June 21, 1993

Docket Nos. 50-282
and 50-306

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

Dear Mr. Anderson:

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NOS. 1 AND 2 - AMENDMENT NOS. 106 AND 99 TO FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60 (TAC NOS. M77371, M77372, M77444 AND M77445)

The Commission has issued the enclosed Amendment No. 106 to Facility Operating License No. DPR-42 and Amendment No. 99 to the Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated June 25, 1991.

The amendments would revise TS Section 3.1.A.2.C and Table TS 4.1-2A and the associated Bases in response to Generic Letter 90-06. Generic Letter 90-06 provided NRC staff guidance on TS changes that should be implemented to improve the reliability of the Pressurizer Power Operated Relief Valves and the availability of the low temperature overpressure protection system.

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:

Marsha Gamberoni, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 106 to DPR-42
2. Amendment No. 99 to DPR-60
3. Safety Evaluation

cc w/enclosures:
See next page

290030

OFFICE	LA:PDIII-1	PM:PDIII-1	EMEB	OGC	(A)PD:PDIII-1
NAME	~~MShuttleworth~~ CJamerson	MGamberoni jkd:sw	J Norberg	R Bachmann	WDean
DATE	6/2/93	6/18/93	5/12/93	6/7/93	6/21/93
COPY	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO

OFFICIAL RECORD COPY - FILENAME: PI77371.AMD

DF01

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

June 21, 1993

Docket Nos. 50-282
and 50-306

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

Dear Mr. Anderson:

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NOS. 1 AND 2 - AMENDMENT NOS. 106 AND 99 TO FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60 (TAC NOS. M77371, M77372, M77444 AND M77445)

The Commission has issued the enclosed Amendment No. 106 to Facility Operating License No. DPR-42 and Amendment No. 99 to the Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated June 25, 1991.

The amendments would revise TS Section 3.1.A.2.C and Table TS 4.1-2A and the associated Bases in response to Generic Letter 90-06. Generic Letter 90-06 provided NRC staff guidance on TS changes that should be implemented to improve the reliability of the Pressurizer Power Operated Relief Valves and the availability of the low temperature overpressure protection system.

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,

Marsha Gamberoni

Marsha Gamberoni, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 106 to DPR-42
2. Amendment No. 99 to DPR-60
3. Safety Evaluation

cc w/enclosures:
See next page

Mr. R. Anderson
Northern States Power Company

Prairie Island Nuclear Generating
Plant

cc:

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

Mr. E. L. Watzl, Site General Manager
Prairie Island Nuclear Generating
Plant
Northern States Power Company
Route 2
Welch, Minnesota 55089

Lisa R. Tiegel
Assistant Attorney General
Environmental Protection Division
Suite 200
520 Lafayette Road
St. Paul, Minnesota 55155

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

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Mr. Jeff Cole, Auditor/Treasurer
Goodhue County Courthouse
Red Wing, Minnesota 55066

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

DATED: June 21, 1993

AMENDMENT NO. 106 TO FACILITY OPERATING LICENSE NO. DPR-42-PRAIRIE ISLAND UNIT 1
AMENDMENT NO. 99 TO FACILITY OPERATING LICENSE NO. DPR-60-PRAIRIE ISLAND UNIT 2

Docket File
NRC & Local PDRs
PDIII-1 Reading
PI Plant File
J. Roe
J. Zwolinski
L. Marsh
C. Jamerson
M. Gamberoni
D. Pickett
OGC-WF
D. Hagan, (MNBB 3206)
G. Hill (2), P-137
Wanda Jones, (MNBB-3701)
C. Grimes (11/F/23)
ACRS (10)
GPA/PA
OC/LFMB
W. Shafer, R-III
J. Norberg (7/E/23)

cc: Plant Service list

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 106
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 June 25, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 106, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Hall for

William M. Dean, Acting Director
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 21, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 106

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.3.1-4	TS.3.1-4
TS.3.1-5	TS.3.1-5
Table TS.4.1-2A	Table TS.4.1-2A
B.3.1-2	B.3.1-2
B.3.1-3	B.3.1-3
B.3.1-4	B.3.1-4
B.3.1-5	B.3.1-5
B.3.1-6	B.3.1-6
B.3.1-7	B.3.1-7
B.3.1-8	B.3.1-8
B.3.1-9	B.3.1-9
-	B.3.1-10

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

(1) Reactor Coolant System average temperature greater than or equal to 310°F*

(a) Reactor coolant system average temperature shall not exceed 310°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 310°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 310°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 200°F and below 310°F*

With Reactor Coolant System temperature greater than or equal to 200°F and less than 310°F*; 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.

