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. 38
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 August 4, 1978, 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.

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 38 , 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

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: August 3, 1979

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. 32
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 August 4, 1978, 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 License No. DPR-60 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 32 , 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

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 3, 1979

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ATTACHMENT TO LICENSE AMENDMENT NOS. 38 AND 32

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

DOCKET NOS. 50-282 AND 50-306

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

Remove	Insert
TS 1-6	TS 1-6
	T, 3.1-19
	TS 3.1-20
TS 3.1-20	TS 3.3-1
Table TS 4.1-1 (p 4 of 4)	Table TS 4.1-1 (p 4 of 4)

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3. Refueling Shutdown

A reactor is in the refueling shutdown condition when a refueling operation is scheduled, the reactor is subcritical by at least 10% Δ k/k and the reactor coolant average temperature is less than 140^{o}F.

Q. Thermal Power

Thermal power of a unit is the total heat transferred from the reactor core to the coolant.

R. Physics Tests

Physics tests are those conducted to measure fundamental characteristics of the core and related instrumentation. Physics tests are conducted such that the core power is sufficiently reduced to allow for the perturbation due to the test and therefore avoid exceeding power distribution limits in Specification 3.10.B.

Low power physics tests are run at reactor powers less than 5% of rated power.

S. Interim Fuel Limits

Interim limits on core power distributions are those values used in the loss-of-coolant accident analysis to demonstrate compliance with (a) the AEC Interim Policy Statement published June 29, 1971, in the Federal Register and (b) the Regulatory Staff's Technical Report on Densification of Light Water Reactor Fuels", published June 14, 1972. The fuel residence time for Unit 1, Cycle 1 shall be limited to 13,000 effective full power hours under design operating conditions.

T. Startup Operation

The process of heating up a reactor above 200^{o}F, making it critical, and bringing it up to power operation.

U. Fire Suppression Water System

The fire suppression water system consists of: Water sources; pumps; and distribution piping with associated sectionalizing isolation valves. Such valves include yard hydrant valves, and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe, or spray system riser.

V. Minimum Pressurization Temperature (MPT)

Reactor coolant system temperature below which reactor coolant system pressure is limited by Figures TS.3.1-1 and TS.3.1-2, Reactor Coolant System Heatup and Cooldown Limitations.

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G. Minimum Conditions for RCS Temperature Less Than MPT

Specification

1. Both pressurizer power operated relief valves (PORV's) shall be operable whenever the RCS temperature is less than the minimum pressurization temperature, except one PORV may be inoperable for seven days. If these conditions are not met, the reactor coolant system must be depressurized and vented to the atmosphere or to the pressurizer relief tank within eight hours.

2. Operability of an overpressure mitigating system PORV requires that the low pressure set point has been selected (enabled), the upstream isolation valve is open and the backup air supply is charged.

3. A reactor coolant pump may be started at RCS temperatures less than MPT only if either of the following conditions is met -

   (a) There is a steam or gas bubble in the pressurizer, or

   (b) The (steam generator minus RCS) temperature difference for either steam generator is less than 50°F.

4. At least one safety injection pump control switch in the control room shall be in pullout whenever RCS temperature is less than MPT, except for conditions satisfying Specification 3.1.G.5.

5. At RCS temperatures less than MPT, both SI pumps may be run for conduct of the integrated SI test only if either of the following conditions is met -

   (a) There is a steam or gas bubble in the pressurizer and the SI pump discharge valves are shut, or

   (b) The reactor vessel head is removed.

BASIS

With RCS temperatures less than MPT, the RCS safety valves and normal setpoints on the pressurizer PORV's do not provide overpressure protection for certain operational transients. The low temperature overpressure mitigating system installed at Prairie Island is designed to prevent pressurizing the RCS above the pressure limits specified in Figures TS.3.1-1 and TS.3.1-2.[1]

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[1] NSP Letter to USNRC, "Reactor Vessel Overpressurization", dated July 22, 1977.

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The system is designed to perform its function in the event of a single failure and is designed to meet the requirements of IEEE-279. The backup air supply provides sufficient air to operate the PORV's following a letdown isolation with one charging pump in operation for a period of ten minutes after receipt of the overpressure alarm. These specifications provide assurance that the overpressure mitigating system will perform its intended function.

Reactor coolant pump start is restricted to RCS conditions where there is pressurizer level indication or low differential temperature across the SG tubes to reduce the probability of positive pressure surges causing overpressurization.

Specification 3.1.G.4 allows use of an SI pump to perform operations required at low RCS temperatures; e.g., raising accumulator levels in order to meet the level requirement of Specification 3.3.A.1.b(2) or ASME Section XI tests of the SI system check valves.

Specification 3.1.G.5 allows use of both SI pumps at low temperatures for conduct of the integrated SI test. In this case, pressurizer level is maintained at less than 50% and the SI pump discharge valves are shut to prevent fluid injection into the RCS. This combination of conditions under strict administrative control assure that overpressurization cannot occur. The option of having the reactor vessel head removed is allowed since in this case RCS overpressurization cannot occur.

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3.3 ENGINEERED SAFETY FEATURES

Applicability

Applies to the operating status of the engineered safety features.

Objective

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

Specification

A. Safety Injection and Residual Heat Removal Systems

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

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

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

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

   c. Two safety injection pumps are operable except that pump control switches in the control room shall meet the requirements of Section 3.1.G whenever the reactor coolant system temperature is less than MPT.

   d. Two residual heat removal pumps are operable.

   e. Two residual heat exchangers are operable.

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

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

TABLE TS.4.1-1

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	Channel Description	Check	Calibrate	Functional Test	Response Test	Remarks
35	Post-Accident Monitoring Instruments	M	NA	NA	NA	Includes all those in FSAR Table 7.7-2 that are not itemized in Table TS.4.1-1.
36.	Steam Exclusion Actuation System	W	R	M	NA	See FSAR Appendix I, Section I.14.6.
37.	Overpressure Mitigation System	NA	R	M	NA	

S - Each Shift

D - Daily

W - Weekly

M - Monthly

Q - Quarterly

R - Each refueling shutdown

P - Prior to each startup if not done previous week

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

NA - Not applicable

* - See Spec 4.1.D