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AUGUST 3 1979

Docket Nos. 50-282
and ~~50-306~~

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

Dear Mr. Mayer:

The Commission has issued the enclosed Amendment Nos. 38 and 32 to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2, respectively, in response to your application dated August 4, 1978.

These amendments incorporate new definitions, limiting conditions for operation and surveillance requirements associated with the reactor cooling system overpressure protection system. During our review of your proposed amendments we found that certain modifications were necessary to meet our requirements. Your staff has agreed to these modifications and they have been incorporated in these amendments.

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

Sincerely,

Original Signed By

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

Cmsln
CCP

Enclosures:
1. Amendment Nos. 38 and 32 to License Nos. DPR-42 and DPR-60
2. Safety Evaluation
3. Notice

cc w/enclosures:
see next page

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

August 3, 1979

Docket Nos. 50-282
and 50-306

Mr. L. O. Mayer, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall - 8th Floor
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Copies of the related Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

A. Schwencer

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

Enclosures:
1. Amendment Nos. 38 and 32 to
   License Nos. DPR-42 and DPR-60
2. Safety Evaluation
3. Notice

cc w/enclosures:
see next page

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 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.

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

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
	TS 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)

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.

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 $50^{\circ}F$.

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]

---

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

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.

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^{o}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.

	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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 38 AND 32 TO FACILITY LICENSE NOS. DPR-42 AND DPR-60

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NOS. 1 AND 2

DOCKET NOS. 50-282 AND 50-306

Introduction

1.0 By letter dated August 4, 1978, Northern States Power Company (NSP) requested amendment of the Technical Specifications amended to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2 (PINGP). The proposed amendments incorporate new definitions, limiting conditions for operation and surveillance requirements associated with the reactor cooling system overpressure protection system. During our review of the proposed amendments we found that certain modifications were necessary to meet our requirements. These modifications were discussed with the NSP staff and they have agreed to the modifications.

By letter dated July 22, 1977[(1)] NSP submitted a plant specific analysis to the Nuclear Regulatory Commission (NRC) in support of the proposed reactor vessel overpressure mitigating system for PINGP. This information supplements other documentation submitted by NSP over the past 12 months.[(2-6)]

The proposed low temperature overpressure protection and mitigating system includes sensors, actuating mechanisms, alarms and valves to prevent a reactor coolant system transient from exceeding the pressure-temperature limits included in the PINGP Technical Specifications as required by Appendix G to 10 CFR Part 50 (Appendix G).

In order to assure proper operation of the overpressure protection system, we requested and NSP has submitted proposed changes to the Technical Specifications that are in accordance with the requirements presented in section 4.2 of this Safety Evaluation Report (SER). Our review of all information submitted by NSP in support of the proposed overpressure mitigating system including the proposed Technical Specification, is completed and we have concluded that the system provides adequate protection from overpressure transients and that the Technical Specifications proposed by NSP are in accordance with the requirements presented in section 4.2 of this SER.

(1) Numbers in parenthesis refer to references which may be found on the last page of this SER.

2.0 Background

Over the last few years, incidents identified as pressure transients have occurred in pressurized water reactors. The term "pressure transients," as used in this report, refers to events during which the Appendix G temperature-pressure limits of the PINGP Technical Specifications, have been exceeded. All of these incidents occurred at relatively low temperature (less than 200°F) where the reactor vessel material toughness (resistance to brittle failure) was reduced.

The "Technical Report on Reactor Vessel Pressure Transients", NUREG 0138 [7] summarizes the technical considerations relevant to this matter, discusses the safety concerns and existing safety margins of operating reactors, and describes the regulatory actions taken to reduce the likelihood of future events at operating reactors. A brief discussion is presented here.

2.1 Vessel Characteristics

Reactor vessels are constructed of high quality steel made to rigid specifications, and fabricated and inspected in accordance with the time-proven rules of the ASME Boiler and Pressure Vessel Code. Steels used are particularly tough at reactor operating conditions. However, if subjected to high pressures at low temperatures, reactor vessel steels are less tough and could possibly fail in a brittle manner. Therefore, power reactors have always operated with restrictions on the pressure allowed during the low temperature portions of startup and shutdown operations.

At operating temperatures, the pressure allowed by Appendix G limits is in excess of the setpoint of currently installed pressurizer code safety valves. However, most operating pressurized water reactors (PWRs) did not have pressure relief devices to prevent a pressure transient during cold conditions from exceeding the lower Appendix G limits that correspond to the lower temperatures involved.

