DOCKET FILE

50-282

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 4, 1998

Mr. Roger O. Anderson, Director
Nuclear Energy Engineering
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: USE OF A PRESSURE AND TEMPERATURE LIMITS REPORT (TAC NOS. MA1121 AND MA1122)

Dear Mr. Anderson:

The Commission has issued the enclosed Amendment No. 135 to Facility Operating License No. DPR-42 and Amendment No. 127 to Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application dated March 6, 1998, as supplemented March 30, 31, and April 13, 1998.

The amendments update the Technical Specification heatup and cooldown rate curves and extend their reactor vessel fluence limit from the current 20 effective full power years (EFPYs) to a new value of 35 EFPYs, incorporate into Technical Specifications the use of a Pressure and Temperature Limits Report, and change the power-operated relief valves temperature requirement for operability.

Your license amendment request was dated March 6, 1998, and requested issuance of the amendment prior to May 1, 1998, due to expiration of the heatup and cooldown curves at 20 EFPYs for Unit 1. Typically, the staff requires a minimum of 6 months for review and approval of Technical Specification amendments. Submittal of this license amendment request was not timely, particularly considering the complexity of the change involved and the number of staff required to review this amendment. Although the submittal was not timely, the staff was able to complete its review in time to not impact plant operations because your package was complete and addressed all the Pressure Temperature Limits Report requirements set forth in Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits."

DFO1

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, Senior 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. 135 to DPR-42
2. Amendment No. 127 to DPR-60
3. Safety Evaluation

cc w/encl: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 135
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 March 6, 1998, as supplemented March 30, 31, and April 13, 1998, 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 to approve the relocation of certain Technical Specification requirements to licensee-controlled documents, as described in the licensee's application dated March 6, 1998, as supplemented March 30, 31, and April 13, 1998, and evaluated in the staff's safety evaluation dated May 4, 1998. This license is also hereby amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-42 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 135 , 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 issuance, with full implementation within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Beth A. Wetzel

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

Attachment: Changes to the Technical Specifications

Date of Issuance: May 4, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 135

FACILITY OPERATING LICENSE NO. DPR-42

DOCKET NO. 50-282

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-xiii	TS-xiii
TS.1-4	TS.1-4
TS.3.1-2	TS.3.1-2
TS.3.1-4	TS.3.1-4
TS.3.1-5	TS.3.1-5
TS.3.1-6	TS.3.1-6
Figure TS.3.1-1	-
Figure TS.3.1-2	-
TS.3.3-1	TS.3.3-1
TS.3.3-3	TS.3.3-3
Table TS.4.1-1c (p. 4 of 4)	Table TS.4.1-1c (p. 4 of 4)
TS.6.7-4	TS.6.7-4
B.3.1-3	B.3.1-3
B.3.1-5	B.3.1-5
B.3.1-6	B.3.1-6
B.3.3-2	B.3.3-2
-	B.3.3-2a

APPENDIX A TECHNICAL SPECIFICATIONS

LIST OF FIGURES

OPERABLE - OPERABILITY

A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE, or likewise satisfy the requirements of this paragraph.

The OPERABILITY of a system or component shall be considered to be established when: (1) it satisfies the Limiting Conditions for Operation in Specification 3.0, (2) it has been tested periodically in accordance with Specification 4.0 and has met its performance requirements, and (3) its condition is consistent with the two paragraphs above.

OPERATIONAL MODE - MODE

An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table TS.1.1.

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the 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 2% of rated power.

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

The PTLR is the document that provides reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 6.7.A.7. Plant operation within these operating limits is addressed in the individual specifications.

3.1.A.1.c. Reactor Coolant System Average Temperature Below 350°F (and Reactor Coolant Level Above the Reactor Vessel Flange)

(1) Whenever the reactor coolant system average temperature is below 350°F, except during REFUELING, at least two methods for removing decay heat shall be OPERABLE with one in operation* (except as specified in 3.1.A.1.c.(2) below). Acceptable methods for removing decay heat are at least one reactor coolant pump and its associated steam generator; or a residual heat removal loop including a pump and its associated heat exchanger.

(2) With only one OPERABLE method of removing decay heat, initiate prompt action to restore two OPERABLE methods of removing decay heat. If the remaining operable method is an RHR loop, be in COLD SHUTDOWN within 24 hours.

(3) With no OPERABLE methods of removing decay heat, suspend all operations involving a reduction in boron concentration of the reactor coolant system and initiate prompt action to restore one OPERABLE method of removing decay heat.

(4) A reactor coolant pump may be started at RCS temperature less than the Over Pressure Protection System Enable Temperature specified in the PTLR, only if either of the following conditions is met:

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

The (steam generator minus RCS) temperature difference for the steam generator in that loop is less than 50°F.

d. Reactor Coolant Level Below or at the Reactor Vessel Flange

(1) Both residual heat removal loops, each consisting of a pump and its associated heat exchanger, shall be OPERABLE with one in operation* (except as specified in 3.1.A.1.d.(2) below).

(2) With one or both residual heat removal loop(s) inoperable, prompt action shall be taken to restore the inoperable residual heat removal loop(s) to an OPERABLE status. During reduced inventory conditions, a safety injection pump may be run as required to maintain adequate core cooling and RCS inventory in the event of a loss of Residual Heat Removal System cooling.

*All pumps may be shutdown for up to one hour provided the reactor is subcritical, no operations are permitted that would cause dilution of the reactor coolant boron concentration and core outlet temperature is maintained at least 10°F below saturation temperature.

3.1.A.2.c Pressurizer Power Operated Relief Valves

(1) Reactor Coolant System average temperature greater than or equal to 350°F

(a) Reactor coolant system average temperature shall not exceed 350°F, unless two power operated relief valves (PORVs) and their associated block valves are OPERABLE (except as specified in 3.1.A.2.c(1)(b) below).

(b) During STARTUP OPERATION or POWER OPERATION, any one of the following conditions of inoperability may exist for each unit. If OPERABILITY is not restored within the time specified or the required action cannot be completed, be in at least HOT SHUTDOWN within the next 6 hours and reduce reactor coolant system average temperature below 350°F, within the following 6 hours.

1. With one or both PORVs inoperable because of excessive seat leakage, within one hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s).

2. With one PORV inoperable due to causes other than excessive seat leakage, within one hour either restore the PORV to OPERABLE status or close and remove power from the associated block valve. Restore the PORV to OPERABLE status within the following 72 hours.

3. With both PORVs inoperable due to causes other than excessive seat leakage, within one hour either restore at least one PORV to OPERABLE status or close and remove power from the associated block valves and be in at least HOT SHUTDOWN within the next 6 hours and reduce reactor coolant system average temperature below 350°F, within the following 6 hours.

4. With one block valve inoperable, within one hour either restore the block valve to OPERABLE status or place its associated PORV in manual control. Restore the block valve to OPERABLE status within the following 72 hours.

5. With both block valves inoperable, within one hour either restore the block valves to OPERABLE status or place the PORVs in manual control. Restore at least one block valve to OPERABLE status within the next hour.

