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

August 28, 1989

Docket Nos. 50-282
and 50-306

Mr. T. M. Parker, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

Dear Mr. Parker:

SUBJECT: AMENDMENT NOS. 89 AND 82 TO FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60: (TAC NOS. 71536 AND 71537)

The Commission has issued the enclosed Amendment No. 89 to Facility Operating License No. DPR-42 and Amendment No. 82 to the Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Units Nos. 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated January 12, 1989. The amendments change the TS by revising the following:

1. The effective limitation of existing reactor coolant system heatup and cooldown curves are revised from 15 to 20 effective full power years.

2. The word "vessel" appearing in TS.3.1.B3 is replaced with the phrase "steam generator shell."

3. The description of the steam generator pressure/temperature limitation and the pressurizer temperature limits was added to the bases section (page TS.3.1-8). The descriptions were inadvertently omitted during the previous revision to the heatup and cooldown curves (i.e., Amendment Nos. 80 and 73 dated November 14, 1986).

4. Modified TS 3.1.G to include requirements involving (1) the safety injection pumps are to be in pullout condition when reactor coolant system (RC) temperature is below 200°F, (2) the safety injection pump may be used for cooling and RCS inventory control when the Residual Heat Removal system is inoperable, and (3) specify a numerical valve of 310°F for Minimum Pressurization Temperature (MRT) appearing in TS.3.1.G.

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A copy of our related Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Dominic C. DiIanni

Dominic C. DiIanni, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V & Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 89 to License No. DPR-42
2. Amendment No. 82 to License No. DPR-60
3. Safety Evaluation
4. Notice

cc w/enclosures:
See next page

Mr. T. M. Parker
Northern States Power Company

Prairie Island Nuclear Generating
Plant

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Red Wing, Minnesota 55066

P137

A copy of our related Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Dominic C. DiIanni

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OGC	BGrimes
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DOCUMENT NAME: PRAIRIE ISLAND AMENDMENT

PM for	DCD	DCD/LY	Subject to prior publication of EA
LA/PD31:DRSP	PM/PD31:DRSP	(A)D/PD31:DRSP	OGC
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Sincerely,

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 89
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 January 12, 1989, 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. 89, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Dominic C. Di Ianni /L.Y.

Lawrence A. Yandell, Acting Director
Project Directorate III-1
Division of Reactor Projects - III,
IV, V & Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 28, 1989

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.82
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 January 12, 1989, 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. 82, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Dominic C. Di Ianni / L.Y.

Lawrence A. Yandell, Acting Director
Project Directorate III-1
Division of Reactor Projects - III,
IV, V & Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: August 28, 1989

ATTACHMENT TO LICENSE AMENDMENTS NOS. 82 AND 89

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

DOCKETS NOS. 50-282 AND 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 marginal lines indicating the area of change.

REMOVE	INSERT
TS.3.1-6	TS.3.1-6
TS.3.1-7	TS.3.1-7
TS.3.1-8	TS.3.1-8
TS.3.1-19	TS.3.1-19
TS.3.1-20	TS.3.1-20
Figure TS.3.1-1	Figure TS.3.1-1
Figure TS.3.1-2	Figure TS.3.1-2
TS.3.3-1	TS.3.3-1

B. HEATUP AND COOLDOWN

Specification:

1. The Unit 1 and Unit 2 reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures TS.3.1-1 and TS.3.1-2.

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

   b. Figures TS.3.1-1 and TS.3.1-2 define limits to assure prevention of non-ductile failure only. For normal operation other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. The limit lines shown in Figures TS.3.1-1, TS.3.1-2 shall be recalculated periodically using methods discussed in the Bases section.

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator shell is below 70°F.

4. The pressurizer heatup rate shall not exceed 100°F/hr and the pressurizer cooldown rate shall not exceed 200°F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

Bases

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 will 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.

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.

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.

Reference

1. USAR Section 4.2

G. Minimum Conditions for RCS Temperature Less Than 310°F*

Specification

1. Both pressurizer power operated relief valves (PORV's) shall be operable whenever the RCS temperature is less than the minimum pressurization temperature (310°F*), except one PORV may be inoperable for seven days. If these conditions are not met, the reactor coolant system must be depressurized and vented to the atmosphere or to the pressurizer relief tank within eight hours.

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

3. A reactor coolant pump may be started at RCS temperatures less than 310°F* only if either of the following conditions is met -

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

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

4. At least one safety injection pump control switch in the control room shall be in pullout whenever RCS temperature is less than 310°F*, except for conditions satisfying Specification 3.1.G.6.

5. Both safety injection pump control switches in the Control Room shall be in pullout whenever RCS temperature is less than 200°F (except as specified in 3.1.G.6 and 3.1.G.7)

6. At RCS temperatures less than 310°F*, both SI pumps may be run for conduct of the integrated SI test only if either of the following conditions is met -

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

   (b) The reactor vessel head is removed.

7. With RCS temperature less than 200°F, 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 during reduced inventory conditions.

*Valid until 20 EFPY

BASIS

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 mitigating 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[1], 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 system is designed to perform its function in the event of a single failure and is designed to meet the requirements of IEEE-279. The backup air supply provides sufficient air to operate the PORV's following a letdown isolation with one charging pump in operation for a period of ten minutes after receipt of the overpressure alarm. These specifications provide assurance that the overpressure mitigating system will perform its intended function.