*Valid until 20 EFPY

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 200°F

With Reactor Coolant System temperature less than 200°F 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 Reactor Coolant Vent System

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.

MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

	Test	Frequency	FSAR Sect. Reference
1. Control Rod Assemblies	Rod Drop Times of full length rods	All rods during each refueling shutdown or following each removal of the reactor vessel head; affected rods following maintenance on or modification to the control rod drive system which could affect performance of those specified rods	7
2. Control Rod Assemblies	Partial movement of all rods	Every 2 weeks	7
3. Pressurizer Safety Valves	Set point	Per ASME Code, Section XI Inservice Testing Program	-
4. Main Steam Safety Valves	Set point	Per ASME Code, Section XI Inservice Testing Program	-
5. Reactor Cavity	Water Level	Prior to moving fuel assemblies or control rods and at least once every day while the cavity is flooded.	
6. Pressurizer PORV Block Valves	Functional	Quarterly, unless the block valve has been closed per Specification 3.1.A.2.c.(1).(b).2 or 3.1.A.2.c.(1).(b).3.	-
7. Pressurizer PORVs	Functional	Every 18 months	-
8. Deleted			
9. Primary System Leakage	Evaluate	Daily	4
10. Deleted			
11. Turbine stop valves, governor valves, and intercept valves. (Part of turbine overspeed protection)	Functional	See (1)	10

(1) Turbine stop valves, governor valves and intercept valves are to be tested at a frequency consistent with the methodology presented in WCAP-11525 "Probabilistic Evaluation of Reduction in Turbine Valve Test Frequency", and in accordance with the established NRC acceptance criteria for the probability of a turbine missile ejection incident of 1.0×10^{-5} per year. In no case shall the turbine valve test interval exceed one year.

3.1 REACTOR COOLANT SYSTEM

Bases continued

A. Operational Components (continued)

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.

The pressurizer is needed to maintain acceptable system pressure during normal plant operation, including surges that may result following anticipated transients. Each of the pressurizer safety valves is designed to relieve 325,000 lbs per hour of saturated steam at the valve set point. Below 350°F and 450 psig in the reactor coolant system, the residual heat removal system can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve therefore provides adequate defense against over-pressurization of the reactor coolant system for reactor coolant temperatures less than 350°F. The combined capacity of both safety valves is greater than the maximum surge rate resulting from complete loss of load (Reference 1).

The requirement that two groups of pressurizer heaters be OPERABLE provides assurance that at least one group will be available during a loss of offsite power to maintain natural circulation. Backup heater group "A" is normally supplied by one safeguards bus. Backup heater group "B" can be manually transferred within minutes to the redundant safeguards bus. Tests have confirmed the ability of either group to maintain natural circulation conditions.

The pressurizer power operated relief valves (PORVs) operate to relieve reactor coolant system pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The PORVs are pneumatic valves operated by instrument air. They fail closed on loss of air or loss of power to their DC solenoid valves. The PORV block valves are motor operated valves supplied by the 480 volt safeguards buses.

The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions:

a. Manual control of PORVs to control reactor coolant pressure. This is a function that is used for the steam generator tube rupture accident and for plant shutdown.

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.

3.1 REACTOR COOLANT SYSTEM

Bases continued

A. Operational Components (continued)

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 310°F*.

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 minimum pressurization temperature (310°F *) is determined from Figure TS.3.1-1 and is the temperature equivalent to the RCS safety relief valve setpoint pressure. 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 310°F* could result in the limits of Figures TS.3.1-1 and TS.3.1-2 being exceeded. Thus the low temperature overpressure protection system, which is designed to prevent pressurizing the RCS above the pressure limits specified in Figures TS.3.1-1 and TS.3.1-2, is enabled at 310°F*. Above 310°F* the RCS safety valves would limit the pressure increase and would prevent the limits of Figures TS.3.1-1 and TS.3.1-2 from 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 (Figures TS.3.1-1 and TS.3.1-2) 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.

*Valid until 20 EFPY

3.1 REACTOR COOLANT SYSTEM

Bases continued

A. Operational Components (continued)

The OPERABILITY of the low temperature overpressure protection system is determined on the basis of their being capable of performing the function to mitigate an overpressure event during low temperature operation. OPERABILITY of a low temperature overpressure protection 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.