(2) Reactor Coolant System average temperature greater than or equal to the temperature specified in the PTLR for disabling both safety injection pumps and below the Over Pressure Protection System Enable Temperature specified in the PTLR

With Reactor Coolant System temperature greater than or equal to the temperature specified in the PTLR for disabling both safety injection pumps and less than the Over Pressure Protection System Enable Temperature specified in the PTLR; both pressurizer power operated relief valves (PORVs) shall be OPERABLE (except as specified in 3.1.A.2.c.(2).(a) and 3.1.A.2.c.(2).(b) below) with the Over Pressure Protection System enabled, the associated block valve open, and the associated backup air supply charged.

3.1.A.2.c.(2).(a) One PORV may be inoperable for 7 days. If these conditions cannot be met, depressurize and vent the reactor coolant system through at least a 3 square inch vent within the next 8 hours.

(b) With both PORVs inoperable, complete depressurization and venting of the RCS through at least a 3 square inch vent within 8 hours.

(3) Reactor Coolant System average temperature below the temperature <u>specified in the PTLR for disabling both safety injection pumps</u>

With Reactor Coolant System temperature less than the temperature specified in the PTLR for disabling both safety injection pumps, when the head is on the reactor vessel and the reactor coolant system is not vented through a 3 square inch or larger vent; both Pressurizer power operated relief valves (PORVs) shall be OPERABLE (except as specified in 3.1.A.2.c.(3).(a) and 3.1.A.2.c.(3).(b) below) with the Over Pressure Protection System enabled, the associated block valve open, and the associated backup air supply charged.

(a) One PORV may be inoperable for 24 hours. If these conditions cannot be met, depressurize and vent the reactor coolant system through at least a 3 square inch vent within 8 hours.

(b) With both PORVs inoperable, complete depressurization and venting of the RCS through at least a 3 square inch vent within 8 hours.

3.1.A.3 <u>Reactor Coolant Vent System</u>

a. A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 200°F unless Reactor Coolant Vent System paths from both the reactor vessel head and pressurizer steam space are OPERABLE and closed (except as specified in 3.1.A.3.b and 3.1.A.3.c below).

b. During STARTUP OPERATION and POWER OPERATION, any one of the following conditions of inoperability may exist for each unit provided STARTUP OPERATION is discontinued until OPERABILITY is restored. If any one of these conditions is not restored to an OPERABLE status within 30 days, be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours:

(1) Both of the parallel vent valves in the reactor vessel head vent path inoperable, or

(2) Both of the parallel vent valves in the pressurizer vent path inoperable, or

(3) The vent valve to the pressurizer relief tank discharge line inoperable, or

(4) The vent valve to the containment atmospheric discharge line inoperable.

c. With no Reactor Coolant Vent System path OPERABLE, restore at least one vent path to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

3.1.B. Pressure/Temperature Limits

1. Reactor Coolant System

   a. The Unit 1 and Unit 2 Reactor Coolant Systems (except the pressurizer) temperature, pressure, heatup rates, and cooldown rates shall be maintained within the limits specified in the Pressure and Temperature Limits Report (PTLR).

   b. If these conditions cannot be satisfied, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT SHUTDOWN within the next 6 hours and reduce the reactor coolant system average temperature and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

2. Pressurizer

   a. The pressurizer temperature shall be limited to:

      1. A maximum heatup of 100°F in any 1-hour period.

      2. A maximum cooldown of 200°F in any 1-hour period.

   b. The pressurizer spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

   c. If these conditions cannot be satisfied, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT SHUTDOWN within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

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.

Specifications

A. Safety Injection and Residual Heat Removal Systems

1. A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 200°F unless the following conditions are satisfied (except as specified in 3.3.A.2 below):

   a. The refueling water tank contains not less than 200,000 gallons of water with a boron concentration of at least 2500 ppm.

   b. Each reactor coolant system accumulator shall be OPERABLE when reactor coolant system pressure is greater than 1000 psig.

      OPERABILITY requires:
      (1) The isolation valve is open
      (2) Volume is 1270 ±20 cubic feet of borated water
      (3) A minimum boron concentration of 1900 ppm
      (4) A nitrogen cover pressure of 740 ± 30 psig

   c. Two safety injection pumps are OPERABLE except as specified in Sections 3.3.A.3 and 3.3.A.4.

   d. Two residual heat removal pumps are OPERABLE.

   e. Two residual heat exchangers are OPERABLE.

3.3.A.2.g. The valve position monitor lights or alarms for motor-operated valves specified in 3.3.A.1.g above may be inoperable for 72 hours provided the valve position is verified once each shift.

3. A maximum of one safety injection pump shall be capable of injecting into the RCS whenever RCS temperature is less than the Over Pressure Protection System Enable Temperature specified in the PTLR except that both SI pumps may be run for up to one hour while conducting the integrated SI test** when either of the following conditions is met:

   (a) There is a steam or gas bubble in the pressurizer and an isolation valve between the SI pump and the RCS is shut, or

   (b) The reactor vessel head is removed.

4. No safety injection pumps*** shall be capable of injecting into the RCS whenever RCS temperature is less than the temperature specified in the PTLR for disabling both safety injection pumps (except one or both pumps may be run as specified in 3.3.A.3 and 3.1.A.1.d.(2)).

5. Both reactor coolant system accumulators shall be isolated* whenever RCS temperature is less than the Over Pressure Protection System Enable Temperature specified in the PTLR.

*This specification does not apply whenever the reactor coolant system accumulators are depressurized or the reactor vessel head is removed.

**Other SI system tests and operations may also be conducted under these conditions.

***This specification does not apply whenever the reactor vessel head is removed.

WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code", August, 1985

WCAP-10924-P-A, "Westinghouse Large-Break LOCA Best-Estimate Methodology", December, 1988

WCAP-10924-P-A, Volume 1, Addendum 4, "Westinghouse Large Break LOCA Best Estimate Methodology", August, 1990

XN-NF-77-57 (A), XN-NF-77-57, Supplement 1 (A), "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors Phase II", May, 1981

WCAP-13677, "10 CFR 50.46 Evaluation Model Report: W-COBRA/TRAC 2-Loop Upper Plenum Injection Model Update to Support ZIRLO$_{TM}$ Cladding Options", April 1993 (approved by NRC SE dated November 26, 1993).

NSPNAD-93003-A, "Transient Power Distribution Methodology",(latest approved version)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be supplied upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

6.7.A.7 Pressure and Temperature Limits Report (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following Technical Specification sections; 3.1.A.1.c(4), 3.1.A.2.c(2), 3.1.A.2.c(3), 3.1.B.1.a, 3.3.A.3, 3.3.A.4, 3.3.A.5, and Table 4.1-1C.

b. The analytical methods used to determine the RCS pressure and temperature limits and Cold Overpressure Mitigation System setpoints shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" (Including any exemption granted by NRC to ASME Code Case N-514)

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto. Changes to the curves, setpoints, or parameters in the PTLR resulting from new or additional analysis of beltline material properties will be submitted to the NRC prior to issuance of an updated PTLR.

B. REPORTABLE EVENTS

The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified by a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the Operations Committee and the results of this review shall be submitted to the Safety Audit Committee and the Vice President Nuclear Generation.

TABLE TS.4.1-1C (Page 4 of 4)

TABLE NOTATIONS

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	Shift
D	Daily
W	Weekly
M	Monthly
Q	Quarterly
S/U	Prior to each startup
Y	Yearly
R	Each refueling shutdown
N.A.	Not applicable

TABLE NOTATION

(30) Prior to each startup following shutdown in excess of two days if not done in previous 30 days.