2.2 Regulatory Actions

By letter dated August 13, 1976,[8] we requested that NSP begin efforts to design and install plant systems to mitigate the consequences of pressure transients at low temperatures. We also requested that operating procedures be examined and administrative changes be made to guard against initiating overpressure events. It was our position that proper administrative controls are required to assure safe operation for the period of time prior to installation of the proposed overpressure mitigating hardware.

NSP responded [2,3] with preliminary information describing interim measures to prevent these transients along with some discussions of proposed hardware. The proposed hardware change was to install an optional low pressure actuation set point on the pressurizer air operated relief valves.

NSP participated as a member of a Westinghouse user's group which was formed to support the analysis effort required to verify the adequacy of the proposed system to prevent overpressure transients. Using input data generated by the user's group, Westinghouse performed transient analyses which are used as the basis for plant specific analysis.

We requested additional information concerning the proposed procedural hardware changes. NSP has provided the required responses[5,6] and the final plant specific analyses[1] for PINGP.

2.3 Design Criteria

Through this series of meetings and correspondence with PWR vendors and licensees, we developed a set of criteria for an acceptable overpressure mitigating system, which will prevent reactor vessel pressures in excess of those allowed by Appendix G. Specific criteria for system performance are:

1) Operator Action: No credit can be taken for operator action for 10 minutes after the operator is aware of a transient.

2) Single Failure: The system must be designed to relieve the pressure transient given a single failure in addition to the failure that initiated the pressure transient.

3) Testability: The system must be testable on a periodic basis consistent with the system's employment.

4) Seismic and IEEE 279 Criteria: Ideally, the system should meet seismic Category I and IEEE 279 criteria. The basic objective is that the system should not be vulnerable to a common failure mode that would both initiate a pressure transient and disable the overpressure mitigating system. Events such as loss of instrument air and loss of offsite power must be considered.

We also requested NSP to provide an alarm which monitors the position of the pressurizer relief valve isolation valves, in conjunction with the low pressure setpoint enabling switch, to assure that the overpressure mitigating system is properly aligned for shutdown conditions.

2.4 Design Basis Events

The overpressure incidents that have occurred at PWRs to date have been the result of operator errors or equipment failures. Two varieties of pressure transients can be identified: a mass input type from charging pumps, safety injection pumps and safety injection accumulators; and a heat addition type which causes thermal expansion of the reactor coolant. Sources of heat are steam generators (SG) and the reactor core.

On Westinghouse plants, the most common cause of the overpressure transients of the mass input type has been initiated by isolation of the letdown path without turning off the charging pump during low pressure operations. Other transients occur with lower frequency. We selected those which result in the most rapid pressure increases for analysis. The most limiting mass input transient we identified was inadvertent injection by the largest safety injection pump. The most limiting thermal expansion transient we identified was the start of a reactor coolant pump when a temperature differential exists between the reactor coolant (cooler) secondary coolant (hotter) across the affected steam generator.

Based on the historical record of overpressure transients and the imposition of more effective administrative controls, we have concluded that the limiting events identified above form an acceptable basis for analyses of the proposed overpressure mitigating system.

3.0 System Description and Evaluation

NSP adopted the "Reference Mitigating System" developed by Westinghouse and the user's group. NSP proposed to modify the actuation circuitry of the existing air operated pressurizer relief valves to provide a low pressure set point during startup and shutdown conditions. When the reactor vessel is at low temperatures, with the valves in low pressure set point mode a pressure transient will be terminated below the Appendix G limit by automatic opening of these relief valves at the low pressure set point. A keylock switch is used to enable and disable the low pressure set point of each relief valve. The overpressure mitigating low pressure set point is manually enabled at a RCS temperature of 275°F during plant cooldown and is manually disabled at the same temperature during plant heatup. The system remains enabled whenever the temperature is below 275°F. We find that the capacity of these air operated pressurizer relief valves is sufficient to prevent overpressure and that the proposed overpressure mitigating system is acceptable. Discussion and evaluation of the details of the system proposed by NSP is presented below.

