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

Dear Mr. Anderson:

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF SUBJECT:

AMENDMENTS RE: FIRE PROTECTION AND DETECTION SYSTEMS - LIMITING

CONDITIONS FOR OPERATION (TAC NOS. M89962 AND M89963)

The Commission has issued the enclosed Amendment No. 120 to Facility Operating License No. DPR-42 and Amendment No. 113 to the Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated July 11, 1994, as supplemented April 18, 1995. This application superseded your February 10, 1993, license amendment request.

The amendments revise TS Section 3.14 (Fire Protection and Detection Systems -Limiting Conditions for Operation), TS Section 4.16 (Fire Detection and Protection Systems - Surveillances), TS Sections 6.1 (Administrative Controls, Organization), and TS Section 6.2 (Administrative Controls, Review and Audit) to relocate the fire protection program elements from the TS and incorporate, by reference, the NRC-approved Fire Protection Program and major commitments, including the fire hazards analysis, into the Updated Safety Analysis Report. In addition, the amendments revise the Operating Licenses to include the NRC's standard fire protection license condition. These changes are made in accordance with the guidance provided in Generic Letter (GL)  $\bar{8}6-10$ , "Implementation of Fire Protection Requirements," and GL 88-12, "Removal of Fire Protection Requirements from Technical Specifications."

A copy of our related Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely.

Original signed by

Beth A. Wetzel, Project Manager Project Directorate III-1 Division of Reactor Projects - III/IV Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosures:

1. Amendment No. 120 to DPR-42

Amendment No. 113 to DPR-60

Safety Evaluation

cc w/encl.: See next page

DOCUMENT NAME: G:\WPDOCS\PRAIRIE\P189962.AMD

"E" = Copy with stachment/enclosure "N" = No copy To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure

E BC:SPLE IL OGC C D:PD31 ٤ LA:PD31 | E | (A)PM:PD3,h/L\_TSB OFFICE JHannon 454 CJamerson CThomas:jkd CGrimes CMcCracken STurk NAME 19/6/95 9/29/95 DATE 10/3/95 9/1\%\/95

OFFICIAL RECORD COPY Signature block changed to reflect current Project Manager. 795-14/

9510200198 951006 PDR ADOCK 05000282

Prairie Island Nuclear Generating
Plant

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

cc:

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

Site General Manager Prairie Island Nuclear Generating Plant Northern States Power Company 1717 Wakonade Drive East Welch, Minnesota 55089

Adonis A. Neblett Assistant Attorney General Office of the Attorney General 455 Minnesota Street Suite 900 St. Paul, Minnesota 55101-2127

U.S. Nuclear Regulatory Commission Resident Inspector's Office 1719 Wakonade Drive East Welch, Minnesota 55089-9642

Regional Administrator, Region III U.S. Nuclear Regulatory Commission 801 Warrenville Road Lisle, Illinois 60532-4351

Mr. Jeff Cole, Auditor/Treasurer Goodhue County Courthouse Box 408 Red Wing, Minnesota 55066-0408

Kris Sanda, Commissioner Department of Public Service 121 Seventh Place East Suite 200 St. Paul, Minnesota 55101-2145

Site Licensing
Prairie Island Nuclear Generating
Plant
Northern States Power Company
1717 Wakonade Drive East
Welch, Minnesota 55089

DATED: October 6, 1995

AMENDMENT NO. 120 TO FACILITY OPERATING LICENSE NO. DPR-42-PRAIRIE ISLAND UNIT 1 AMENDMENT NO. 113 TO FACILITY OPERATING LICENSE NO. DPR-60-PRAIRIE ISLAND UNIT 2

Pocket File
PUBLIC
PUBLIC
PDIII-1 Reading
J. Roe
C. Jamerson
B. Wetzel
C. Thomas
OGC-WF
G. Hill (4)
C. Grimes, O-11F23
C. McCracken
ACRS
M. Jordan, RIII
SEDB

cc: Plant Service list



# UNITED STATES NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

#### NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

### PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 1

### AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 120 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 July 11, 1994, as supplemented April 18, 1995, 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 following sections under paragraph 2.C of Facility Operating License No. DPR-42 are hereby amended to read as follows:

## 2.C.(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 120, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

### 2.C.(4) Fire Protection

Northern States Power Company shall implement and maintain in effect all provisions of the approved fire protection program as described and referenced in the Updated Safety Analysis Report for the Prairie Island Nuclear Generating Plant, Units 1 and 2, and as approved in Safety Evaluation Reports dated February 14, 1978, September 6, 1979, April 4, 1980, December 29, 1980, July 28, 1981, September 12, 1984, June 25, 1985, October 27, 1989, and October 6, 1995, subject to the following provision:

The licensee may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

3. This license amendment is effective as of the date of issuance to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Beth A. Wetzel, Project Manager

Project Directorate III-1

Beth a. Wetel

Division of Reactor Projects - III/IV Office of Nuclear Reactor Regulation

Attachments:
Pages 3 and 4 of License No. DPR-42\*
Changes to the Technical
Specifications

Date of Issuance: October 6, 1995

\*Pages 3 and 4 are attached, for convenience, for the composite license to reflect these changes.

# ATTACHMENT TO LICENSE AMENDMENT NO. 120

# FACILITY OPERATING LICENSE NO. DPR-42

# **DOCKET NO. 50-282**

# UNIT 1 LICENSE

REMOVE	<u>INSERT</u>
Pages 3	Pages 3 4
5	-

# TECHNICAL SPECIFICATIONS

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

<sup>\*</sup>Corrected page

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended as of May 11, 1976.

Amdt. No. 12 5-11-7

- (3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;
- (4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components;
- (5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility;
- (6) Pursuant to the Act and 10 CFR Parts 30 and 70, to transfer byproduct materials from other NSP job sites for the purposes of volume reduction and decontamination.