(31) When the reactor trip system breakers are closed and the control rod drive system is capable of rod withdrawal.

(32) Following rod motion in excess of six inches when the computer is out of service.

(33) Transfer logic to Refueling Water Storage Tank.

(34) When either main steam isolation valve is open.

(35) Includes those instruments named in the emergency procedure.

(36) Except for containment hydrogen monitors and refueling water storage tank level which are separately specified in this table.

(37) When RHR is in operation.

(38) When the reactor coolant system average temperature is less than the Over Pressure Protection System Enable Temperature specified in the PTLR.

(39) Whenever CONTAINMENT INTEGRITY is required.

3.1 REACTOR COOLANT SYSTEM

Bases continued

A. Operational Components (continued)

b. Maintaining the integrity of the reactor coolant pressure boundary. This is a function that is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage.

c. Manual control of the block valve to: (1) unblock an isolated PORV to allow it to be used for manual control of reactor coolant system pressure (Item a above), and (2) isolate a PORV with excessive seat leakage (Item b. above).

d. Manual control of a block valve to isolate a stuck-open PORV.

The OPERABILITY of two PORVs or an RCS vent opening of at least 3 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when the RCS temperature is less than the Over Pressure Protection System Enable Temperature specified in the PTLR.

The PORV control switches are three position switches, Open-Auto-Close. A PORV is placed in manual control by placing its control switch in the Closed position.

The RCS safety valves and normal setpoints on the pressurizer PORV's do not provide overpressure protection for certain low temperature operational transients. Inadvertent pressurization of the RCS at temperatures below the Over Pressure Protection System Enable Temperature specified in the PTLR could result in the ASME Appendix G brittle fracture pressure/temperature limits specified in the PTLR being exceeded.

The setpoint for the low temperature overpressure protection system is derived by analysis which models the performance of the low temperature overpressure protection system assuming various mass input and heat input transients. The low temperature overpressure protection system setpoint is updated whenever the RCS heatup and cooldown curves specified in the PTLR are revised.

The 3 square inch RCS vent opening is based on the 2.956 square inch cross sectional flow area of a pressurizer PORV. Because the RCS vent opening specification is based on the flow capacity of a PORV, a PORV maintained in the open position may be utilized to meet the RCS vent requirements.

3.1 REACTOR COOLANT SYSTEM

Bases continued

B. Pressure/Temperature Limits

Appendix G of 10 CFR Part 50, and the ASME Code require that the reactor coolant pressure boundary be designed with sufficient margin to insure that, when stressed under operating, maintenance, testing, and postulated accident conditions, the boundary behaves in a nonbrittle manner, the probability of rapidly propagating fracture is minimized and the design reflects the uncertainties in determining the effects of irradiation on material properties. The pressure/temperature limit curves specified in the PTLR are based on the properties of the most limiting material in either unit's reactor vessel (Unit 1 reactor vessel nozzle to intermediate shell forging circumferential weld) and are effective to 35 EFPY. The curves in the PTLR have not been adjusted for pressure and temperature sensing instruments' uncertainties. The curves incorporated into plant operating procedures will incorporate instrument uncertainties.

The curves define a region where brittle fracture will not occur and are determined from the material characteristics, irradiation effects, pressure stresses and stresses due to thermal gradients across the vessel wall.

Heatup Curves

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. At the inner wall of the vessel, the thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure, which tends to make the coolant temperature limit higher. However, the coolant temperature is higher than the metal temperature in the heatup condition, which tends to reduce the coolant temperature limit. These two phenomena tend to cancel each other. Therefore, an inside-radius pressure-temperature curve based on a comparison of the steady state conditions (i.e., no thermal stresses) and the finite heatup rate conditions must be performed.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are dependent on both the rate of heatup and coolant temperature during the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Therefore, each heatup rate of interest must be analyzed on an individual basis. The heatup limit curve is a composite curve prepared by determining the most conservative case in a point by point comparison, with either the inside steady state curve, the inside finite heatup rate curve, or the outside finite heatup rate curve, for any heatup rate up to 100°F per hour.

3.1 REACTOR COOLANT SYSTEM

Bases (continued)

Cooldown Curves

During cooldown, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from tensile at the inner wall to compressive at the outer wall. The thermal induced tensile stresses at the inner wall are additive to the pressure induced tensile stresses which are already present. Therefore, the controlling location is always the inside wall.

The cooldown limit curves were prepared utilizing the same type of analysis used to calculate the heatup curve except that the controlling location is always the inside wall.

Limit lines for cooldown rates between those presented may be obtained by interpolation.

Criticality Limits

Appendix G of 10 CFR Part 50 requires that for a given pressure, the reactor must not be made critical unless the temperature of the reactor vessel is 40°F above the minimum permissible temperature specified on the heatup curve and above the minimum permissible temperature for the inservice hydrostatic pressure test. For Prairie Island the curves were prepared, requiring that criticality must occur above the maximum permissible temperature for the inservice hydrostatic pressure test.

ASME Code Section XI Inservice Test Limits

The pressure temperature limits for the ASME Code Section XI Inservice Test Limits (hydrostatic pressure test) are less restrictive than the heatup and cooldown curves to allow for the periodic inservice hydrostatic test. These limits are allowed to be less restrictive because the hydrostatic test is based on a 1.5 safety factor versus the 2.0 safety factor built into the heatup and cooldown curves and because the test is run at a constant temperature so the thermal stresses in the vessel are minimal.

Steam Generator Pressure/Temperature Limitations

The limitations on steam generator pressure and temperature ensure that the pressure induced stress in the steam generators do not exceed the maximum allowable fracture toughness stress limits and thus prevent brittle fracture of the steam generator shell.

Pressurizer Limits

Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with ASME Code requirements.

3.3 ENGINEERED SAFETY FEATURES

Bases continued

(1) Assuring with high reliability that the safety system will function properly if required to do so.

(2) Allowance of sufficient time to complete required repairs and testing using safe and proper procedures.

Assuming the reactor has been operating at full RATED THERMAL POWER for at least 100 days, the magnitude of the decay heat decreases as follows after initiating HOT SHUTDOWN.

Time After Shutdown	Decay Heat, % of RATED POWER
1 min.	4.5
30 min.	2.0
1 hour	1.62
8 hours	0.96
48 hours	0.62

Thus, the requirement for core cooling in case of a postulated loss-of-coolant accident while in the shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during POWER OPERATION. Putting the reactor in the HOT SHUTDOWN condition significantly reduced the potential consequences of a loss-of-coolant accident, and also allows more free access to some of the engineered safeguards components in order to effect repairs.

The accumulator and refueling water tank conditions specified are consistent with those assumed in the LOCA analysis (Reference 2).

Specification 3.3.A.3 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 3.3.A.3 also allows use of both SI pumps at low temperatures for conduct of the integrated SI test and other SI system tests and operations providing the pumps run for less than 1 hour. In this case, pressurizer level is maintained at less than 50% and a positive means of isolation is provided between the SI pumps and the RCS to prevent fluid injection into the RCS. This isolation is accomplished by using either a closed manual valve or a closed motor operated valve with the power removed. 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.