The low temperature overpressure protection 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 PORVs 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 low temperature overpressure protection system will perform its intended function.

The reactor coolant vent system is provided to exhaust noncondensible gases from the reactor coolant system that could inhibit natural circulation core cooling. The OPERABILITY of at least one vent path from both the reactor vessel head and pressurizer steam space ensures the capability exists to perform this function.

The vent path from the reactor vessel head and the vent path from the pressurizer each contain two independently emergency powered, energize to open, valves in parallel and connect to a common header that discharges either to the containment atmosphere or to the pressurizer relief tank. The lines to the containment atmosphere and pressurizer relief tank each contain an independently emergency powered, energize to open, isolation valve. This redundancy provides protection from the failure of a single vent path valve rendering an entire vent path inoperable. An inoperable vent path valve is defined as a valve which cannot be opened or whose position is unknown.

A flow restriction orifice in each vent path limits the flow from an inadvertent actuation of the vent system to less than the flow of the reactor coolant makeup system.

References

1. USAR, Section 14.4.8.

2. Testimony by J Knight in the Prairie Island Public Hearing on January 28, 1975.

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

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. Figures TS.3.1-1 and 2 have been developed (Reference 1) in accordance with these regulations. The curves are based on the properties of the most limiting material in either unit's reactor vessel (Unit 1 reactor vessel weld W-3) and are effective to 20 EFPY. The curves have been adjusted for possible errors in the pressure and temperature sensing instruments.

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. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

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 the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. For the cases in which the outer wall of the vessel becomes the stress controlling location, 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, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour.

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.

3.1 REACTOR COOLANT SYSTEM

Bases (continued)

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.

The criticality limit specified in Figure TS.3.1-1 provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization whenever the reactor vessel is in the nil ductility temperature range. Heatup to this temperature will be accomplished by operating the reactor coolant pumps and by the pressurizer heaters. The pressurizer heater and associated power cables have been sized for continuous operation at full heater power.

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.

Reference

1. USAR Section 4.2

3.1 REACTOR COOLANT SYSTEM

Bases continued

C. Reactor Coolant System Leakage

Leakage from the reactor coolant system is collected in the containment or by other systems. These systems are the main steam system, condensate and feedwater system and the chemical and volume control system.

Detection of leaks from the reactor coolant system is by one or more of the following (Reference 1):

1. An increased amount of makeup water required to maintain normal level in the pressurizer.

2. A high temperature alarm in the leakoff piping provided to collect reactor head flange leakage.

3. Containment sump water level indication.

4. Containment pressure, temperature, and humidity indication.

If there is significant radioactive contamination of the reactor coolant, the radiation monitoring system provides a sensitive indication of primary system leakage. Radiation monitors which indicate primary system leakage include the containment air particulate and gas monitors, the area radiation monitors, the condenser air ejector monitor, the component cooling water monitor, and the steam generator blowdown monitor (Reference 2).

A leak rate of 1 gpm corresponds to a through wall crack less than 0.6 inches long based on test data. Steam generator tubes having a 0.6-inch long through-wall crack have been shown to resist failure at pressures resulting from normal operation, LOCA, or steam line break accidents (Reference 3).

Specification 3.1.C.3 specifies actions to be taken in the event of failure or excessive leakage of a check valve which isolates the high pressure reactor coolant system from the low pressure RHR system piping.

References

1. USAR, Section 6.5
2. USAR, Section 7.5.1
3. Testimony by J Knight in the Prairie Island public hearing on January 28, 1975, pp 13-17.

3.1 REACTOR COOLANT SYSTEM

Bases continued

D. Maximum Coolant Activity

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Prairie Island site, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

Specification 3.1.D.2, permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcuries/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure TS.3.1-3, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 microcuries/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure TS.3.1-3 should be minimized since the activity levels allowed by Figure TS.3.1-3 increase the 2 hour thyroid dose at the SITE BOUNDARY by a factor of up to 20 following a postulated steam generator tube rupture.

Reducing RCS temperature to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements in Table TS.4.1-2B provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action.

3.1 REACTOR COOLANT SYSTEM

Bases continued

E. Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration

By maintaining the oxygen, chloride and fluoride concentrations in the reactor coolant below the normal steady-state operation limits specified, the integrity of the reactor coolant system is assured under all operating conditions (Reference 1).