Amdt. No. 88 7-25-8

C. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

#### (1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 1650 megawatts thermal.

### (2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 120, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

Unit 1

### (3) Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Prairie Island Nuclear Generating Plant Physical Security Plan, " with revisions submitted through November 30. 1987; "Prairie Island Nuclear Generating Plant Guard Training and Qualification Plan, " with revisions submitted through February 26, 1986; and "Prairie Island Nuclear Generating Plant Safeguards Contingency Plan, with revisions submitted through August 20, 1980. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

Amdt. No. 85 1-5-89

### (4) Fire Protection

Northern States Power Company shall implement and maintain in effect all provisions of the approved fire protection program as described and referenced in the Updated Safety Analysis Report for the Prairie Island Nuclear Generating Plant, Units 1 and 2, and as approved in Safety Evaluation Reports dated February 14, 1978, September 6, 1979, April 4, 1980, December 29, 1980, July 28, 1981, September 12, 1984, June 25, 1985, October 27, 1989, and October 6, 1995, subject to the following provision:

Amdt. No. 120 10-6-99

The licensee may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

D. This license is effective as of the date of issuance and shall expire at midnight August 9, 2013.

Amdt. No. 79 9-23-8

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by Roger S. Boyd

A. Giambusso, Deputy Director for Reactor Projects Directorate of Licensing

Attachment: Change No. 3 to Appendices A and B

Date of Issuance: APR 5 1974

Unit 1

TS SECTION	TITLE	PAGE
3.10	Control Rod and Power Distribution Limits A. Shutdown Margin B. Power Distribution Limits C. Quadrant Power Tilt Ratio	TS.3.10-1 TS.3.10-1 TS.3.10-1 TS.3.10-4
	D. Rod Insertion Limits	TS.3.10-5
	E. Rod Misalignment Limitations	TS.3.10-6
	F. Inoperable Rod Position Indicator Channels	TS.3.10-6
	G. Control Rod Operability Limitations	TS.3.10-7
	H. Rod Drop Time	TS.3.10-7
	I. Monitor Inoperability Requirements	TS.3.10-8
	J. DNB Parameters	TS.3.10-8
	Core Surveillance Instrumentation	TS.3.11-1
	Snubbers	TS.3.12-1
3.13	Control Room Air Treatment System	TS.3.13-1
	A Control Room Special Ventilation System	TS.3.13-1
3.14	Deleted	
3.15	Event Monitoring Instrumentation	TS.3.15-1
	A. Process Monitors	TS.3.15-1
	B. Radiation Monitors	TS.3.15-1
	C. Reactor Vessel Level Instrumentation	TS.3.15-2
		10.3.13-2

TS SECTIO	N TITLE	PAGE
4.0 St	RVEILLANCE REQUIREMENTS	TS.4.0-1
4.1	Operational Safety Review	TS.4.1-1
4.2	Inservice Inspection and Testing of Pumps and	******
	valves kequirements	TS.4.2-1
	A. Inspection Requirements	TS.4.2-1
	B. Corrective Heasures	TS.4.2-2
	C. Records	TS.4.2-3
4.3	Primary Coolant System Pressure Isolation Valves	TS.4.3-1
4.4	Containment System Tests	TS.4.4-1
	A. Containment Leakage Tests	TS.4.4-1
	B. Emergency Charcoal Filter Systems	TS.4.4-3
	C. Containment Vacuum Breakers	TS.4.4-4
	D. Residual Heat Removal System	TS.4.4-4
	E. Containment Isolation Valves	TS.4.4-5
	F. Post Accident Containment Ventilation System G. Containment and Shield Building Air	TS.4.4-5
	Temperature	TS.4.4-5
	H. Containment Shell Temperature	TS.4.4-5
4.5	I. Electric Hydrogen Recombiners	TS.4.4-5
4.5	Engineered Safety Features	TS.4.5-1
	A. System Tests	TS.4.5-1
	1. Safety Injection System	TS.4.5-1
	2. Containment Spray System	TS.4.5-1
	3. Containment Fan Coolers	TS.4.5-2
	4. Component Cooling Water System	TS.4.5-2
	5. Cooling Water System B. Component Tests	TS.4.5-2
	1. Pumps	TS.4.5-3
	2. Containment Fan Motors	TS.4.5-3
	3. Valves	TS.4.5-3
4.6	Periodic Testing of Emergency Power System	TS.4.5-3
	A. Diesel Generators	TS.4.6-1
	B. Station Batteries	TS.4.6-1
	C. Pressurizer Heater Emergency Power Supply	TS.4.6-3
4.7	Main Steam Isolation Valves	TS.4.6-3
4.8	Steam and Power Conversion Systems	TS.4.7-1
	A. Auxiliary Feedwater System	TS.4.8-1 TS.4.8-1
	B. Steam Generator Power Operated Relief Valves	TS.4.8-2
	C. Steam Exclusion System	TS.4.8-2
4.9	Reactivity Anomalies	TS.4.9-1
4.10	Radiation Environmental Monitoring Program	TS.4.10-1
	A. Sample Collection and Analysis	TS.4.10-1
	B. Land Use Census	TS.4.10-2
	C. Interlaboratory Comparison Program	TS.4.10-2
4.11	Radioactive Source Leakage Test	TS.4.11-1
Prairie Island Prairie Island	Unit 1 - Amendment No. 69, 73, 91, 101, 103 Unit 2 - Amendment No. 63, 66, 84, 94, 96	

TS SECTION	TITLE	PAGE
4.12 Steam Generato	or Tube Surveillance	TS.4.12-1
A. Steam Generat Inspecti	or Sample Selection and on	TS.4.12-1
	or Tube Sample Selection	TS.4.12-1
C. Inspection Fr		TS.4.12-3
D. Acceptance Cr		TS.4.12-4
E. Reports		TS.4.12-5
4.13 Snubbers		TS.4.13-1
4.14 Control Room A	ir Treatment System Tests	TS.4.14-1
4.15 Spent Fuel Poo	1 Special Ventilation System	TS.4.15-1
4.16 Deleted		
4.17 Radioactive Ef	fluents Surveillance	TS.4.17-1
A. Liquid Efflue	nts	TS.4.17-1
B. Gaseous Efflu	ents	TS.4.17-2
C. Solid Radioac	tive Waste	TS.4.17-4
D. Dose from All	Uranium Fuel Cycle Sources	TS.4.17-4
4.18 Reactor Coolan		TS.4.18-1
A. Vent Path Ope		TS.4.18-1
B. System Flow T		TS.4.18-1
4.19 Auxiliary Buil	ding Crane Lifting Devices	TS.4.19-1