Maintaining the safety injection pumps incapable of injecting into the RCS, as specified in 3.3.A.3 and 3.3.A.4, and isolating the accumulators, as specified in 3.3.A.5, will provide assurance that the plant operating conditions will be bounded by the assumptions applied to the determination of the OPPS setpoints in the mass injection transient analysis. These setpoints will actuate the PORVs upon an RCS pressure increase to maintain RCS pressure within the acceptable operating region of the pressure/temperature (brittle fracture) limit curves in the PTLR. The provisions of these specifications are not applicable when the reactor vessel head is removed since in that condition, RCS overpressurization can not occur.

The safety injection pumps are rendered incapable of injecting into the RCS by employing at least two independent means to prevent a pump start such that a single action will not result in an injection into the RCS. This may be accomplished through the pump control switch being placed in pullout with a blocking device installed over the control switch that would prevent an unplanned pump start.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 127
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 March 6, 1998, as supplemented March 30, 31, and April 13, 1998, 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 to approve the relocation of certain Technical Specification requirements to licensee-controlled documents, as described in the licensee's application dated March 6, 1998, as supplemented March 30, 31, and April 13, 1998, and evaluated in the staff's safety evaluation dated May 4 , 1998. This license is also hereby amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-60 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 127, 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 issuance, with full implementation within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Beth A. Wetzel

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

Attachment: Changes to the Technical Specifications

Date of Issuance: May 4, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 127

FACILITY OPERATING LICENSE NO. DPR-60

DOCKET NO. 50-306

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-xiii	TS-xiii
TS.1-4	TS.1-4
TS.3.1-2	TS.3.1-2
TS.3.1-4	TS.3.1-4
TS.3.1-5	TS.3.1-5
TS.3.1-6	TS.3.1-6
Figure TS.3.1-1	-
Figure TS.3.1-2	-
TS.3.3-1	TS.3.3-1
TS.3.3-3	TS.3.3-3
Table TS.4.1-1c (p. 4 of 4)	Table TS.4.1-1c (p. 4 of 4)
TS.6.7-4	TS.6.7-4
B.3.1-3	B.3.1-3
B.3.1-5	B.3.1-5
B.3.1-6	B.3.1-6
B.3.3-2	B.3.3-2
-	B.3.3-2a

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 135

TO FACILITY OPERATING LICENSE NO. DPR-42

AND AMENDMENT NO. 127 TO FACILITY OPERATION LICENSE NO. DPR-60

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By application dated March 6, 1998, as supplemented March 30, 31, and April 13, 1998, the Northern States Power Company (NSP or the licensee) requested amendments to the Technical Specifications (TS) appended to Facility Operating License No. DPR-42 for the Prairie Island Nuclear Generating Plant, Unit 1, and Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Unit 2. The proposed amendments would update the TS heatup and cooldown rate curves and extend their reactor vessel fluence limit from the current 20 effective full power years (EFPYs) to a new value of 35 EFPYs, incorporate into TS the use of a Pressure and Temperature Limits Report (PTLR), and change the power-operated relief valves (PORVs) temperature requirement for operability.

NSP supplemented the March 6, 1998, submittal by letters dated March 30, 31, and April 13, 1998. The supplemental submittals provided additional clarifying information within the scope of the original *Federal Register* notice 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 include TS 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: (1) safety limits, safety system settings and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls, and states also that the Commission may include such additional TS as it finds to be appropriate. However, the regulation does not specify the particular requirements to be included in a plant's TS.

10 CFR 50.36 identifies four criteria to be used in determining whether a particular matter is required to be included in the TS, as follows: (1) installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant

pressure boundary; (2) a process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (3) a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design-basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (4) a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety. As a result, existing TS requirements that fall within or satisfy any of these criteria must be retained in the TS, while those TS requirements that do not fall within or satisfy these criteria may be relocated to other licensee-controlled documents.

3.0 EVALUATION

All components of the reactor coolant system (RCS) are designed to withstand the effects of cyclic loads resulting from system pressure and temperature changes. These loads are introduced by heatup and cooldown operations, power transients, and reactor trips. In accordance with Appendix G to 10 CFR Part 50, TS limit the pressure and temperature changes during RCS heatup and cooldown within the design assumptions and the stress limits for cyclic operation. These limits are defined by the pressure-temperature (P/T) limit curves for heatup and cooldown. Each curve defines an acceptable region for normal operation. The curves are used for operational guidance during heatup and cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The licensee submitted its revised vessel material surveillance reports in a letter dated March 6, 1998. In these reports the licensee documents its evaluation of recently removed vessel material capsule specimens from Prairie Island Units 1 and 2 reactor vessels. These specimens were pulled from both Units 1 and 2 during recent outages and analyzed to determine how much radioactive fluence they have received over the Units 1 and 2 reactor core lives thus far. The licensee then performed a metallurgical analysis which provided a best estimate of the degree of radiation induced embrittlement that will occur in the Prairie Island Units 1 and 2 reactor vessels through 35 EFPYs. The licensee's best estimate results indicated that the Prairie Island vessels will have sufficient protection against the occurrence of pressurized thermal shock (PTS), and will meet the fracture toughness requirements specified in 10 CFR Part 50 Appendix G. The new heatup and cooldown curves which dictate operational limits were developed based on the results of these analyses. The staff finds it acceptable for the licensee to extend the period of applicability of the heatup and cooldown curves from 20 to 35 EFPYs, based on the staff's review of the licensee's analyses.

The licensee used the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Case N-514, and requested an exemption from 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for light water nuclear power reactors for normal operation." The licensee demonstrated that the exemption was warranted under 10 CFR 50.12(a)(2)(ii). The exemption to permit use of Code Case N-514 was granted on April 30, 1998, for Prairie Island Units 1 and 2.

The temperature dependency for PORV operability has been changed from 310 °F to 350 °F. The historical value of 310 °F was used in place of the lower calculated value of 243 °F, which results from the summation of the analytical limit of 225 °F and the indicating instrument channel uncertainty of 18 °F. Analysis has shown that an overpressure protection system (OPPS) enable temperature of 310 °F with the constant value OPPS pressure relief setpoint of 500 psig will protect the 10 CFR Part 50 Appendix G brittle fracture P/T limits as modified by the ASME Code Case N-514 from the minimum unvented RCS temperature through rated operating temperature. Changing the temperature from 310 °F to 350 °F will be consistent with the temperature in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." This provides for a 40 °F temperature band to permit lining up the PORVs from operation with the OPPS enabled (RCS<310 °F) to operation in the normal pressure relief mode (RCS>350 °F).

The licensee-proposed changes to the TS are in accordance with the guidance in Generic Letter (GL) 96-03 "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," as follows:

1. TS 1.0

   The definition contained in GL 96-03 for the term Pressure and Temperature Limits Report (PTLR) is added with two exceptions. The PTLR is not unit specific for Prairie Island, and the Prairie Island specific terminology, Over Pressure Protection System (OPPS) is used in place of the generic letter terminology, low temperature overpressure protection system (LTOP).

2. TS 3.1.A.1.c(4), TS 3.1.A.2.c(2), TS 3.3.A.3, and Table TS.4.1-1c

   The temperature limit, "310°F", has been replaced with the wording, "the Over Pressure Protection System Enable Temperature specified in the PTLR" and the footnote "Valid until 20 EFPY" has been deleted. The OPPS enable temperature value and the associated reactor fluence limit have been relocated to the PTLR.