If these steady-state limits are exceeded, measures can be taken to correct the condition during reactor operation, e.g., replacement of ion exchange resin or adjustment of the hydrogen concentration in the volume control tank (Reference 2). Because of the time dependent nature of any adverse effects from oxygen, chloride, and fluoride concentrations in excess of the limits, it is unnecessary to shut down immediately since the conditions for corrective action to restore concentrations within the steady-state limits has been established. If the corrective action has not been effective at the end of the 24-hour period, then the reactor will be brought to the COLD SHUTDOWN condition and the corrective action will continue.

The effects of contaminants in the reactor coolant are temperature dependent. It is consistent, therefore, to permit transient concentrations to exist for 48 hours for coolant temperatures less than 250°F and still provide the assurance the integrity of the primary coolant system will be maintained.

In order to restore the contaminant concentrations to within specification limits in the event such limits were exceeded, mixing of the primary coolant with the reactor coolant pumps may be required. This will result in a small heatup of short duration and will not increase the average coolant temperature above 250°F.

References

1. USAR, Section 4.5.2
2. USAR, Section 10.2.3

3.1 REACTOR COOLANT SYSTEM

Bases continued

F. Isothermal Temperature Coefficient (ITC)

At the beginning of a fuel cycle the moderator temperature coefficient has its most positive or least negative value. As the boron concentration is reduced throughout the fuel cycle, the moderator temperature coefficient becomes more negative. The isothermal temperature coefficient is defined as the reactivity change associated with a unit change in the moderator and fuel temperatures. Essentially, the isothermal temperature coefficient is the sum of the moderator and fuel temperature coefficients. This coefficient is measured directly during low power PHYSICS TESTS in order to verify analytical prediction. The units of the isothermal temperature coefficient are pcm/°F, where 1pcm = 1×10^{-5} $\Delta k/k$,

For extended optimum fuel burnup it is necessary to either load the reactor with burnable poisons or increase the boron concentration in the reactor coolant system. If the latter approach is emphasized, it is possible that a positive isothermal temperature coefficient could exist at beginning of cycle (BOC). Safety analyses verify the acceptability of the isothermal temperature coefficient for limits specified in 3.1.F. Other conditions, e.g., higher power or partial rod insertion would cause the isothermal coefficient to have a more negative value. These analyses demonstrate that applicable criteria in the NRC Standard Review Plan (NUREG 75/087) are met.

Physics measurements and analyses are conducted during the reload startup test program to (1) verify that the plant will operate within safety analyses assumptions and (2) establish operational procedures to ensure safety analyses assumptions are met. The 3.1.F requirements are waived during low power PHYSICS TESTS to permit measurement of reactor temperature coefficient and other physics design parameters of interest. Special operating precautions will be taken during these PHYSICS TESTS. In addition, the strong negative Doppler coefficient (Reference 1) and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

References:

1. FSAR Figure 3.2.10

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 99
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 June 25, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 99 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Hall for

William M. Dean, Acting Director
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 21, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 99

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.3.1-4	TS.3.1-4
TS.3.1-5	TS.3.1-5
Table TS.4.1-2A	Table TS.4.1-2A
B.3.1-2	B.3.1-2
B.3.1-3	B.3.1-3
B.3.1-4	B.3.1-4
B.3.1-5	B.3.1-5
B.3.1-6	B.3.1-6
B.3.1-7	B.3.1-7
B.3.1-8	B.3.1-8
B.3.1-9	B.3.1-9
-	B.3.1-10

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

(1) Reactor Coolant System average temperature greater than or equal to 310°F*

(a) Reactor coolant system average temperature shall not exceed 310°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 310°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 310°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 200°F and below 310°F*

With Reactor Coolant System temperature greater than or equal to 200°F and less than 310°F*; 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.

*Valid until 20 EFPY

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 200°F

With Reactor Coolant System temperature less than 200°F 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 Reactor Coolant Vent System

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.

MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

	Test	Frequency	FSAR Sect. Reference
1. Control Rod Assemblies	Rod Drop Times of full length rods	All rods during each refueling shutdown or following each removal of the reactor vessel head; affected rods following maintenance on or modification to the control rod drive system which could affect performance of those specified rods	7
2. Control Rod Assemblies	Partial movement of all rods	Every 2 weeks	7
3. Pressurizer Safety Valves	Set point	Per ASME Code, Section XI Inservice Testing Program	-
4. Main Steam Safety Valves	Set point	Per ASME Code, Section XI Inservice Testing Program	-
5. Reactor Cavity	Water Level	Prior to moving fuel assemblies or control rods and at least once every day while the cavity is flooded.	
6. Pressurizer PORV Block Valves	Functional	Quarterly, unless the block valve has been closed per Specification 3.1.A.2.c.(1).(b).2 or 3.1.A.2.c.(1).(b).3.	-
7. Pressurizer PORVs	Functional	Every 18 months	-
8. Deleted			
9. Primary System Leakage	Evaluate	Daily	4
10. Deleted			
11. Turbine stop valves, governor valves, and intercept valves. (Part of turbine overspeed protection)	Functional	See (1)	10

(1) Turbine stop valves, governor valves and intercept valves are to be tested at a frequency consistent with the methodology presented in WCAP-11525 "Probabilistic Evaluation of Reduction in Turbine Valve Test Frequency", and in accordance with the established NRC acceptance criteria for the probability of a turbine missile ejection incident of 1.0×10^{-5} per year. In no case shall the turbine valve test interval exceed one year.

3.1 REACTOR COOLANT SYSTEM

Bases continued

A. Operational Components (continued)

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.

The pressurizer is needed to maintain acceptable system pressure during normal plant operation, including surges that may result following anticipated transients. Each of the pressurizer safety valves is designed to relieve 325,000 lbs per hour of saturated steam at the valve set point. Below 350°F and 450 psig in the reactor coolant system, the residual heat removal system can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve therefore provides adequate defense against over-pressurization of the reactor coolant system for reactor coolant temperatures less than 350°F. The combined capacity of both safety valves is greater than the maximum surge rate resulting from complete loss of load (Reference 1).

The requirement that two groups of pressurizer heaters be OPERABLE provides assurance that at least one group will be available during a loss of offsite power to maintain natural circulation. Backup heater group "A" is normally supplied by one safeguards bus. Backup heater group "B" can be manually transferred within minutes to the redundant safeguards bus. Tests have confirmed the ability of either group to maintain natural circulation conditions.

The pressurizer power operated relief valves (PORVs) operate to relieve reactor coolant system pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The PORVs are pneumatic valves operated by instrument air. They fail closed on loss of air or loss of power to their DC solenoid valves. The PORV block valves are motor operated valves supplied by the 480 volt safeguards buses.

The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions:

a. Manual control of PORVs to control reactor coolant pressure. This is a function that is used for the steam generator tube rupture accident and for plant shutdown.

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.

3.1 REACTOR COOLANT SYSTEM

Bases continued

A. Operational Components (continued)

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 310°F*.

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 minimum pressurization temperature (310°F *) is determined from Figure TS.3.1-1 and is the temperature equivalent to the RCS safety relief valve setpoint pressure. 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 310°F* could result in the limits of Figures TS.3.1-1 and TS.3.1-2 being exceeded. Thus the low temperature overpressure protection system, which is designed to prevent pressurizing the RCS above the pressure limits specified in Figures TS.3.1-1 and TS.3.1-2, is enabled at 310°F*. Above 310°F* the RCS safety valves would limit the pressure increase and would prevent the limits of Figures TS.3.1-1 and TS.3.1-2 from 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 (Figures TS.3.1-1 and TS.3.1-2) 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.

*Valid until 20 EFPY

3.1 REACTOR COOLANT SYSTEM

Bases continued

A. Operational Components (continued)

The OPERABILITY of the low temperature overpressure protection system is determined on the basis of their being capable of performing the function to mitigate an overpressure event during low temperature operation. OPERABILITY of a low temperature overpressure protection 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.

The low temperature overpressure protection 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 PORVs 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 low temperature overpressure protection system will perform its intended function.

The reactor coolant vent system is provided to exhaust noncondensible gases from the reactor coolant system that could inhibit natural circulation core cooling. The OPERABILITY of at least one vent path from both the reactor vessel head and pressurizer steam space ensures the capability exists to perform this function.

The vent path from the reactor vessel head and the vent path from the pressurizer each contain two independently emergency powered, energize to open, valves in parallel and connect to a common header that discharges either to the containment atmosphere or to the pressurizer relief tank. The lines to the containment atmosphere and pressurizer relief tank each contain an independently emergency powered, energize to open, isolation valve. This redundancy provides protection from the failure of a single vent path valve rendering an entire vent path inoperable. An inoperable vent path valve is defined as a valve which cannot be opened or whose position is unknown.