TS BASES	SECT	ION TITLE	PAGE
2.0	SETTI		
	2.1	Safety Limit, Reactor Core	B.2.1-1
	2.2	Safety Limit, Reactor Coolant System Pressure	B.2.2-1
	2.3	Limiting Safety System Settings, Protective Instrumentation	B.2.3-1
3.0		FOR LIMITING CONDITIONS FOR OPERATION	
		Applicability	B.3.0-1
	3.1	Reactor Coolant System	B.3.1-1
		A. Operational Components	B.3.1-1
		B. Pressure/Temperature Limits	B.3.1-4
		C. Reactor Coolant System Leakage	B.3.1-6
		D. Maximum Coolant Activity	B.3.1-7
		E. Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration	B.3.1-8
	3.2	F. Isothermal Temperature Coefficient (ITC) Chemical and Volume Control System	B.3.1-9
	3.3	Engineered Safety Features	B.3.2-1
	3.4		B.3.3-1
		Instrumentation System	B.3.4-1
	3.6	Containment System	B.3.5-1
	3.7	Auxiliary Electrical System	B.3.6-1 B.3.7-1
	3.8	Refueling and Fuel Handling	B.3.7-1 B.3.8-1
	3.9	Radioactive Effluents	B.3.9-1
		A. Liquid Effluents	B.3.9-1
		B. Gaseous Effluents	B.3.9-2
		C. Solid Radioactive Waste	B.3.9-4
		D. Dose From All Uranium Fuel Cycle Sources	B.3.9-5
		E. & F. Effluent Monitoring Instrumentation	B.3.9-5
•	3.10	Control Rod and Power Distribution Limits	B.3.10-1
		A. Shutdown Margin	B.3.10-1
		B. Power Distribution Control	B.3.10-1
		C. Quadrant Power Tilt Ratio	B.3.10-6
		D. Rod Insertion Limits	B.3.10-8
		E. Rod Misalignment Limitation	B.3.10-9
		F. Inoperable Rod Position Indicator Channels	B.3.10-9
		G. Control Rod Operability Limitations	B.3.10-9
		H. Rod Drop Time	B.3.10-10
		I. Monitor Inoperability Requirements	B.3.10-10
•		J. DNB Parameters	B.3.10-10
	.11	Core Surveillance Instrumentation	B.3.11-1
		Snubbers	B.3.12-1
		Control Room Air Treatment System	B.3.13-1
3	.14	Deleted	
3	.15	Event Monitoring Instrumentation	B.3.15-1

TS BASES	S SECTI	ON TITLE	PAGE
4.0	BASES	FOR SURVEILLANCE REQUIREMENTS	
	4.1	Operational Safety Review	B.4.1-1
	4.2	Inservice Inspection and Testing of Pumps	
		and Valves Requirements	B.4.2-1
	4.3	Primary Coolant System Pressure Isolation	10 / 2 1
		Valves	B.4.3-1
	4.4	Containment System Tests	B.4.4-1
	4.5	Engineered Safety Features	B.4.5-1
	4.6	Periodic Testing of Emergency Power Systems	
	4.7	Main Steam Isolation Valves	B.4.6-1
	4.8	Steam and Power Conversion Systems	B.4.7-1
	4.9	Reactivity Anomalies	B.4.8-1
	4.10	Radiation Environmental Monitoring Program	B.4.9-1
		A. Sample Collection and Analysis	B.4.10-1
		B. Land Use Census	B.4.10-1
			B.4.10-1
	4.11	C. Interlaboratory Comparison Program	B.4.10-1
		Radioactive Source Leakage Test	B.4.11-1
	٠.12 ٨ 13	Steam Generator Tube Surveillance Snubbers	B.4.12-1
	4.13	Shubbers	B.4.13-1
	4.14	Control Room Air Treatment System Tests	B.4.14-1
	4.15	Spent Fuel Pool Special Ventilation System	B.4.15-1
	4.16	Deleted	
	4.17	Radioactive Effluents Surveillance	B / 17 1
	4.18	Reactor Coolant Vent System Paths	B.4.17-1
	4.19	Auxiliary Building Crane Lifting Devices	B.4.18-1
		A Print Diame Fricing DeAlces	B.4.19-1

# TECHNICAL SPECIFICATIONS

# LIST OF TABLES

<b>5</b> 6 <b>5.5</b> 5	ATST OF TABLES
IS TABLE	TITLE
1-1	Operational Modes
3.5-1	Engineered Safety Features Initiation Instrument Limiting Set Points
3.5-2A 3.5-2B	Reactor Trip System Instrumentation Engineered Safety Feature Actuation System Instrumentation
3.9-1 3.9-2	Radioactive Liquid Effluent Monitoring Instrumentation Radioactive Gaseous Effluent Monitoring instrumentation
3.15-1	Event Monitewine Town
3.15-2	Event Monitoring Instrumentation - Process & Containment Event Monitoring Instrumentation - Radiation
4.1-1A	Reactor Trip System Instrumentation Surveillance Requirements
4.1-1B	Surveillance Requirements
4.1-1C 4.1-2A	Miscellaneous Instrumentation Surveillance Paraday
4.1-2B	Minimum Frequencies for Equipment Tests Minimum Frequencies for Sampling Tests
4.2-1	Special Inservice Inspection Requirements
4.10-1	Radiation Environmental Monitoring Program (REMP) Sample Collection and Analysis
4.10-2	RFMP - Maximum Values for the lawer target and
4.10-3	RFMP - Reporting Levels for Radioactivity Concentrations in Environmental Samples
4.12-1	Steam Generator Tube Inspection
4.13-1	Snubber Visual Inspection Interval
4.17-1	Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements
4.17-2	Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements
4.17-3	Radioactive Liquid Waste Camping and Angles
4.17-4	Radioactive Gaseous Waste Sampling and Analysis Program
5.5-1	Anticipated Annual Release of Radioactive Material in Liquid Effluents From Prairie Island Nuclear Generating Plant (Per Unit)
5.5-2	Anticipated Annual Release of Radioactive Nuclides in Gaseous Effluent From Prairie Island Nuclear Generating Plant (Per Unit)
6.1-1	Minimum Shift Crew Composition
Prairie Isla Prairie Isla	Amendment No. 98,187, 111, 120 and Unit 2 Amendment No. 91,188, 104, 113

## DOSE EQUIVALENT 1-131

DOSE EQUIVALENT I-131 is that concentration of I-131 (uCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites".

# E-AVERAGE DISINTEGRATION ENERGY

E shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

# GASEOUS RADWASTE TREATMENT SYSTEM

The GASEOUS RADWASTE TREATMENT SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

- 3. At least two licensed operators shall he present in the control room during a reactor startup, a scheduled reactor shutdown, and during recovery from a reactor trip. These operators are in addition to those required for the other reactor.
- 4. An individual qualified in radiation protection procedures shall be on site when fuel is in a reactor.
- 5. All refueling operations shall be directly supervised by a licensed Senior Reactor Operator or a Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.
- 6. The General Superintendent Plant Operations shall be formerly licensed or hold a current license on a similar type plant.
- 7. At least one member of plant management holding a current Senior Reactor Operator license shall be assigned to the plant operations group on a long term basis (approximately two years). This individual shall not be assigned to a rotating shift.
- D. Each member of the plant staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the General Superintendent Radiation Protection who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and (2) the Shift Manager who shall have a bachelors degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents, and (3) the General Superintendent Plant Operations who shall meet the requirements of ANSI N18.1-1971, except that NRC license requirements are as specified in Specification 6.1.C.7. The training program shall be under the direction of a designated member of Northern States Power management.

