation since these two operations are performed in series and the elapsed time between them could be lengthy; and 2) the requirements of the proposed change are consistent with those of the standard technical specifications and represent an improvement in the requirements over that presently existing in the technical specifications.

Based on our review, we agree with the licensee that the proposed change clarifies the operability status of the residual heat removal loops during reactor vessel head replacement and control latching and unlatching operations and the change would result in meeting the objectives of the standard TS.

Based on this evaluation, we find the proposed change does not reduce the level of plant safety and does clarify the requirement. Therefore we find the proposed change acceptable.

In addition, the licensee proposes to correct a typographical error appearing in the basis of TS 3.8 regarding the adsorber and the discussion of charcoal efficiency testing. We have reviewed this typographical correction and find that the proposed correction does not change any of the TS requirements nor the intent of the statement that is affected by this proposed correction. Since this change serves only to correct the error as described above, it does not involve significant new safety information of a type not considered by a previous Commission safety review of the facility. We therefore find this proposed change to correct the typographical error acceptable.

3. TS Table 3.12-1 - Safety Related Snubbers

The licensee proposes additions and corrections to TS Table 3.12-1 which lists hydraulic snubbers related to safety systems. TS 3.12 establishes the operability requirements of the safety related snubbers to ensure structural integrity of safety related systems when the reactor is above cold shutdown. TS Table 3.12-1 identifies the snubbers, by number and plant location, that must meet the operability requirements. The proposed change includes corrections that resulted from a plant audit to have the table agree with as-built field conditions. This phase of the proposed change merely serves to clarify and to assure that there will not be any misinterpretations when the table is compared to the actual snubber locations throughout the plant.

The proposed additonal snubbers shown in Table TS 3.12-1 resulted from changes to the pipe support systems related to the issues raised in I&E Bulletins 79-02 and 79-14. Specifically, I&E Bulletin 79-02 requested all licensees to review pipe support systems with concrete expansion anchor bolts to assure the designs meet the criterion for seismic category I systems as defined by Regulatory Guide 1.29, "Seismic Design Classification, Revision 1," dated August 1973, or as defined in the applicable FSAR. I&E Bulletin 79-14 requested all licensees to verify that seismic analysis applies to the actual field configurations of safety-related pipe systems. Therefore results of these reviews identified additional snubbers associated with the main steam lines from the steam generators for both units that must be operable to assure safe reactor operations.

Based on the above, we conclude that this proposed change improves the level of plant safety and therefore is acceptable.

4. TS 3.15 - Event Monitoring Instrumentation

The licensee proposes to place limiting conditions for operation and surveillance requirements in the TS for the radiation monitoring instrumentations as required by NUREG-0737 Items II.F.1.1 and II.F.1.2. A post implementation review of these two items was performed by Region III and Report Nos. 50-282/8203, 50-306/8203, 50-282/8207 and 50-306/8207 addressing these items were issued on February 25 and June 4, 1982, respectively. These reports concluded that the licensee meets the objectives of NUREG-0737 Items II.F.1.1 and II.F.1.2. By letter dated November 9, 1981 we provided the licensee a model TS that could be used as guidance in preparing the TSs for Items II.F.1.1 and II.F.1.2. Based on our review, we conclude that the licensee's proposed TS changes, except for the surveillance requirements in TS Table 4.1-1, are within these guidelines. The licensee proposed surveillance for the radiation monitoring instrumentation that requires a monthly check and a calibration check during each refueling outage but no functional test nor any response test. We agree with the licensee that a response test does not serve a useful purpose and therefore is not applicable. However, in order to meet the objective of the model TS, the licensee's proposed change was modified to include a monthly functional test and a daily check. This modification to the proposed change was discussed with and agreed to by the licensee. On this basis, we find the proposed TSs related to the limiting conditions for operations and the surveillance requirements for NUREG-0737, Items II.F.1.1 and II.F.1.2 acceptable.

5. TS 4.5 - Engineered Safety Feature

The licensee proposes to revise TS 4.5.A.5(b) to read, "At least once every 18 months, subject each diesel engine to a thorough inspection in accordance with procedures prepared in conjunction with the manufacturer's recommendations for this class of standby service". The existing TS states that each diesel engine shall be inspected at each refueling shutdown. The purpose of the proposed TS change is to clarify the inspection interval since the Prairie Island Nuclear Generating Plant has two refueling shutdowns each year. The existing requirement can be interpreted to mean that each diesel engine would be inspected every six months since they service both units. Such an interpretation is in conflict with TS 4.6 "Periodic Testing of Emergency Power System" which requires an 18 month inspection interval for this component. The 18 month periodic testing of the diesel generators and engines is a standard requirement appearing in our standard TS. In addition the diesel engines are tested at monthly intervals and during each refueling shutdown to satisfy the requirements of TS 4.5.A.5(a) and 4.5.B.1(b). Therefore, the engines are tested 14 times per year which would give early indication of a potential problem.

We have reviewed this change and find that neither the TS requirements nor objectives of what now exists in the TS are affected by the proposed change. Since this proposed change serves to correct a conflict that exists in the TS as described above, it does not involve significant new safety information of a type not considered by a previous Commission safety review of the facility. We therefore find the proposed change 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 an accident previously evaluated, do not create the possibility of an accident of a type different from any evaluated previously, and do not involve a significant reduction in a margin of safety, 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: March 23, 1983

Principal Contributor:
D. C. DiIanni

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-282 AND 50-306

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 63 and 57 to Facility Operating License Nos. DPR-42 and DPR-60 issued to Northern States Power Company (the licensee), which revised Technical Specifications for operation of Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2 (the facilities) located in Goodhue County, Minnesota. The amendments are effective as of the date of issuance.

The amendments revise the Technical Specifications (TS) concerned with Containment Purge issue B-24 and NUREG-0737 Item II.E.4.2, Addition of new Snubbers, and Event Monitoring Instrumentation, NUREG-0737, Items II.F.1.1 and II.F.1.2.

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

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

For further details with respect to this action, see (1) the application for amendments dated October 29, 1982, (2) Amendment Nos. 63 and 57 to License Nos. DPR-42 and DPR-60, 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, 300 Nicollet Mall, Minneapolis, Minnesota 55401. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 23rd day of March, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

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

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

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Docket No. 50-282 and 50-306

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: NORTHERN STATES POWER COMPANY, Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies ( 6 ) of the Notice are enclosed for your use.

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

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

☐ Notice of Availability of Applicant's Environmental Report.

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

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

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

☐ Notice of Limited Work Authorization.

☐ Notice of Availability of Safety Evaluation Report.

☐ Notice of Issuance of Construction Permit(s).

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

X☐ Other: Notice of Consideration. Please contact Patty on X24556 with 30-day date after publication in FR to be entered in 3rd paragraph on page 3 of Notice.

Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

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| OFFICE→ | ORB#3:DL | | | | | |
| SURNAME→ | PMKreutzer/pn | | | | | |
| DATE→ | 7/8/83 | | | | | |

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