A flow restriction orifice in each vent path limits the flow from an inadvertent actuation of the vent system to less than the flow of the reactor coolant makeup system.

References

1. USAR, Section 14.4.8.

2. Testimony by J Knight in the Prairie Island Public Hearing on January 28, 1975.

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

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. Figures TS.3.1-1 and 2 have been developed (Reference 1) in accordance with these regulations. The curves are based on the properties of the most limiting material in either unit's reactor vessel (Unit 1 reactor vessel weld W-3) and are effective to 20 EFPY. The curves have been adjusted for possible errors in the pressure and temperature sensing instruments.

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. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

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 the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. For the cases in which the outer wall of the vessel becomes the stress controlling location, 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, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour.

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.

3.1 REACTOR COOLANT SYSTEM

Bases (continued)

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.

The criticality limit specified in Figure TS.3.1-1 provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization whenever the reactor vessel is in the nil ductility temperature range. Heatup to this temperature will be accomplished by operating the reactor coolant pumps and by the pressurizer heaters. The pressurizer heater and associated power cables have been sized for continuous operation at full heater power.

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.

Reference

1. USAR Section 4.2

3.1 REACTOR COOLANT SYSTEM

Bases continued

C. Reactor Coolant System Leakage

Leakage from the reactor coolant system is collected in the containment or by other systems. These systems are the main steam system, condensate and feedwater system and the chemical and volume control system.

Detection of leaks from the reactor coolant system is by one or more of the following (Reference 1):

1. An increased amount of makeup water required to maintain normal level in the pressurizer.

2. A high temperature alarm in the leakoff piping provided to collect reactor head flange leakage.

3. Containment sump water level indication.

4. Containment pressure, temperature, and humidity indication.

If there is significant radioactive contamination of the reactor coolant, the radiation monitoring system provides a sensitive indication of primary system leakage. Radiation monitors which indicate primary system leakage include the containment air particulate and gas monitors, the area radiation monitors, the condenser air ejector monitor, the component cooling water monitor, and the steam generator blowdown monitor (Reference 2).

A leak rate of 1 gpm corresponds to a through wall crack less than 0.6 inches long based on test data. Steam generator tubes having a 0.6-inch long through-wall crack have been shown to resist failure at pressures resulting from normal operation, LOCA, or steam line break accidents (Reference 3).

Specification 3.1.C.3 specifies actions to be taken in the event of failure or excessive leakage of a check valve which isolates the high pressure reactor coolant system from the low pressure RHR system piping.

References

1. USAR, Section 6.5
2. USAR, Section 7.5.1
3. Testimony by J Knight in the Prairie Island public hearing on January 28, 1975, pp 13-17.

3.1 REACTOR COOLANT SYSTEM

Bases continued

D. Maximum Coolant Activity

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Prairie Island site, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

Specification 3.1.D.2, permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcuries/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure TS.3.1-3, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 microcuries/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure TS.3.1-3 should be minimized since the activity levels allowed by Figure TS.3.1-3 increase the 2 hour thyroid dose at the SITE BOUNDARY by a factor of up to 20 following a postulated steam generator tube rupture.

Reducing RCS temperature to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements in Table TS.4.1-2B provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action.

3.1 REACTOR COOLANT SYSTEM

Bases continued

E. Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration

By maintaining the oxygen, chloride and fluoride concentrations in the reactor coolant below the normal steady-state operation limits specified, the integrity of the reactor coolant system is assured under all operating conditions (Reference 1).

If these steady-state limits are exceeded, measures can be taken to correct the condition during reactor operation, e.g., replacement of ion exchange resin or adjustment of the hydrogen concentration in the volume control tank (Reference 2). Because of the time dependent nature of any adverse effects from oxygen, chloride, and fluoride concentrations in excess of the limits, it is unnecessary to shut down immediately since the conditions for corrective action to restore concentrations within the steady-state limits has been established. If the corrective action has not been effective at the end of the 24-hour period, then the reactor will be brought to the COLD SHUTDOWN condition and the corrective action will continue.