- E. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, and key maintenance personnel. Procedures shall include the following provisions:
  - 1. Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a nominal 40-hour week while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:
    - a. An individual should not be permitted to work more than 16 hours straight excluding shift turnover time.
    - b. Overtime should be limited for all nuclear plant staff personnel so that total work time does not exceed 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 84 hours in any seven day period, all excluding shift turnover time. Individuals should not be required to work more than 15 consecutive days without two consecutive days off.
    - c. A break of at least eight hours including shift turnover time should be allowed between work periods.
    - d. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.
    - e. Shift Emergency Coordinator (SEC) on-site rest time periods shall not be considered as hours worked when determining the total work time for which the above limitations apply.

- f. Investigations of all Reportable Events and events requiring Special Reports to the Commission.
- g. Drills on emergency procedures (including plant evacuation) and adequacy of communication with offsite support groups.
- h. All procedures required by these Technical Specifications, including implementing procedures of the Emergency Plan, and the Security Plan (except as exempted in Section 6.5.F), shall be reviewed initially and periodically with a frequency commensurate with their safety significance but at an interval of not more than two years.

  Maintenance work requests and their associated procedures shall be reviewed per the requirements of Section 6.2.C.
- i. Special reviews and investigations, as requested by the Safety Audit Committee.
- j. Review of investigative reports of unplanned releases of radioactive material to the environs.
- k. All changes to the Process Control Program (PCP) and the Offsite Dose Calculation Manual (ODCM).
- 1. The review of safety evaluations, when safety evaluations are required by 10 CFR Part 50, Section 50.59, for procedures or procedure changes to verify that such actions do not constitute an unreviewed safety question.
- m. Fire Protection Program and implementing procedures and the submittal of recommended changes to the Safety Audit Commmittee.

### 5. Authority

The OC shall be advisory to the Plant Manager. In the event of a disagreement between the recommendations of the OC and the Plant Manager, the course determined by the Plant Manager to be the more conservative will be followed. A written summary of the disagreement will be sent to the Vice President Nuclear Generation and the Chairman of the SAC for review.

#### 6. Records

Minutes shall be recorded for all meetings of the OC and shall identify all documentary material reviewed. The minutes shall be distributed to each member of the OC, the Chairman and each member of the Safety Audit Committee, the Vice President Nuclear Generation and others designated by the OC Chairman.

#### 7. Procedures

A written charter for the OC shall be prepared that contains:

a. Responsibility and authority of the group



# UNITED STATES NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

## NORTHERN STATES POWER COMPANY

### **DOCKET NO. 50-306**

### PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 2

### AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 113 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 July 11, 1994, as supplemented April 18, 1995, 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 following sections under paragraph 2.C of Facility Operating License No. DPR-60 are hereby amended to read as follows:

# 2.C.(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 113, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

# 2.C.(4) <u>Fire Protection</u>

Northern States Power Company shall implement and maintain in effect all provisions of the approved fire protection program as described and referenced in the Updated Safety Analysis Report for the Prairie Island Nuclear Generating Plant, Units 1 and 2, and as approved in Safety Evaluation Reports dated February 14, 1978, September 6, 1979, April 4, 1980, December 29, 1980, July 28, 1981, September 12, 1984, June 25, 1985, October 27, 1989, and October 6, 1995, subject to the following provision:

The licensee may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

 This license amendment is effective as of the date of issuance to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Beth A. Wetzel, Project Manager

Project Directorate III-1

Beth a Wetel

Division of Reactor Projects - III/IV Office of Nuclear Reactor Regulation

Attachments:
Pages 3 and 4 of License No. DPR-60\*
Changes to the Technical
Specifications

Date of Issuance: October 6, 1995

<sup>\*</sup>Pages 3 and 4 are attached, for convenience, for the composite license to reflect these changes.

### ATTACHMENT TO LICENSE AMENDMENT NO. 113

# FACILITY OPERATING LICENSE NO. DPR-60

# **DOCKET NO. 50-306**

# UNIT 2 LICENSE

REMOVE	INSERT
Pages 3	Pages 3 4

### TECHNICAL SPECIFICATIONS

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
TS-iv	TS-iv
TS-v	TS-v*
TS-vi	TS-vi
TS-x	TS-x
TS-xi	TS-xi
TS-xii TS.1-3	TS-xii
TS.3.14-1	TS.1-3
TS.3.14-2	<del>-</del> -
TS.3.14-3	<b></b>
TS.3.14-4	<b></b>
TABLE TS.3.14-1, page 1	
TABLE TS.3.14-1, page 2	<del></del>
TABLE TS.3.14-1, page 3	
TS.4.16-1	
TS.4.16-2	
TS.4.16-3	
TS.4.16-4	
TS.4.16-5	
TS.4.16-6	
TS.6.1-2	TS.6.1-2
TS.6.1-3	TS.6.1-3
TS.6.2-6	TS.6.2-6
B.3.14-1	
B.3.14-2	
B.4.16-1	
B.4.16-2	~

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components;

Amdt. No. 6 5-11-76

- (5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility;
- (6) Pursuant to the Act and 10 CFR Parts 30 and 70, to transfer byproduct materials from other NSP job sites for the purposes of volume reduction and decontamination.

Amdt. No. 81 7-25-89

C. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

### (1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 1650 megawatts thermal.

### (2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 113, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

### (3) Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Prairie Island Nuclear Generating Plant Physical Security Plan," with revisions submitted through November 30,

Amdt. No. 78 1-5-89

## (3) Physical Protection -- continued

1987; "Prairie Island Nuclear Generating Plant Guard Training and Qualification Plan," with revisions submitted through February 26, 1986; and "Prairie Island Nuclear Generating Plant Safeguards Contingency Plan," with revisions submitted through August 20, 1980. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

Amdt. No. 78 1-5-89

### (4) Fire Protection

Northern States Power Company shall implement and maintain in effect all provisions of the approved fire protection program as described and referenced in the Updated Safety Analysis Report for the Prairie Island Nuclear Generating Plant, Units 1 and 2, and as approved in Safety Evaluation Reports dated February 14, 1978, September 6, 1979, April 4, 1980, December 29, 1980, July 28, 1981, September 12, 1984, June 25, 1985, October 27, 1989, and October 6, 1995, subject to the following provision:

Amdt. No. 11 10-6-9

The licensee may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

 D. This license is effective as of the date of issuance and shall expire at midnight October 29, 2014.

Amdt. No. 72 9-23-8

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by A. Giambusso

A. Giambusso, Deputy Director for Reactor Projects Directorate of Licensing

Date of Issuance: OCT 29 1974

TS SECTION	TITLE	PAGE
3.10	Control Rod and Power Distribution Limits A. Shutdown Margin B. Power Distribution Limits C. Quadrant Power Tilt Ratio B. Rod Insertion Limits E. Rod Misalignment Limitations F. Inoperable Rod Position Indicator Channels C. Control Rod Operability Limitations	TS.3.10-1 TS.3.10-1 TS.3.10-1 TS.3.10-4 TS.3.10-5 TS.3.10-6
• • •	<ul><li>H. Rod Drop Time</li><li>I. Monitor Inoperability Requirements</li><li>J. DNB Parameters</li></ul>	TS.3.10-7 TS.3.10-7 TS.3.10-8 TS.3.10-8
3.11	Core Surveillance Instrumentation Snubbers	TS.3.11-1
3.13		TS.3.12-1 TS.3.13-1 TS.3.13-1
3.14	Deleted	
3.15	Event Monitoring Instrumentation A. Process Monitors B. Radiation Monitors C. Reactor Vessel Level Instrumentation	TS.3.15-1 TS.3.15-1 TS.3.15-1 TS.3.15-2