3.1 Air Supply

The power operated relief valves (PORVs) are spring-loaded-closed. Air is required to open the valves, which are supplied by a control air source. To assure ability of the valves to open upon loss of control air, a backup air supply is provided. The backup air supply consists of a Seismic Category I passive air accumulator for each PORV. Each accumulator contains 36 ft^3 of air at a pressure of 80-100 psig and provides air for approximately 15 valve openings. Check valves in the existing air lines prevent depressurizing the accumulators in the event of a ruptured air line.

Pressure switches are installed to indicate loss of normal air supply and gauges are provided to verify operability of the backup supply. We find the backup air supply to be acceptable.

3.2 Electrical Controls

The proposed overall approach to eliminating overpressure events incorporates administrative, procedural and hardware controls with reliance upon the plant operator for the principal line of defense. Preventive administrative/procedural measures include: (1) procedural precautions; (2) de-energizing non-essential components during the cold shutdown mode of operation; and (3) maintaining a non-water solid reactor coolant system condition whenever possible.

The basic design criteria which we apply in determining the adequacy of the electrical, instrumentation and control systems aspects of the low temperature overpressure protection system are:

1. Operator Action - No credit can be taken for operator action until 10 minutes after the operator is aware, through an action alarm, that a pressure transient is in progress.

2. Single Failure Criteria - The pressure protection system shall be designed to protect the reactor vessel given a single failure in addition to the failure that initiated the pressure transient.

3. Testability - The system design shall include provisions for testing on a schedule consistent with the frequency that the system is relief upon for pressure protection.

4. Seismic Design and IEEE-279 Criteria - The pressure protection system shall satisfy both seismic Category I and IEEE-279 criteria.

In addition to complying with the above criteria, NSP has agreed to provide an interlock for the Enable/Disable switch with the position of the isolation valves associated with the PORV. This design feature will ensure a complete pathway from the pressurizer to a quench tank for pressure relief.

3.2.1 System Electrical Description

The overpressure protection system has been designed to comply with both seismic Category I and IEEE Std.279-1971 criteria with the exception of the position alarm provided on the Enable/Disable switch. Since this alarm function is similar to that provided by the Bypass and Inoperable Status Panels, it likewise is not required to satisfy the seismic design requirement.

The proposed overpressure protection system is designed with two independent channels and full testability. Moreover, the installed pressure and temperature instrumentation at PINGP will provide a continuous and permanent record over the full range of both pressure and temperature. A keylock switch in each channel is used to manually enable or disable the low pressure set point of each PORV relief valve. The overpressure protection system's low pressure setpoint is enabled at a reactor coolant system (RCS) temperature of 275°F during plant cooldown and is disabled at the same temperature during plant heatup. The system remains enabled whenever the RCS temperature is below 275°F with the Enable/Disable switch awarm interlocked with the position of the two PORV/isolation valves. We find these design features to be acceptable.

3.2.2 Electrical Equipment Modification

The following equipment will be added to each instrument loop for the overpressure protection system:

a. One dual bistable with an alarm output and a valve actuation output will be counted in spare locations in the RCS analog racks including the test jacks and test panels.

b. Three control relays which provide instrument interface and appropriate train isolation will be mounted in spare locations in the relay room racks.

c. One test jack for calibration signal input.

d. One Enable/Disable switch per channel will be mounted on a panel in the control room.

e. One test panel with a bistable test switch and output indicating lights. A spare point on an existing control room annunciator panel will be used for the alarm.

With the exception of the position alarm provided on the Enable/Disable switch, all electrical equipment added to each instrument loop will be protection grade and meet IEEE Std.279-1971 criteria.

All additional equipment will be installed in accordance with the as-built design criteria for protection grade equipment. We find these electrical modifications acceptable.

3.2.3 Alarm System Design

In accordance with our position on the isolation alarm, NSP has added a design in the system which interlocks the position of the two PORV isolation valves with the Enable/Disable switch alarm. This feature ensures a complete pathway from the pressurizer to a quench tank for pressure relief.

The overpressure system annunciator logic design provides alarms in the control room if: (1) the system is enabled and a high reactor coolant system pressure exists; (2) the system is enabled and a PORV isolation motor operated valve (MOV) is closed; (3) the air supply to the PORV accumulators is not available; or (4) the system is disabled.

3.2.4 PORV Backup Air Supply

Each PORV will have a backup air supply consisting of a seismic Category I passive air accumulator. Check valves to prevent accumulator depressurization in the event of a ruptured air line and pressure switches to indicate loss of air and gauges will be installed to verify backup air supply operability. We find this design acceptable.