3. TS 3.1.A.2.c(2), TS 3.1.A.2.c(3), and TS 3.3.A.4

   The temperature limit, "200°F", has been replaced with the wording, "the temperature specified in the PTLR for disabling both safety injection pumps." The mass addition transient calculation in OPPS Setpoint Analysis used 200 °F for an analytical limit and assumed that only three charging pumps and no SI [safety injection] pumps would be available to inject into the RCS when below 200 °F. This analytical limit has been relocated to the PTLR.

4. TS 3.1.A.2.c

   The temperature dependency for PORV limiting conditions for operation has been changed from "310 °F" to "350 °F". The reference to the reactor fluence limit associated with the OPPS Enable Temperature has been removed. The 350 °F temperature value is not related to the brittle fracture limitations on the reactor vessel material.

5. TS 3.1.B.1.a

Changed wording to require that the RCS temperature and pressure limits and heatup and cooldown rates shall be maintained within the limits specified in the PTLR. Deleted the specific wording identifying maximum heatup and cooldown rates.

6. Figure TS.3.1-1 and Figure TS.3.1-2

These figures have been deleted. Revised figures are provided in the PTLR.

7. TS 3.3.A.1.c

Reference to the OPPS enable temperature of "310 °F" has been removed. Reference to TS 3.1.A.d(2) and the footnote have been removed.

8. TS 3.3.A.5

Added the requirements that the RCS accumulators are isolated from the RCS whenever RCS temperature is less than the OPPS enable temperature. This specification will not apply whenever the accumulator is depressurized or the reactor vessel head is removed.

9. TS 6.7.A.7

Added the requirements for the PTLR and reporting in accordance with the guidance of GL 96-03.

10. Basis 3.1.A, Basis 3.1.B and Basis 3.3

The bases for Specifications 3.1.A, 3.1.B, and 3.3 are revised in accordance with the changes made in the specifications stated above.

Relocation of the P/T curves and OPPS setpoints does not eliminate the requirement to operate in accordance with the limits specified in Appendix G to 10 CFR Part 50. The requirement to operate within the limits in the PTLR is specified in, and controlled by, the TS. Only the figures, values, and parameters associated with the P/T limits and OPPS setpoints are to be relocated to the PTLR. In order for the curves to be relocated to a PTLR, a methodology for their development must be reviewed and approved in advance by the NRC. The methodology to be approved by the NRC is to be developed in accordance with GL 96-03. This generic letter provides guidance regarding referencing the methodology and development of the PTLR including, but not limited to, the requirements of Appendix G to 10 CFR Part 50. Since the methodology is referenced in the TS, changes to the methodology must be approved by the NRC. Further, when changes are made to the figures, values, and parameters contained in the PTLR, the PTLR is to be updated and submitted to the NRC upon issuance.

On this basis the NRC staff concludes that the licensee provided an acceptable means of establishing and maintaining the detailed values of the P/T limit curves and OPPS system limits. Further, because plant operation continues to be limited in accordance with the requirements of 10 CFR Part 50 Appendix G and the P/T and OPPS limits in the TS will be established using a methodology approved by the NRC, these changes will not impact plant safety.

The staff also concludes that the above-relocated requirements relating to the P/T limits and OPPS limits are not required to be in the TS under 10 CFR 50.36, 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. Accordingly, the staff concludes that the proposed changes are acceptable and that these requirements may be relocated from the TS to the PTLR.

A detailed discussion of the staff's basis for acceptance of the licensee's proposed methodology is provided in the attached letter from C. A. Carpenter, NRC, to R. Anderson, NSP, "Prairie Island Nuclear Generating Plant, Units 1 and 2, Acceptance for Referencing of Pressure Temperature Limits Report," dated April 29, 1998.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement 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 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 (63 *FR* 14972). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). 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.

6.0 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 Contributor: B. Wetzel

Date: May 4, 1998

Attachment: As stated

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001
April 29, 1998

Mr. Roger O. Anderson, Director
Nuclear Energy Engineering
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2, ACCEPTANCE FOR REFERENCING OF PRESSURE TEMPERATURE LIMITS REPORT (TAC NOS. MA1121 AND MA1122)

Dear Mr. Anderson:

REFERENCES:

1. WCAP-14040-NP-A, Revision 2, Westinghouse Electric Corporation, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," January 15, 1996.
2. NRC Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," January 31, 1996.
3. Letter from Joel P. Sorensen, Northern States Power Company, to NRC Document Control Desk, "Amendment of Technical Specifications to Update the Heatup and Cooldown Rate Curves, Incorporate the Use of a Pressure Temperature Limits Report, and Change the Pressurizer Power Operated Relief Valves Operability Temperature," March 6, 1998. (WCAP-14780 and WCAP-14637 are Attached).
4. Letter from Joel P. Sorensen, Northern States Power Company, to NRC Document Control Desk, "Revised Prairie Island Units 1 and 2 Reactor Vessel Material Surveillance Reports," March 6, 1998. (WCAP-14779, Revision 2, WCAP-14781, Revision 3, WCAP-14613, Revision 2, and WCAP-14638, Revision 2 are Attached).
5. Letter from Joel P. Sorensen, Northern States Power Company, to NRC Document Control Desk, "Response to March 16 and 19, 1998 Requests for Additional Information for License Amendment Request dated March 6, 1998," March 30, 1998.
6. Letter from Joel P. Sorensen, Northern States Power Company, to NRC Document Control Desk, "Response to March 13, 1998 Request for Additional Information for License Amendment Request dated March 6, 1998," March 31, 1998.
7. Letter from Joel P. Sorensen, Northern States Power Company, to NRC Document Control Desk, "Supplement to the License Amendment Request dated March 6, 1998," April 13, 1998.

The NRC staff has completed its review of the pressure temperature (P/T) limit curves and low temperature overpressure protection (LTOP) system limits methodology and the pressure temperature limits report (PTLR) submitted by the Northern States Power Company (NSP). We

Contact: Maggalean W. Weston, (301) 415-3151

Attachment to safety evaluation
dated May 4, 1998

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find the methodology to be acceptable for referencing in the administrative controls section of the Prairie Island Nuclear Generating Plant, Units 1 and 2, Technical Specifications to the extent specified and under the limitations delineated in your submittals and the associated NRC Safety Evaluation, which is enclosed. The Safety Evaluation defines the basis for acceptance of the submittals. Our acceptance applies only to the matters described in the submittals.

The NRC notes that NSP should address the method for assessing the credibility of the reactor pressure vessel surveillance capsule data consistent with the criteria provided in the revised rule (10 CFR 50.61) or in Regulatory Guide 1.99, Revision 2, in future revisions to the PTLR.

The methodology for review relating to the P/T limit curves and the LTOP system limits was provided in the references listed above. WCAP-14040-NP-A provided, in part, the methodology used for determining the acceptance of the Prairie Island methodology.

The methodology in WCAP-14040-NP-A, along with supplements provided by NSP, will be used to calculate future changes to the P/T limit curves. NSP may generate new P/T limit curves in accordance with this methodology without prior approval of the staff. However, changes to the methodology must first be reviewed and approved by the staff. System limits may be subject to audit by the staff through inspections as necessary.