IS SECT	ION	PAGE	
4.0	SURVEILLANCE REQUIREMENTS		
4.1	Operational Safety Review	TS.4.0-1	
4.2	Inservice Inspection and Testing of Pumps and	TS.4.1-1	
	vedatiements	TS.4.2-1	
	A. Inspection Requirements	TS.4.2-1	
	B. Corrective Measures	TS.4.2-2	
4.3	C. Records	TS.4.2-3	
4.4	The state of the s	TS.4.3-1	
<b></b>		TS.4.4-1	
	A. Containment Leakage Tests	TS.4.4-1	
	B. Emergency Charcoal Filter Systems	TS.4.4-3	
	C. Containment Vacuum Breakers	TS.4.4-4	
	D. Residual Heat Removal System	TS.4.4-4	
	E. Containment Isolation Valves	TS.4.4-5	
	F. Post Accident Containment Ventilation System G. Containment and Shield Building Air	TS.4.4-5	
	Temperature	TS.4.4-5	
	H. Containment Shell Temperature	TS.4.4-5	
4.5	I. Electric Hydrogen Recombiners	TS.4.4-5	
•••	Engineered Safety Features A. System Tests	TS.4.5-1	
	1 Cafaty Interests a	TS.4.5-1	
	1. Safety Injection System 2. Containment Server Survey	TS.4.5-1	
	<ol> <li>Containment Spray System</li> <li>Containment Fan Coolers</li> </ol>	TS.4.5-1	
	4. Component Cooling Uses a	TS.4.5-2	
	4. Component Cooling Water System 5. Cooling Water System	TS.4.5-2	
	B. Component Tests	TS.4.5-2	
	1. Pumps	TS.4.5-3	ı
	2. Containment Fan Motors	TS.4.5-3	Į
	3. Valves	TS.4.5-3	•
4.6	Periodic Testing of Emergency Power System	TS.4.5-3	
	A. Diesel Generators	TS.4.6-1	
	B. Station Batteries	TS.4.6-1	
	C. Pressurizer Heater Emergency Power Supply	TS.4.6-3	
4.7	AND THE PERSON AND AND AND AND AND AND AND AND AND AN	TS.4.6-3	
4.8	Steam and Power Conversion Systems	TS.4.7-1	
	A. Auxiliary Feedwater System	TS.4.8-1	
	B. Steam Generator Power Operated Relies Walnut	TS.4.8-1	
	o. Sceni Exclusion System	TS.4.8-2	
4.9	Reactivity Anomalies	TS.4.8-2	
4.10	Radiation Environmental Monitoring Program	TS.4.9-1	
	A. Sample Collection and Analysis	TS.4.10-1	
	B. Land Use Census	TS.4.10-1	
	C. Interlaboratory Comparison Program	TS.4.10-2	
4.11	Radioactive Source Leakage Test	TS.4.10-2 TS.4.11-1	
Prairie Island	Unit 1 - Amendment No. 44 72 dr var	19.4.11.1	
Prairie Island	Unit 2 - Amendment No. 63, 66, 84, 94, 96		

TS SECTION	TITLE	PAGE
4.12 Steam Generator	Tube Surveillance	TS.4.12-1
A. Steam Generator Inspection	Sample Selection and	TS.4.12-1
B. Steam Generator and Inspect	Tube Sample Selection	TS.4.12-1
C. Inspection Freque	uencies	<b>m</b> o / •• •
D. Acceptance Crite	eria	TS.4.12-3
E. Reports		TS.4.12-4
4.13 Snubbers		TS.4.12-5
4.14 Control Room Air	Treatment System Tacks	TS.4.13-1
4.15 Spent Fuel Pool S	pecial Ventilation System	TS.4.14-1
	Poolar ventifactor system	TS.4.15-1
4.16 Deleted		
4.17 Radioactive Efflu	ents Surveillance	<b>5</b> 0 / 37 4
A. Liquid Effluents		TS.4.17-1
B. Gaseous Effluent		TS.4.17-1
C. Solid Radioactiv		TS.4.17-2
D. Dose from All Ur.	anium Fuel Cycle Sources	TS.4.17-4
4.18 Reactor Coolant V	ent System Patha	TS.4.17-4
A. Vent Path Operab	flite	TS.4.18-1
B. System Flow Test:	tna	TS.4.18-1
4.19 Auxiliary Building	r Crana Ilfal D	TS.4.18-1
	s crane Litting Devices	TS.4.19-1

TS BASES SE	CTION TITLE	PAGE
SE	SES FOR SAFETY LIMITS AND LIMITING SAFETY SYSTEM	
2.	Safety Limit, Reactor Core	B.2.1-1
2.	Safety Limit, Reactor Coolant System Pressure	B.2.2-1
Z.,	Limiting Safety System Settings, Protective Instrumentation	B.2.3-1
3.0 BA	ES FOR LIMITING CONDITIONS FOR OPERATION	
3.1	Applicability	B.3.0-1
3	Reactor Coolant System	B.3.1-1
	A. Operational Components	B.3.1-1
	B. Pressure/Temperature Limits	B.3.1-4
	C. Reactor Coolant System Leakage	B.3.1-6
	D. Maximum Coolant Activity	B.3.1-7
	E. Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration	B.3.1-8
3.2	F. Isothermal Temperature Coefficient (ITC) Chemical and Volume Control System	B.3.1-9
3.3	The state of the s	B.3.2-1
	Steam and Power Conversion Systems	B.3.3-1
3.5	Instrumentation System	B.3.4-1
3.6	Containment System	B.3.5-1
3.7	Auxiliary Electrical System	B.3.6-1
3.8	Refueling and Fuel Handling	B.3.7-1
3.9		B.3.8-1
	A. Liquid Effluents	B.3.9-1
	B. Gaseous Effluents	B.3.9-1
	C. Solid Radioactive Waste	B.3.9-2
	D. Dose From All Uranium Fuel Cycle Sources	B.3.9-4
	E. & F. Effluent Monitoring Instrumentation	B.3.9-5
3.10	Control Rod and Power Distribution Limits	B.3.9-5
	A. Shutdown Margin	B.3.10-1
	B. Power Distribution Control	B.3.10-1 B.3.10-1
	C. Quadrant Power Tilt Ratio	B.3.10-6
	D. Rod Insertion Limits	B.3.10-8
	E. Rod Misalignment Limitation	B.3.10-9
	F. Inoperable Rod Position Indicator Channels	B.3.10-9
	G. Control Rod Operability Limitations	B.3.10-9
	H. Rod Drop Time	B.3.10-10
	I. Monitor Inoperability Requirements	B.3.10-10
	J. DNB Parameters	B.3.10-10
3.11		B.3.11-1
	Snubbers	B.3.12-1
3.13	Control Room Air Treatment System	B.3.13-1
3.14	Deleted	
3.15	Event Monitoring Instrumentation	B.3.15-1