3.2.5 Inadvertent Operation of Components

Our position with regard to the inadvertent operation of safety injection system (SIS) components during cold shutdown operations requires the deenergizing of SIS pumps and closure of safety injection (SI) header/discharge valves.

NSP has agreed to deenergize high pressure safety injection (HPSI) pumps and valves and other valves (except for operations allowed by the Technical Specifications) as denoted in the following list:

a. HPSI SIS Pump, #11;
b. HPSI SIS Pump, #12;
c. HPSI Injection Valve # 8806A
d. HPSI Injection Valve # 8806B;
e. Boron Discharge Valve # 8809C;
f. Accumulator Isolation Valve # 8800A; and
g. Accumulator Isolation Valve # 8800B.

All of the valves will be placed in the closed position prior to deenergizing them. The SI pump and valve breakers are located in the Auxiliary Building and the breakers are controlled from the control room. Position indication at the control switch for all valves which have their breakers in the OFF position will be lost. We find this to be acceptable because position indication is available to control room operators from the safety injection monitor panels. Required valve positions, valve breaker positions and pump motor breaker positions are also specified in the plant startup and shutdown procedures and require operator sign off prior to each step during startup or shutdown.

The procedures for starting (jogging) a reactor coolant pump will provide for the RCS ΔT across the affected steam generator to be less than $50^{o}F$ when the RCS is in a water solid condition. This limitation will not be applied when a steam bubble exists in the pressurizer.

3.2.6 High Pressure Alarm

Our position requires that a high pressure alarm for use during low RCS temperature be incorporated such that the operator's attention will be attracted to a transient in progress.

NSP has agreed to install a permanent computer high pressure alarm to satisfy this requirement. When the RCS temperature is less than 275^{o}F, the computer will scan RCS pressure once every minute and, for a pressure above the setpoint, continue to alarm at this interval until the situation is corrected. The temperature input is received from a cold leg reactor thermal detector (RTD) and the pressure input is received from a wide range pressure instrument. The permanent alarm will be located on the control board in the control room. We find this design feature acceptable.

3.2.7 System Testability

Our position with regard to testability is that the system be tested prior to any reliance upon it for overpressure protection. NSP has agreed to perform a system functional test and a setpoint verification test prior to enabling the overpressure protection system during plant cooldown. The system will be calibrated at refueling intervals. The operability of the backup air supply system check valves will also be verified at refueling intervals. We find this to be acceptable.

3.3 Testability

The equipment design includes provision for testing. NSP has stated that testing will be done on a schedule consistent with the frequency that the system is used for pressure protection. A setpoint verification and functional test of the system will be performed prior to enabling the system during plant cooldown. The system will be calibrated at refueling intervals and the operability of the check valves in the backup air supply system will also be verified at refueling intervals. We find the type and frequency of system testing to be acceptable.

3.4 Appendix G

The Appendix G curve submitted by NSP for purposes of overpressure transient analysis is based on ten effective full power years of irradiation. This is based on the zero degree heatup curve since most pressure transients occur during isothermal metal conditions. Further, margins of 45 psig and 10^{o}F are included for possible instrument errors. The Appendix G limit of 100^{o}F according to this curve is 632 psig. We find that use of this curve is acceptable as a basis for evaluation of the overpressure mitigating system performance.

3.5 Set Point Analysis

The one loop version of the LOFTRAN code[9] was used to perform the mass input analyses. The four loop version was used for the heat input analysis. Both versions require some input modeling and initialization changes. LOFTRAN is currently under review by the NRC staff and is judged to be an acceptable code for treating problems of this type.

The results of this analysis are provided in terms of PORV setpoint overshoot. The predicted maximum transient pressure is simply the sum of the overshoot magnitude and the setpoint magnitude. The PORV setpoint is established so that even with the setpoint overshoot the resultant pressure is still below that allowed by Appendix G limits discussed in 3.4 above.

NSP used the following PINGP characteristics in determining the pressure reached for the design basis pressure transients:

SI Pump Flowrate @ 500 psig	83.4 lb/sec
RCS Volume	6,528 ft^3
PORV Opening Time	3 sec
SG Heat Transfer area	51,500 ft^2
Relief Valve setpoint	500 psig

The Unit 1 Appendix G curve for ten years irradiation was presented with the analysis. This curve is conservative with respect to Unit 2.