We do not intend to repeat our review of the matters described in the submittals if the submittals appear as references in other license applications relating to your plants, except to ensure that the material in the submittals is still applicable to your plants as indicated in the conclusion section of the Safety Evaluation.

Should our criteria or regulations change so that our conclusions as to the acceptability of the methodology is invalidated, licensees referencing these documents will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the documents without revision of their respective documentation.

Sincerely,

Cynthia A Carpenter

Cynthia A. Carpenter, Director
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosure: Safety Evaluation

cc w/encl: See next page

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REVIEW OF PRESSURE TEMPERATURE LIMITS REPORT AND

METHODOLOGY FOR THE RELOCATION OF THE REACTOR COOLANT SYSTEM

PRESSURE TEMPERATURE LIMIT CURVES AND LOW TEMPERATURE OVERPRESSURE

PROTECTION SYSTEM LIMITS

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By letter dated March 6, 1998 (Reference 3), and supplemented by letters dated March 6, 1998 (Reference 4), March 30, 1998 (Reference 5), March 31, 1998 (Reference 6), and April 13, 1998 (Reference 7), Northern States Power Company (NSP) requested changes to the technical specifications (TS) for the Prairie Island Nuclear Generating Plant, Units 1 and 2. The requested changes included (1) revising the reactor coolant system (RCS) pressure temperature (P/T) limit curves and low temperature overpressure protection (LTOP) system limits, (2) relocating the P/T limit curves and LTOP system limits from the TS to a licensee-controlled document identified as a Pressure Temperature Limits Report (PTLR), and (3) changing the affected limiting conditions for operation and bases accordingly. The P/T limit curves and LTOP system setpoints were developed, in part, using the staff-approved methodology documented in WCAP-14040-NP-A, Revision 2 (Reference 1). These changes are made in accordance with Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," dated January 31, 1996 (Reference 2). Generic Letter 96-03 provides licensees the option to relocate the P/T limit curves and the LTOP system setpoints to a licensee-controlled PTLR provided that the limiting curves and setpoints are developed using an NRC-approved methodology. The licensee proposes to extend the period of applicability of the P/T limit curves and LTOP system setpoints to 35 effective full power years (EFPYs) of reactor operation.

2.0 BACKGROUND

2.1 Neutron Fluence

The fluence evaluation which is the basis for the proposed revised P/T curves is documented in WCAP-14779, Revision 2, and WCAP-14613, Revision 2, for Units 1 and 2, respectively. The evaluation of the pressurized thermal shock is documented in WCAP-14781, Revision 3, and

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WCAP-14638, Revision 2, for Units 1 and 2, respectively. The surveillance capsule reports in WCAP-14779 and WCAP-14613 document the evaluation of capsules S and P for Units 1 and 2, respectively. In addition they document the reevaluation of previously removed capsules (V,P,R) and (V,T, R) for Units I and 2, respectively. The analyses were performed using the BUGLE-93 cross sections in the DOT computer program. The BUGLE-93 cross sections are based on the ENDF/B-VI cross section file which is the staff-recommended file. In addition the analyses utilized the P_3 and S_8 approximations for angular elastic scattering and spacial quadrature, respectively. These approximations are recommended by the staff.

2.2 Pressure Temperature Limits

The methodologies for assessing P/T limits and reactor pressure vessel (RPV) surveillance programs are discussed, in part, in the following documents: (1) 10 CFR Part 50, "Appendix G - Fracture Toughness Requirements"; (2) 10 CFR Part 50, "Appendix H - Reactor Vessel Material Surveillance Program Requirements"; (3) 10 CFR 50.60 - "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation"; (4) 10 CFR 50.61 - "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events"; and (5) Regulatory Guide (RG) 1.99, Revision 2 - "Radiation Embrittlement of Reactor Vessel Materials."

NSP has applied the methodology of WCAP-14040-NP-A, Revision 2, as the general methodology for generating the P/T limit curves for heatup, cooldown, and hydrostatic testing conditions of the PI-1 and PI-2 [Prairie Island] reactor coolant pressure boundaries (RCPBs). Westinghouse Electric Corporation (WEC) submitted this methodology to the staff as its basis for developing the cold overpressure mitigating system setpoints and RCPB heatup and cooldown limit curves for WEC-designed nuclear reactors.

Pursuant to 10 CFR Part 50, Appendix G, the P/T limits and minimum temperatures established for RPVs must meet the requirements for these parameters as set forth in Table 1 of the rule. In addition to the minimum requirements, the P/T limit curves are required to be at least as conservative as those that would be obtained by following the methods of analysis and the safety margins found in the 1989 Edition of the American Society of Mechanical Engineers (ASME) Code, Section XI, Appendix G.

2.3 Low Temperature Overpressure Protection System

The licensee designated the LTOP system as the Over Pressure Protection System (OPPS). The OPPS mitigates overpressure transients at low temperatures so that the integrity of the RCPB is not compromised by violating the 10 CFR Part 50, Appendix G, P/T limits under steady-state operating conditions. Prairie Island Units 1 and 2 OPPS uses the pressurizer power-operated relief valves (PORVs) or an RCS vent with the reactor depressurized to accomplish this function. The system is manually enabled by operators and uses a single setpoint as the lift pressure for the PORVs. The design basis of Prairie Island Units 1 and 2 OPPS considers both mass-addition and heat-addition transients. The mass-addition analyses in the supporting PTLR account for the injection from up to three charging pumps to the RCS in the full range of P/T conditions starting from 68 °F. For an RCS temperature greater than or equal to 200 °F, an inadvertent injection from one safety injection pump and a maximum of three charging pumps are assumed. The heat-addition analyses account for heat input from the secondary side of the steam generators into the RCS upon starting a single reactor coolant pump (RCP) when the RCS temperature is as much as 50 °F lower than the steam generator secondary side temperature. The proposed TS provided restriction in plant operation within the configuration assumed in the analysis for OPPS design.

The Prairie Island Units 1 and 2 proposed design of OPPS including the determination of its enable temperature and the PORV actuation setpoint was established using the staff-approved methodology documented in WCAP-14040-NP-A. Also, the licensee has applied for an exemption from certain requirements of 10 CFR Part 50, Appendix G, and adopted a provision in ASME Code Case N-514 that permits a 10% relaxation of the P/T limits in its design of OPPS.

3.0 EVALUATIONS

3.1 Neutron Fluence

The surveillance capsule reports document the measured and calculated values of four capsules for each plant. The measured to calculated (M/C) ratios of the pressure vessel fluence values are consistent, and their deviation from unity is reasonable. Likewise, the different dosimeter response M/C ratios are consistent and close to unity. Overall, the measured values are slightly higher than the corresponding calculated values. The licensee adopted the calculated values (for each unit) for the estimation of the P/T curves. This is conservative and, therefore, it is acceptable.

The Unit 1 estimated pressure vessel inside diameter fluence value for 35 EFPYs is 3.95×10^{19} n/cm^2. The corresponding value for Unit 2 is 4.18×10^{19} n/cm^2. These values were taken into account in the P/T curves. These values were derived using staff-recommended cross sections and approximations; therefore, we find them acceptable.