TS BASES	SECTI	<u>ON</u> TITLE	PAGE
4.0	BASES	FOR SURVEILLANCE REQUIREMENTS	
	4.1		B.4.1-1
	4.2		B.4.2-1
	4.3	Primary Coolant System Pressure Isolation Valves	B.4.3-1
	4.4	Containment System Tests	B.4.4-1
	4.5	Engineered Safety Features	B.4.5-1
	4.6		B.4.6-1
	4.7	Main Steam Isolation Valves	B.4.7-1
	4.8	Steam and Power Conversion Systems	B.4.8-1
	4.9	Reactivity Anomalies	B.4.9-1
	4.10	Radiation Environmental Monitoring Program	B.4.10-1
		A. Sample Collection and Analysis	B.4.10-1
		B. Land Use Census	B.4.10-1
		C. Interlaboratory Comparison Program	B.4.10-1
	4.11	Radioactive Source Leakage Test	B.4.11-1
	4.12	Steam Generator Tube Surveillance	B.4.12-1
		Snubbers	B.4.13-1
	4.14	Control Room Air Treatment System Tests	B.4.14-1
	4.15	Spent Fuel Pool Special Ventilation System	B.4.15-1
	4.16	Deleted	
		Radioactive Effluents Surveillance	B.4.17-1
	4.18	Reactor Coolant Vent System Paths	B.4.18-1
		Auxiliary Building Crane Lifting Devices	B.4.19-1

Amendment No. 98,187, 111, 120 Amendment No. 91,188, 104, 113

# TECHNICAL SPECIFICATIONS

# LIST OF TABLES

IS TABLE	TITLE
1-1	Operational Modes
3.5-1	Engineered Safety Features Initiation Instrument Limiting Set Points
3.5-2A 3.5-2B	Reactor Trip System Instrumentation Engineered Safety Feature Actuation System Instrumentation
3.9-1 3.9-2	Radioactive Liquid Effluent Monitoring Instrumentation Radioactive Gaseous Effluent Monitoring instrumentation
3.15-1	Event Monitoring Instrumentation - Process & Containment
3.15-2	Event Monitoring Instrumentation - Radiation
4.1-1A 4.1-1B	Reactor Trip System Instrumentation Surveillance Requirements Engineered Safety Feature Actuation System Instrumentation Surveillance Requirements
4.1-1C	Miscellaneous Instrumentation Surveillance Requirements
4.1-2A	Minimum Frequencies for Equipment Tests
4.1-2B	Minimum Frequencies for Sampling Tests
4.2-1	Special Inservice Inspection Requirements
4.10-1	Radiation Environmental Monitoring Program (REMP) Sample Collection and Analysis
4.10-2	RFMP - Maximum Values for the lover limits of December
4.10-3	RFMP - Reporting Levels for Radioactivity Concentrations in Environmental Samples
4.12-1	Steam Generator Tube Inspection
4.13-1	Snubber Visual Inspection Interval
4.17-1	Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements
4.17-2	Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements
4.17-3	Radioactive Liquid Waste Sampling and Analysis Program
4.17-4	Radioactive Gaseous Waste Sampling and Analysis Program
5.5-1	Anticipated Annual Release of Radioactive Material in Liquid Effluents From Prairie Island Nuclear Generating Plant (Per Unit)
5.5-2	Anticipated Annual Release of Radioactive Nuclides in Gaseous Effluent From Prairie Island Nuclear Generating Plant (Per Unit)
6.1-1	Minimum Shift Crew Composition

Prairie Island Unit 1 Prairie Island Unit 2

### DOSE EQUIVALENT 1-131

DOSE EQUIVALENT I-131 is that concentration of I-131 (uCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites".

# E-AVERAGE DISINTEGRATION ENERGY

E shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

### GASEOUS RADWASTE TREATMENT SYSTEM

The GASEOUS RADWASTE TREATMENT SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

- 3. At least two licensed operators shall he present in the control room during a reactor startup, a scheduled reactor shutdown, and during recovery from a reactor trip. These operators are in addition to those required for the other reactor.
- 4. An individual qualified in radiation protection procedures shall be on site when fuel is in a reactor.
- 5. All refueling operations shall be directly supervised by a licensed Senior Reactor Operator or a Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.
- 6. The General Superintendent Plant Operations shall be formerly licensed or hold a current license on a similar type plant.
- 7. At least one member of plant management holding a current Senior Reactor Operator license shall be assigned to the plant operations group on a long term basis (approximately two years). This individual shall not be assigned to a rotating shift.
- D. Each member of the plant staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the General Superintendent Radiation Protection who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and (2) the Shift Manager who shall have a bachelors degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents, and (3) the General Superintendent Plant Operations who shall meet the requirements of ANSI N18.1-1971, except that NRC license requirements are as specified in Specification 6.1.C.7. The training program shall be under the direction of a designated member of Northern States Power management.

- E. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, and key maintenance personnel. Procedures shall include the following provisions:
  - 1. Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a nominal 40-hour week while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:
    - a. An individual should not be permitted to work more than 16 hours straight excluding shift turnover time.
    - b. Overtime should be limited for all nuclear plant staff personnel so that total work time does not exceed 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 84 hours in any seven day period, all excluding shift turnover time. Individuals should not be required to work more than 15 consecutive days without two consecutive days off.
    - c. A break of at least eight hours including shift turnover time should be allowed between work periods.
    - d. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.
    - e. Shift Emergency Coordinator (SEC) on-site rest time periods shall not be considered as hours worked when determining the total work time for which the above limitations apply.