The effects of contaminants in the reactor coolant are temperature dependent. It is consistent, therefore, to permit transient concentrations to exist for 48 hours for coolant temperatures less than 250°F and still provide the assurance the integrity of the primary coolant system will be maintained.

In order to restore the contaminant concentrations to within specification limits in the event such limits were exceeded, mixing of the primary coolant with the reactor coolant pumps may be required. This will result in a small heatup of short duration and will not increase the average coolant temperature above 250°F.

References

1. USAR, Section 4.5.2
2. USAR, Section 10.2.3

3.1 REACTOR COOLANT SYSTEM

Bases continued

F. Isothermal Temperature Coefficient (ITC)

At the beginning of a fuel cycle the moderator temperature coefficient has its most positive or least negative value. As the boron concentration is reduced throughout the fuel cycle, the moderator temperature coefficient becomes more negative. The isothermal temperature coefficient is defined as the reactivity change associated with a unit change in the moderator and fuel temperatures. Essentially, the isothermal temperature coefficient is the sum of the moderator and fuel temperature coefficients. This coefficient is measured directly during low power PHYSICS TESTS in order to verify analytical prediction. The units of the isothermal temperature coefficient are pcm/°F, where 1pcm = 1×10^{-5} $\Delta k/k$,

For extended optimum fuel burnup it is necessary to either load the reactor with burnable poisons or increase the boron concentration in the reactor coolant system. If the latter approach is emphasized, it is possible that a positive isothermal temperature coefficient could exist at beginning of cycle (BOC). Safety analyses verify the acceptability of the isothermal temperature coefficient for limits specified in 3.1.F. Other conditions, e.g., higher power or partial rod insertion would cause the isothermal coefficient to have a more negative value. These analyses demonstrate that applicable criteria in the NRC Standard Review Plan (NUREG 75/087) are met.

Physics measurements and analyses are conducted during the reload startup test program to (1) verify that the plant will operate within safety analyses assumptions and (2) establish operational procedures to ensure safety analyses assumptions are met. The 3.1.F requirements are waived during low power PHYSICS TESTS to permit measurement of reactor temperature coefficient and other physics design parameters of interest. Special operating precautions will be taken during these PHYSICS TESTS. In addition, the strong negative Doppler coefficient (Reference 1) and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

References:

1. FSAR Figure 3.2.10

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 106 AND 99 TO

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

NORTHERN STATES POWER COMPANY

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

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By letter dated June 25, 1991, the Northern States Power Company (NSP or the licensee) requested amendments to the Technical Specifications (TS) appended to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2. The proposed amendments would revise TS Section 3.1.A.2.C and Table TS 4.1-2A and the associated Bases in response to Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)." Generic Letter 90-06 provided NRC staff guidance on TS changes that should be implemented to improve the reliability of the Pressurizer Power Operated Relief Valves (PORV) and the availability of the low temperature overpressure protection system.

Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," involves the evaluation of the reliability of power-operated relief valves (PORVs) and block valves and their safety significance in PWR plants. The generic letter discussed how PORVs are increasingly being relied on to perform safety-related functions and the corresponding need to improve the reliability of both PORVs and their associated block valves. Proposed staff positions and improvements to the plants' technical specifications were recommended to be implemented at all affected facilities. This issue is applicable to all Westinghouse, Babcox & Wilcox, and Combustion Engineering designed facilities with PORVs.

Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors," addresses concerns with the implementation of the requirements set forth in the resolution of Unresolved Safety Issue (USI) A-26, "Reactor Vessel Pressure Transient Protection (Overpressure Protection)." The generic letter discussed the continuing occurrence of overpressure events and the need to further restrict the allowed outage time for a low-temperature overpressure protection channel in operating MODEs 4, 5, and 6. This issue is only applicable to Westinghouse and Combustion Engineering facilities.

2.0 EVALUATION

The changes to the Prairie Island TS being proposed in reference to Generic Letter (GL) 90-06 are described below.

A. TS Section 3.1.A.2.c (1)

1. The PORV Specification for reactor coolant average temperature greater than or equal to 310° is being revised to delete the limitations on criticality and startup operation. These limitations on criticality and startup operation are mode change limitations. The modified Standard Technical Specifications (STS) provided in Attachment A-1 to GL 90-06 specified that the provisions of Specification 3.0.4 are not applicable to the PORV Specification.