3.2 Pressure Temperature (P/T) Limits

3.2.1 Revised P/T Limit Heatup and Cooldown Curves for the PI-1 and PI-2 RPVs

The staff performed an independent analysis using the methods described in 10 CFR Part 50, Appendix G and in Standard Review Plan (SRP) 5.3.2, "Pressure Temperature Limits," in order to determine whether NSP's methods for determining the minimum allowable RCS pressures and temperatures during heatup, cooldown, and hydrostatic testing conditions were conservative relative to the staff's analysis. For the staff's evaluations of the beltline materials, the staff applied the methodology found in 10 CFR Part 50, Appendix G. The staff's analysis methods were consistent with those applied by NSP with the following exceptions:

- The staff's pressure stress equation was based on a simple hoop stress equation.
- The staff's method for evaluating the thermal gradient across the RPV wall was based on a simple steady-state thermal gradient.
- The staff's method for determining the stress intensities due to thermal stresses was based on Figure 4-5 of the Welding Research Council (WRC) Bulletin 175.

The staff followed the criteria of the revised rule 10 CFR 50.61 and RG 1.99, Revision 2, as its basis for calculating the end of life (EOL) 1/4t and 3/4t RT_{NDT} values, and the RT_{PTS} values for the beltline materials in the PI-1 and PI-2 RPVs.

3.2.2 Assessment of the RT_{PTS} Values for the PI-1 and PI-2 Beltline Materials

The staff performed an independent assessment of the RT_{PTS} values for the beltline materials in the Prairie Island reactor vessels. The staff determined that the licensee's calculations of the RT_{PTS} values were in agreement with those that would be generated if the methods of the revised rule 10 CFR 50.61 or RG 1.99, Revision 2, were applied. For purposes of calculating the limiting RT_{PTS} value, the staff verified that the licensee correctly calculated the limiting projected RT_{PTS} values for the PI-1 and PI-2 vessels to be 162 °F and 143 °F, which are the values calculated for the PI-1 nozzle to intermediate shell forging circumferential weld and for the PI-2 upper forging to intermediate forging seam weld W2, respectively. These values are significantly less than the screening criterion of 300 °F as stated in the revised rule 10 CFR 50.61, and indicate that the Prairie Island vessels will continue to satisfy the requirements of the rule throughout the projected lives of the plants.

3.2.3 Assessment of the EOL 1/4t and 3/4t RT_{NDT} Values and the Proposed Heatup, Cooldown, and Hydrostatic Testing Curves for PI-1 and PI-2

The staff also performed an independent assessment of the EOL 1/4t and 3/4t RT_{NDT} values for the beltline materials in the PI-1 and PI-2 reactor vessels, and of the proposed P/T limit curves for the PI-1 and PI-2 reactor vessels during heatup, cooldown, and hydrostatic testing conditions. The assessment of each unit is discussed below.

For the PI-1 reactor vessel, the licensee determined that the most limiting material at the 1/4t and 3/4t locations is the nozzle to intermediate shell circumferential weld. This weld was fabricated using weld wire heat 2269. The licensee calculated an RT_{NDT} value of 154 °F at the 1/4t location and 136 °F at the 3/4t location at 35 EFPYs. The neutron fluence used in the RT_{NDT} calculation was 1.47 X 10^{19} n/cm^2 at the 1/4t location and 0.66 X 10^{19} n/cm^2 at the 3/4t location. The initial RT_{NDT} value for the limiting weld was 0 °F. The margin term used in the calculation for the limiting weld was 66 °F for both the 1/4t and 3/4t locations. This number is consistent with the number that is generated when a generic mean value is used to establish the unirradiated RT_{NDT} for a beltline weld.

The staff performed an independent calculation of the RT_{NDT} values for the limiting material using the methodology in RG 1.99, Revision 2. Based on these calculations, the staff verified that the licensee's limiting material for the PI-1 reactor vessel is the nozzle to intermediate shell circumferential weld that was fabricated using weld wire heat 2269. The staff's calculated RT_{NDT} value for the limiting material agreed with the licensee's calculated RT_{NDT} value at 35 EFPYs. Substituting the RT_{NDT} values for the PI-1 limiting weld into the equations in SRP 5.3.2, the staff verified that the proposed P/T limits satisfy the requirements in paragraph IV.A.2 of Appendix G of 10 CFR Part 50.

For the PI-2 reactor vessel, the licensee determined that the most limiting material at the 1/4t and 3/4t locations is the upper to intermediate shell weld seam. This weld was fabricated using weld wire heat 1752. The licensee calculated an RT_{NDT} value of 134 °F at the 1/4t location and 116 °F at the 3/4t location at 35 EFPYs. The neutron fluence used in the RT_{NDT} calculation was 1.59 X 10^{19} n/cm^2 at the 1/4t location and 0.71 X 10^{19} n/cm^2 at the 3/4t location. The initial RT_{NDT} value for the limiting weld was -13 °F. The margin term used in the calculation for the limiting weld was 56 °F for both the 1/4t and 3/4t locations. This number is consistent with the number that is generated when a generic mean value is used to establish the unirradiated RT_{NDT} for a beltline weld.

The staff performed an independent calculation of the RT_{NDT} values for the limiting material using the methodology in RG 1.99, Revision 2. Based on these calculations, the staff verified that the licensee's limiting material for the PI-2 reactor vessel is the upper to intermediate shell weld seam that was fabricated using weld wire heat 1752. The staff's calculated RT_{NDT} value for the limiting material agreed with the licensee's calculated RT_{NDT} value at 35 EFPYs. Substituting the RT_{NDT} values for the PI-2 limiting weld into the equations in SRP 5.3.2, the staff verified that the proposed P/T limits satisfy the requirements in paragraph IV.A.2 of Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes a minimum temperature at the closure head flange based on the reference temperature for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120 °F for normal operation and by 90 °F for hydrostatic pressure tests and leak tests. The RT_{NDT} values for the limiting flange materials in the PI-1 and PI-2 reactor vessels are -4 °F and -22 °F, respectively. The staff has determined that the proposed P/T limits satisfy the requirement for the closure flange region during normal operation and hydrostatic pressure tests and leak tests.

3.3 Low Pressure Overpressure Protection System

The proposed Limiting Conditions for Operation in TS 3.1.A.2.c require that an OPPS be enabled with two operable PORVs when the RCS temperature is below the OPPS enable temperature. Also, (1) when the RCS temperature is above the temperature in which the safety injection pumps are not disabled, one PORV may be inoperable for 7 days. If these conditions cannot be met, the RCS must be depressurized and vented through at least a 3-square inch vent within the next 8 hours. In the case where both PORVs become inoperable, the RCS must be depressurized and vented through at least a 3-square inch vent within 8 hours, and (2) when the RCS is below the temperature for disabling both safety injection pumps, one PORV may be inoperable for 24 hours. If these conditions cannot be met, the RCS must be depressurized and vented through at least a 3-square inch vent within 8 hours. In the case where both PORVs become inoperable, the RCS must be depressurized and vented through at least a 3 square inch vent within 8 hours. The setpoints related to the design of OPPS and applicable to both Prairie Island Units 1 and 2 are listed in the licensee's PTLR. The staff evaluation of these setpoints is presented below.