- f. Investigations of all Reportable Events and events requiring Special Reports to the Commission.
- g. Drills on emergency procedures (including plant evacuation) and adequacy of communication with offsite support groups.
- h. All procedures required by these Technical Specifications, including implementing procedures of the Emergency Plan, and the Security Plan (except as exempted in Section 6.5.F), shall be reviewed initially and periodically with a frequency commensurate with their safety significance but at an interval of not more than two years. Maintenance work requests and their associated procedures shall be reviewed per the requirements of Section 6.2.C.
- i. Special reviews and investigations, as requested by the Safety Audit Committee.
- j. Review of investigative reports of unplanned releases of radioactive material to the environs.
- k. All changes to the Process Control Program (PCP) and the Offsite Dose Calculation Manual (ODCM).
- 1. The review of safety evaluations, when safety evaluations are required by 10 CFR Part 50, Section 50.59, for procedures or procedure changes to verify that such actions do not constitute an unreviewed safety question.
- m. Fire Protection Program and implementing procedures and the submittal of recommended changes to the Safety Audit Commmittee.

#### 5. Authority

The OC shall be advisory to the Plant Manager. In the event of a disagreement between the recommendations of the OC and the Plant Manager, the course determined by the Plant Manager to be the more conservative will be followed. A written summary of the disagreement will be sent to the Vice President Nuclear Generation and the Chairman of the SAC for review.

### 6. Records

Minutes shall be recorded for all meetings of the OC and shall identify all documentary material reviewed. The minutes shall be distributed to each member of the OC, the Chairman and each member of the Safety Audit Committee, the Vice President Nuclear Generation and others designated by the OC Chairman.

#### 7. Procedures

A written charter for the OC shall be prepared that contains:

a. Responsibility and authority of the group



# UNITED STATES NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 120 AND 113 TO

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

NORTHERN STATES POWER COMPANY

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

**DOCKET NOS. 50-282 AND 50-306** 

### 1.0 INTRODUCTION

Section 50.48, "Fire protection," of Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50) requires that each operating nuclear power plant have a fire protection plan that satisfies Criterion 3 (GDC 3), "Fire protection," of Appendix A to 10 CFR Part 50. The fire protection plan must describe the overall fire protection program for the facility, outline the plans for fire protection, fire detection, and fire suppression capability, and limitations of fire damage. The program must also describe specific features necessary to implement the program, such as administrative controls and personnel requirements for fire prevention and manual fire suppression activities, automatic and manually operated fire detection and suppression systems, and the means to limit fire damage to structures. systems, or components important to safety so that the capability to safely shut down the plant is ensured. The U.S. Nuclear Regulatory Commission (NRC) staff approved the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, (Prairie Island) fire protection program in Safety Evaluation Reports dated February 14, 1978, September 6, 1979, April 4, 1980, December 29, 1980, July 28, 1981, September 12, 1984, June 25, 1985, and October 27, 1989.

By letter dated July 11, 1994, as supplemented by letter dated April 18, 1995, the Northern States Power Company (the licensee) submitted a request for changes to the Prairie Island fire protection program in accordance with the guidance provided in Generic Letter (GL) 86-10, "Implementation of Fire Protection Requirements," and GL 88-12, "Removal of Fire Protection Requirements from Technical Specifications." Specifically, the licensee proposed to incorporate the NRC-approved fire protection program and major commitments, including the fire hazard analysis, into the Updated Safety Analysis Report (USAR), and to revise the Operating Licenses to include the NRC's standard fire protection license condition. In addition, the licensee proposed to relocate the requirements of Technical Specifications (TS) Section 3.14 (Fire Protection and Detection Systems - Limiting Conditions for Operation), TS Section 4.16 (Fire Detection and Protection Systems -Surveillances), TS Section 6.1 (Administrative Controls, Organization), and TS Section 6.2 (Administrative Controls, Review and Audit) from the TS to the revised fire protection program.

GL 86-10 and GL 88-12 referred to removing fire protection requirements from TS. License amendments that relocate the fire protection requirements to the FSAR in accordance with GL 86-10 and GL 88-12 do not revise the requirements for fire protection operability, testing, or inspections. Such amendments simply replace the fire protection TS sections with the standard fire protection license condition. The license condition implements and maintains the NRC-approved fire protection program, including the fire protection requirements previously specified in the TS, in accordance with 10 CFR 50.48. Therefore, such amendments, including the one proposed by the licensee, are administrative in nature and have no effect on the public health and safety.

The letter of April 18, 1995, provided clarifying information within the scope of the original submittal and did not change the staff's initial proposed no significant hazards consideration determination.

### 2.0 BACKGROUND

Section 182a of the Atomic Energy Act (the "Act") requires applicants for nuclear power plant operating licenses to state TS to be included as part of the license. The Commission's regulatory requirements related to the content of TS are set forth in 10 CFR 50.36. That regulation requires that the TS include items in five specific categories, including (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. However, the regulation does not specify the particular requirements to be included in a plant's TS.

The Commission has provided guidance for the contents of TS in its "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" ("Final Policy Statement"), 58 FR 39132 (July 22, 1993), in which the Commission indicated that compliance with the Final Policy Statement satisfies Section 182a of the Act. In particular, the Commission indicated that certain items could be relocated from the TS to licensee-controlled documents, consistent with the standard enunciated in Portland General Electric Co. (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). In that case, the Atomic Safety and Licensing Appeal Board indicated that "technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety." The criteria set forth in the policy statement have been incorporated into 10 CFR 50.36 (60 FR 36953).

Following the fire at the Browns Ferry Nuclear Power Plant on March 22, 1975, the Commission undertook a number of actions to ensure that improvements were implemented in the fire protection programs for all power reactor facilities. Because of the extensive modification of fire protection programs and the number of open issues resulting from staff evaluations, a number of revisions and alterations occurred in these programs over the years. Consequently, licensees were requested by GL 86-10 to incorporate the final NRC-approved fire protection program in their Final Safety Analysis Reports (FSARs). In

this manner, the fire protection program, including the systems, certain administrative and technical controls, the organization, and other plant features associated with fire protection, would have a status consistent with that of other plant features described in the FSAR. In addition, the Commission concluded that a standard license condition, requiring compliance with the provisions of the fire protection program as described in the FSAR, should be used to ensure uniform enforcement of the fire protection requirements. Finally, the Commission stated that with the required actions, licensees may request an amendment to delete the fire protection TS that would now be unnecessary. Subsequently, the NRC issued GL 88-12 to give guidance for the preparation of the license amendment request to implement GL 86-10.