Westinghouse identified certain assumptions and input parameters as conservative with respect to the analysis. Some of these are listed here.

1) One PORV was assumed to fail.

2) A three-second opening time was assumed for the PORV.

3) The RCS pressure boundary was assumed to be rigid and water solid with respect to expansion.

4) Conservative heat transfer coefficients were assumed for the steam generator.

3.5.1 Mass Input Case

The inadvertent start up of a high pressure safety injection pump with the plant in a cold shutdown condition was selected as the limiting mass input case.

Westinghouse provided NSP with a series of curves based on the LOFTRAN analysis of a generic plant design which indicates PORV setpoint overshoot for this transient system volume, relief valve opening time and relief valve setpoint. These sensitivity analyses were then applied to the PINGP parameters to obtain a conservative estimate of the PORV setpoint overshoot. We find this method of analysis to be acceptable.

Using the Westinghouse methodology, the PINGP PORV setpoint overshoot was determined to be 126 psi. With a relief valve setpoint of 500 psig, a final pressure of 625 psig would be reached for the worst case mass input transient. Since the Unit 1 ten year Appendix G limit (applicable to Unit 2 also) at temperatures above 100°F is above 625 psig, we have concluded that the system performance is acceptable with a 500 psig low pressure relief valve setpoint.

3.5.2 Heat Input Case

The limiting heat input case selected was inadvertent startup of a reactor coolant pump with a primary to secondary temperature differential across the steam generator of 50°F, and with the plant in a water solid condition. For the heat input case, Westinghouse provided the licensee with a series of curves based on the LOFTRAN analysis of a generic plant design to determine the PORV setpoint overshoot as a function of RCS volume, steam generator overall heat transfer capability and initial RCS temperature. For this transient, the reference relief valve selected was assumed to have a total opening of three seconds from the instant the signal to open is received until the valve reached the full open position.

The calculated peak pressure for the heat input transient for a fixed ΔT of 50°F was dependent upon the initial RCS temperature as shown here:

RCS Temperature	Maximum Pressure
100	531
140	561
180	596
250	651

In all these cases, for the given RCS temperature, the Appendix G limits were not exceeded.

We find that the analyses of the limiting mass input and heat input cases show a maximum pressure transient below that allowed by Appendix G limits and are therefore acceptable.

3.6 Implementation Schedule

NSP has made all electrical modifications required for the long term overpressure mitigating system on both units and the backup air supply has been installed for both units.

4.0 Administrative Controls

To supplement the hardware modifications and to limit the magnitude of postulated pressure transients to within the bounds of the analyses provided by NSP, a defense in depth approach has been adopted using procedural and administrative controls. Those specific conditions required to assure that the plant is operated within the bounds of the analysis are included in the proposed Technical Specifications.

4.1 Procedures

A number of provisions for prevention of pressure transients are contained in the PINGP Operating procedures. These include precautions in the RCP operating procedures regarding temperature gradients in the RCS and temperature difference between the RCS and secondary side inventory in the steam generator. Secondary side steam generator temperatures are monitored locally using surface-mounted thermocouples on the outside of the steam generators. RCS temperatures are required by resistance temperature dectectors (RTD) in the loops. The procedures for startup (and jogging) an RCD require the temperature difference between reactor coolant and the steam generator secondary side inventory be less than 50°F before starting a pump associated with that steam generator when the RCS is water solid. This limitation is not applicable when there is a steam bubble in the pressurizer.

Also, shutdown procedures have been revised to include provisions for maintaining a steam bubble in the pressurizer during most cold shutdown conditions. In addition when the RCS is completely depressurized, procedures require that the pressurizer be vented to the pressurizer relief tank. Although the RCS will be in a water solid condition for some time whenever the plant is cooled down for refueling or for RCS maintenance the period of water solid operations has been reduced to the maximum extent possible, compatible with availability considerations (the total period of water solid operation for each cooldown/heatup cycle has been reduced to approximately 48 hours). Both HPSI pumps are de-energized below 200°F to prevent inadvertent starts (except for operations allowed by the Technical Specifications). Above 200°F, the maximum allowable pressure by Appendix G is 1134 psig. The relief capacity of one pressurizer PORV is sufficient to relieve the combined flowrate from both HPSI pumps. Thus, it is acceptable to have both HPSI pumps on line above 200°F.