   The pressurizer PORV Specification mode change restrictions were incorporated into the Prairie Island TS by License Amendments 91 and 84, approved October 27, 1989 which incorporated a large upgrade into the Prairie Island TS. They were added to the PORV Specification as part of an effort to standardize the LCOs throughout the Prairie Island TS. However, at that time NSP was not aware of the STS exception to the requirements of Specification 3.0.4 for pressurizer PORVs. These mode change restrictions were mistakenly incorporated into the Prairie Island PORV Specification as part of standardized LCO wording.

   The pressurizer PORV mode change restrictions will be eliminated from the Prairie Island TS because the STS state that the provisions of 3.0.4 are not applicable and because the restrictions were mistakenly incorporated by a previous license amendment.

2. The requirement for the PORVs to be operable whenever the reactor is critical is being deleted because it is redundant to the requirement that the PORVs be operable whenever reactor coolant average temperature is greater than 310°F. The requirement to be operable above 310°F is more restrictive and encompasses the reactor critical condition.

3. The required action with one or both PORVs inoperable is being clarified to differentiate between the actions to be taken if the PORV is inoperable due to excessive seat leakage or is inoperable for other reasons. These changes include new requirements on maintaining power to the block valves and new time limitations/shutdown requirements for the inoperability of PORVs for reasons other than excessive seat leakage. The current action statement does not address the cause of the inoperability or the status of the power supply to the block valves.

4. The required action with one or more PORV block valves inoperable is being revised to eliminate the option of closing the inoperable block valve and to include new actions, including new time limitations, shutdown requirements and restrictions on automatic PORV operation. This action would eliminate the option of continued operation with an inoperable PORV block valve.

B. Technical Specification Section 3.1.A.2.c.(2)

1. The requirements for the operability of the Prairie Island low temperature overpressure protection system are currently provided by the PORV Specification for reactor coolant system average temperature below 310°F (Specification 3.1.A.2.c.(2)). This specification is being revised to provide separate low temperature overpressure protection system specifications for two reactor coolant system low temperature ranges, below 200°F and between 200°and 310°F.

2. Expanded action statements are being incorporated for the inoperability of PORVs during either low temperature range. These expanded action statements include new time limitations for the inoperability of one or both PORVs and specific requirements for the size of the reactor coolant vent to be utilized if the PORVs are inoperable longer than the allowed out of service time. The proposed 3 square inch reactor coolant system vent opening is based on the 2.956 square inch cross sectional flow area of a pressurizer PORV.

C. Technical Specification Table TS.4.1-2A

1. A note is being added to the PORV block valve quarterly surveillance requirement (Item 6) which states that a block valve quarterly surveillance need not be performed if the valve has been closed in response to proposed action statements 3.1.A.2.c(1).(b).2 or 3.1.A.2.c.(1).(b).3.

2. A typographical error is being corrected in the frequency discussion of Item 5. The word "floodes" is being corrected to "flooded"

3. A typographical error is being corrected in the frequency discussion of Item 6. The word "Quaterly" is being corrected to "Quarterly."

4. Item 12, which is just a reference to a previously deleted item, is being deleted to provide room to incorporate the new text in Item 6.

D. Technical Specification Bases

1. Per the guidance provided in Attachment A-3 to Generic Letter 90-06, information is being incorporated into the PORV TS Bases to aid in the determination of the operability of the PORVs.

2. Information on the basis of the 3 square inch reactor coolant system vent opening is being incorporated into the Bases for the PORV TS.

3. Guidance on the manual operation of the PORVs is being incorporated to support proposed action statements 3.1.A.2.c.(1).(b).4 and 5.

4. Editorial changes are being made to the PORV TS Bases such that consistent terminology is utilized when discussing the low temperature overpressure protection system.

5. Information on the basis for the low temperature overpressure protection system PORV setpoint, and when it is required to be updated, is being incorporated into the Bases for the low temperature overpressure protection TS.

6. Per the guidance provided in Attachment B-2 to Generic Letter 90-06, information is being incorporated into the low temperature overpressure protection TS Bases to aid in the determination of the operability of the low temperature overpressure protection system.

7. Several pages of the Bases for TS Section 3.1 are being renumbered because of the additional text being incorporated by the changes described above.

Since the proposed modifications are consistent with the staff's position previously stated in the generic letter and any deviations are justified in the above analysis, the staff finds the proposed modifications to be acceptable.

3.0 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.

4.0 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. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (56 FR 43811). 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.

5.0 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: M. Gamberoni

Date: June 21, 1993