### 3.0 PROPOSED CHANGES

The specific TS changes proposed by the licensee are as follows:

1. Revise License Condition 2.C.(4) for both units as follows:

Northern States Power Company shall implement and maintain in effect all provisions of the approved fire protection program as described and referenced in the Updated Safety Analysis Report for the Prairie Island Nuclear Generating Plant, Unit 1 and 2, and as approved in Safety Evaluation Reports dated February 14, 1978, September 6, 1979, April 4, 1980, December 29, 1980, July 28, 1981, September 12, 1984, June 25, 1985, October 27, 1989, April 21, 1995, and October XX, 1995 subject to the following provision:

The licensee may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

- Delete TS 3.14.A (Fire Detection Instrumentation), TS 3.14.B (Fire Suppression Water System), TS 3.14.C (Spray and Sprinkler Systems), TS 3.14.D.(Carbon Dioxide System), TS 3.14.E (Fire Hose Stations), TS 3.14.F (Yard Fire Hydrant Hose Houses), TS 3.14.G (Penetration Fire Barriers) and their associated bases and Table, and incorporate them, by reference, into the Prairie Island USAR.
- 3. Delete TS 4.16.A (Fire Detection and Protection Systems), 4.16.B (Fire Detection Instrumentation), 4.16.C (Spray and Sprinkler Systems), 4.16.D (Carbon Dioxide System), 4.16.E (Fire Hose Stations), 4.16.F (Fire Hydrant Hose House), 4.16.G 9 (Penetration Fire Barriers), and their associated bases and Table, and incorporate them, by reference, into the Prairie Island USAR.
- 4. Delete definition of "Fire Suppression Water System" from the TS.
- 5. Delete TS 6.1.C.6, Fire Brigade staffing, and its associated footnote and TS 6.1.E and incorporate them, by reference, into the Prairie Island USAR.

- 6. Add new TS 6.2.B.4.m under Operations Committee responsibilities.
- 7. Revise the Table of Contents and List of Tables to reflect the TS changes.

### 4.0 EVALUATION

The NRC staff reviewed the license amendment requests for Prairie Island against the guidance provided in GLs 86-10 and 88-12. GL 86-10 requested that the licensee incorporate the NRC-approved fire protection program in its USAR for the facility and specified a standard fire protection license condition. GL 88-12 addressed the elements a licensee should include in a license amendment request to remove the fire protection requirements from the plant TS. These elements are (1) the NRC-approved fire protection program must be incorporated into the USAR; (2) the Limiting Conditions for Operation (LCOs) and Surveillance Requirements associated with fire detection systems, fire suppression systems, fire barriers, and the administrative controls that address fire brigade staffing would be relocated from the TS (the existing administrative controls associated with fire protection audits and specifications related to the capability for safe shutdown following a fire would be retained); (3) all operational conditions, remedial actions, and test requirements presently included in the TS for these systems, as well as the fire brigade staffing requirements, shall be incorporated into the fire protection program; (4) the standard fire protection license condition specified in GL 86-10 must be included in the facility operating license; (5) the Unit Review Group (Onsite Review Group) shall be given responsibility for the review of the fire protection program and implementing procedures and for the submittal of recommended changes to the Company Nuclear Review and Audit group (Offsite or Corporate Review Group); and (6) fire protection program implementation shall be added to the list of elements for which written procedures shall be established, implemented, and maintained. The licensee incorporated the NRC-approved fire protection program by reference into the Prairie Island USAR in December 1981. The licensee has, therefore, satisfied Element 1 of GL 88-12.

The licensee stated in its submittal of July 11, 1994, that it will incorporate the current TS LCOs and surveillance requirements for the fire detection systems, fire suppression systems, and the TS requirements related to fire brigade staffing into the Prairie Island Fire Protection Program. Therefore, the licensee will have satisfied Elements 2 and 3 of GL 88-12, with the exception of technical specifications for alternate safe shutdown equipment. The staff will pursue implementation of alternative safe shutdown equipment TS, consistent with GL 81-12, independent of this license amendment.

The licensee proposed incorporating the standard fire protection license condition specified in GL 86-10 for Prairie Island. The licensee has, therefore, satisfied Element 4 of GL 88-12.

To satisfy Element 5 of GL 88-12, the licensee addressed changes to the administrative controls sections of the TS. The licensee will require the Operations Committee to review the fire protection program and implementing

procedures as well as recommended changes as an additional responsibility. The licensee has, therefore, satisfied Element 5 of GL 88-12.

Element 6 of GL 88-12 specified that the licensee add fire protection program implementation to the administrative controls Section of the TS. This change is made to the list of elements for which written procedures shall be established, implemented, and maintained. Since TS 6.5 currently addresses the fire protection program, and this TS will remain in place following this amendment, no changes are required and the licensee has, therefore, satisfied Element 6 of GL 88-12.

The licensee's proposed TS amendments for Prairie Island are in accordance with NRC staff guidance provided in GLs 86-10 and 88-12.

In summary, the licensee has proposed to incorporate the existing TS fire protection requirements as stated above into the fire protection program which is, by reference, incorporated in the USAR. This conforms to staff guidance in GL 86-10, "Implementation of Fire Protection Requirements," and GL 88-12, "Removal of Fire Protection Requirements from Technical Specifications," for removing unnecessary fire protection TS in four major areas: fire detection systems, fire suppression systems, fire barriers and fire brigade staffing requirements. In addition, incorporating these requirements in the USAR is consistent with NUREG-1434 and 10 CFR 50.36, as amended, because these TS do not impact reactor operations, do not identify a parameter which is an initial condition assumption for a design-basis accident or transient, do not identify a significant abnormal degradation of the reactor coolant pressure boundary and do not provide any mitigation of a design-basis event.

The fire protection plan required by 10 CFR 50.48, as implemented and maintained by the fire protection license condition, provides reasonable assurance that fires will not give rise to an immediate threat to public health and safety. Although there are aspects of the fire detection and mitigation functions that have been determined to be risk significant, such that Criterion 4 of 10 CFR 50.36 would otherwise seem to apply, the minimum requirements for those functions were established in GDC 3 and 10 CFR 50.48, and further controls are not necessary since the licensee must comply with these minimum requirements regardless of whether they are restated in the TS or not.

The licensee's fire protection program is required by 10 CFR 50.48, and any changes to that program are governed by 10 CFR 50.48 and license condition 2.C.(4), set forth above. Therefore, the requirements relocated to the USAR may be controlled in accordance with 10 CFR 50.59.

These relocated requirements relating to fire protection features are not required to be in the TS under 10 CFR 50.36 or other regulations, or by Section 182a of the Atomic Energy Act, and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. In addition, the staff finds that sufficient regulatory controls exist under 10 CFR 50.48 and 10 CFR 50.59 to

address future changes to these requirements. Accordingly, the staff has concluded that these requirements may be relocated from the TS to the licensee's USAR.

During our review of the TS pages a page number error was detected on Table of Contents page TS-v in that page number changes made in Amendments 101/94 were not carried through with Amendments 103/96. This typographical error introduced by the staff with Amendments 103/96 is hereby corrected by changing the page number for "4.0 Surveillance Requirements," to read "TS.4.0-1." The corrected pages are included with these amendments.

### 5.0 STATE CONSULTATION

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

# 6.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The amendments also change administrative procedures and requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (59 FR 65818). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

### 7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: C. R. Thomas/K. S. West

Date: October 6, 1995