We find that the procedural and administrative controls described are acceptable, however, we have also required that certain procedural and administrative controls be included in the proposed Technical Specifications. NSP has submitted Technical Specifications in accordance with the guidance in the following section.

4.2 Technical Specifications

To assure operation of the overpressure mitigating system, the licensee has submitted, for staff review, Technical Specifications to be incorporated into the licensees for Prairie Island Units 1 and 2. These specifications are consistent with the intent of the statements listed below. NSP has assured that the Technical Specifications proposed are compatible with other NSP requirements.

1. Both PORVs must 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 primary system must be depressurized and vented to the atmosphere or to the pressurizer relief tank within eight hours.

2. Operability of the overpressure mitigating system requires that the low pressure set point will be selected, the upstream isolation valves open and the backup air supply charged.

3. No more than one HPSI pump may be energized at RCS temperatures below 200°F (except as permitted by Technical Specification 3.1.G.5).

4. A reactor coolant pump may be started only if there is a bubble in the pressurizer or the SG/RCS temperature difference is less than 50°F.

5. The overpressure mitigating system must be tested on a periodic basis consistent with the need for its use.

5.0 Summary

The administrative controls and hardware changes proposed by NSP provide protection of PINGP from pressure transients at low temperatures by reducing the probability of initiation of a transient and by limiting the pressure of such a transient to below the limits set by Appendix G. We find that the overpressure mitigating system meets the criteria established by the NRC and is acceptable as a long term solution to the problem of overpressure transients. Any future revisions of Appendix G limits for PINGP must be considered with respect to this evaluation. At that time, the overpressure mitigating system setpoint should be adjusted accordingly and corresponding adjustments in the license should be made.

The overall low temperature overpressure protection system design in the area of electrical, instrumentation and control (EI&C) is in accordance with those design criteria originally prescribed by the staff and later amended in subsequent staff discussions with NSP.

We find the EI&C aspects of the proposed design to be acceptable on the basis that: (1) the proposed system complies with IEEE Std. 279-1971 criteria and is designed as a seismic Category I system; (2) the system is redundant and satisfies the single failure criterion; (3) the design is such that the system requires no operator action for ten minutes after receipt of an overpressure action alarm; (4) the system is testable on a periodic basis.

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 Section 51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: August 3, 1979

REFERENCES

1. Northern States Power Company letter (Mayer) to NRC (Davis) "Reactor Vessel Overpressurization" dated July 22, 1977.

2. NSP letter (Mayer) to NRC (Ziemann) "Reactor Vessel Overpressurization" dated September 3, 1976.

3. NSP letter (Mayer) to NRC (Ziemann) "Reactor Vessel Overpressurization" dated October 13, 1975.

4. "Pressure Mitigating System Transient Analysis Results" prepared by Westinghouse for the Westinghouse Occurrences Group on Reactor Coolant System Overpressurization, dated July, 1977.

5. NSP letter (Mayer) to NRC (Ziemann) "Reactor Vessel Overpressurization" dated February 25, 1977.

6. NSP letter (Mayer) to NRC (Ziemann) "Reactor Vessel Overpressurization" dated April 4, 1977.

7. "Staff Discussion of Fifteen Technical Issues listed in Attachment G November 3, 1976 Memorandum from Director, NRR to NRR Staff." NUREG-0138, November, 1976.

8. NRC letter (Ziemann) to NSP (Mayer) "Prairie Island Nuclear Generating Plant Units Nos. 1 and 2" dated August 13, 1976.

9. "Loftran Code Description", WCAP 7907, October 1972.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-282 AND 50-306

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY

OPERATNG LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 38 and 32 to Facility Operating License Nos. DPR-42 and DPR-60, issued to the Northern States Power Company (the licensee), which revised Technical Specifications for operation of Unit Nos. 1 and 2 of the Prairie Island Nuclear Generating Plant (the facilities) located in Goodhue County, Minnesota. The amendments are effective as of their date of issuance.

These amendments incorporate new definitions, limiting conditions for operation and surveillance requirements associated with the reactor cooling system overpressure protection system.

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 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 August 4, 1978, (2) Amendment Nos. 38 and 32 to License Nos. DPR-42 and DPR-60, respectively and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. and at the Environmental Conservation Library of the Minneapolis Public Library, 300 Nicollet Mall, Minneapolis, Minnesota 55401. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors, Office of Nuclear Reactor Regulation.

Dated at Bethesda, Maryland, this 3rd day of August, 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

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