3.3.1 Enable Temperature

The OPPS enable temperature is the temperature below which the OPPS is required to be operable. The licensee has established an OPPS enable temperature using the methodology presented in WCAP-14040-NP-A with the provision permitted by ASME Code Case N-514. This Code Case requires the OPPS to be effective at an RCS temperature less than 200 °F or at an RCS temperature corresponding to a reactor vessel metal temperature less than RT_{NDT} + 50 °F at the beltline location (1/4t). Therefore, the licensee proposed to calculate the enable temperature as RT_{NDT} + 50 °F + temperature difference between RCS and metal + Instrument Uncertainties. Using the above equation with limiting material adjusted reference temperature for Unit 1 as input, the calculated minimum enable temperature applicable for both Prairie Island Units 1 and 2 is 243 °F. The licensee proposed an enable temperature of 310 °F that includes an additional margin of 67 °F.

The staff finds that this proposed OPPS enable temperature is conservative with respect to the enable temperature allowed by ASME Code Case N-514 and therefore is acceptable.

3.3.2 Disabling Safety Injection Pump(s) and Isolating Accumulators

In the licensee's analysis for the design of OPPS, it is assumed that when the RCS temperature is below 200 °F, the OPPS will provide adequate protection for a mass addition from a maximum of three charging pumps. When the RCS is between 200 °F and the calculated OPPS enable temperature, the OPPS will provide protection for a mass addition from one safety injection pump plus three charging pumps. To support these analysis assumptions, the licensee proposed requirements in TS 3.3.A and the PTLR to disable one safety injection pump when the RCS is above 218 °F and to disable two safety injection pumps when the RCS temperature is below 218 °F. The setpoint of 218 °F includes an instrument uncertainty of 18 °F. Also, TS 3.3.A requires that both accumulators be isolated when the RCS temperature is below the OPPS enable temperature.

The licensee in its letter dated March 31, 1998, indicated that the safety injection pumps are rendered incapable of injecting into the RCS by employing at least two independent means to prevent a pump start such that a single action will result in an injection into RCS. This may be accomplished through the pump control switch being placed in pullout with a block device installed over the control switch that would prevent an unplanned start. This method of disabling the safety injection pump has been stated in the TS Bases 3.3. We find that the licensee-proposed method of disabling the safety injection pump is acceptable.

3.3.3 PORV Actuation Setpoint

OPPS is designed to mitigate overpressure transients at low temperatures to prevent violating 10 CFR Part 50, Appendix G, P/T limits. Additionally, since overpressure events most likely occur during isothermal conditions in the RCS, the NRC has accepted the use of the steady-state Appendix G limits for the design of the OPPS. The OPPS actuation setpoint is the pressure at which the PORVs will lift, when the OPPS is enabled, to limit the peak RCS pressure during a pressurization transient.

Prairie Island Units 1 and 2 use PORVs to provide pressure relief capacity for the OPPS. The methodology used for determining the PORV actuation setpoint is consistent with the methodology presented in WCAP-14040-NP-A.

The licensee-proposed PORV actuation setpoint of 500 psig in the PTLR was calculated in accordance with the proposed methodology. The licensee, in its submittal dated March 6, 1998, provided a tabulation listing PORV setpoints, transient pressure overshoot, instrumentation uncertainties, pressure difference between the pressure transmitter and the reactor vessel mid-plane with one or two RCPs in operation and corresponding P/T limits under various temperature conditions below the OPPS enable temperature. The data presented in this tabulation confirms that the proposed PORV setpoint of 500 psig will provide adequate protection to the P/T limits established by 10 CFR Part 50, Appendix G, with the provision of ASME Code Case N-514 under steady-state conditions during a design-basis overpressure transient (mass-addition or heat-addition) as described in Section 1.0 of this report. Based on the above discussion, we find the proposed PORV setpoint acceptable.

3.3.4 RCS Vent Size

The proposed TS 3.1.A.2.c specifies a vent size of 3 square inches as an alternative to an operable OPPS when the RCS is depressurized, The bases for the 3-square inch vent is stated in the TS Bases, page B.3.1.-3. It states that the vent size is based on the 2.956-square inch cross sectional flow area of a pressurizer PORV. Since the vent size is compatible with the PORV size, which is sufficient to mitigate a design-basis overpressure transient, we find it acceptable.

4.0 CONCLUSIONS

Based upon the staff evaluations, as discussed in Section 3.0 above, the NRC staff concludes that it is acceptable for NSP to relocate the P/T limit curves and LTOP system limits from the Prairie Island Units 1 and 2 TS to a licensee-controlled PTLR. The proposed heatup, cooldown, and hydrostatic testing curves for PI-1 and PI-2 will expire at 35 EFPYs.

The staff has reviewed the proposed fluence values for Prairie Island Units 1 and 2 for the revision of the P/T curves and finds that the proposed values are conservative and, therefore, acceptable. In addition, the staff has determined that the proposed P/T limit curves are acceptable for use and are consistent with the requirements of Appendix G to 10 CFR Part 50, and Appendix G to Section XI of the ASME Code.

However, since WCAP-14040-NP-A does not address the credibility of RPV surveillance material, the licensee, in future P/T limit evaluations, should address the credibility of the surveillance material.

The staff also reviewed the licensee's analyses related to the proposed setpoints of the OPPS as discussed in Section 3.0 above. The licensee has considered instrument uncertainties in its setpoint calculation using Instrument Society of America S67.04-1994. The staff finds that the licensee's analyses were performed in a manner consistent with the approved methodology and that the results of the analyses conservatively demonstrated that the P/T limits established by 10 CFR Part 50, Appendix G, with provisions provided by ASME Code Case N-514 will be adequately protected with these setpoints and, therefore, finds NSP's analyses acceptable.

The staff has determined that the proposed PTLR meets the criteria of Generic Letter 96-03, and is acceptable to the staff.

5.0 REFERENCES

1. WCAP-14040-NP-A, Revision 2, Westinghouse Electric Corporation, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," January 15, 1996.

2. NRC Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," January 31, 1996.

3. Letter from Joel P. Sorensen, Northern States Power Company, to NRC Document Control Desk, "Amendment of Technical Specifications to Update the Heatup and Cooldown Rate Curves, Incorporate the Use of a Pressure Temperature Limits Report, and Change the Pressurizer Power Operated Relief Valves Operability Temperature," March 6, 1998. (WCAP-14780 and WCAP-14637 are Attached).

4. Letter from Joel P. Sorensen, Northern States Power Company, to NRC Document Control Desk, "Revised Prairie Island Units 1 and 2 Reactor Vessel Material Surveillance Reports," March 6, 1998. (WCAP-14779, Revision 2, WCAP-14781, Revision 3, WCAP-14613, Revision 2, and WCAP-14638, Revision 2 are Attached).

5. Letter from Joel P. Sorensen, Northern States Power Company, to NRC Document Control Desk, "Response to March 16 and 19, 1998 Requests for Additional Information for License Amendment Request dated March 6, 1998, " March 30, 1998.

6. Letter from Joel P. Sorensen, Northern States Power Company, to NRC Document Control Desk, "Response to March 13, 1998 Request for Additional Information for License Amendment Request dated March 6, 1998, " March 31, 1998.

7. Letter from Joel P. Sorensen, Northern States Power Company to NRC Document Control Desk, "Supplement to the License Amendment Request dated March 6, 1998," April 13, 1998.

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