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

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

Maintaining both safety injection pump Control Room control switches in pullout, as specified in 3.1.G.5, will ensure that the RCS pressure/temperature limitations specified in Figures TS.3.1-1 and TS.3.1-2 will not be exceeded, at low RCS temperatures, as the result of mass input into the RCS from an inadvertent safety injection pump start.

Specification 3.1.G.6 allows use of both SI pumps at low temperatures for conduct of the integrated SI test. In this case, pressurizer level is maintained at less than 50% and the SI pump discharge valves are shut to prevent fluid injection into the RCS. This combination of conditions under strict administrative control assure that overpressurization cannot occur. The option of having the reactor vessel head removed is allowed since in this case RCS overpressurization cannot occur.

Specification 3.1.G.7 allows the use of one safety injection pump to ensure that adequate core cooling and reactor coolant system inventory can be maintained in the event of a loss of Residual Heat Removal System cooling during reduced inventory conditions. A reduced inventory condition, as defined by Generic Letter 88-17, Loss of Decay Heat Removal, exists whenever the reactor vessel water level is lower than three feet below the reactor vessel flange. The operation of a safety injection pump under such conditions would be controlled by an approved emergency operating procedure.

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

*Valid until 20 EFPY

FIGURE TS 3.1-1

UNIT 1 and UNIT 2
REACTOR COOLANT SYSTEM HEAT UP LIMITATIONS
(Applicable for First 20 EFPY of Operation)

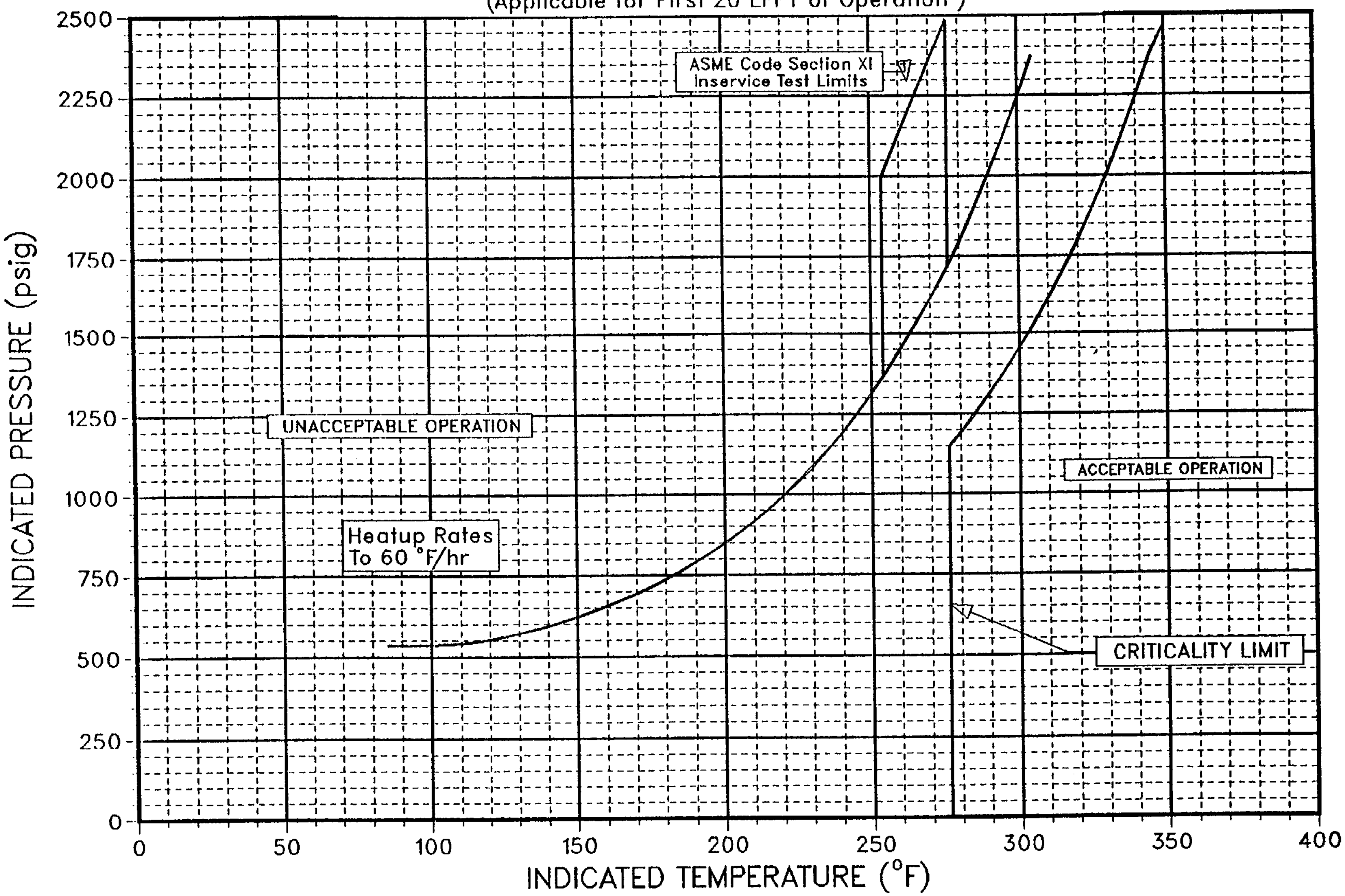
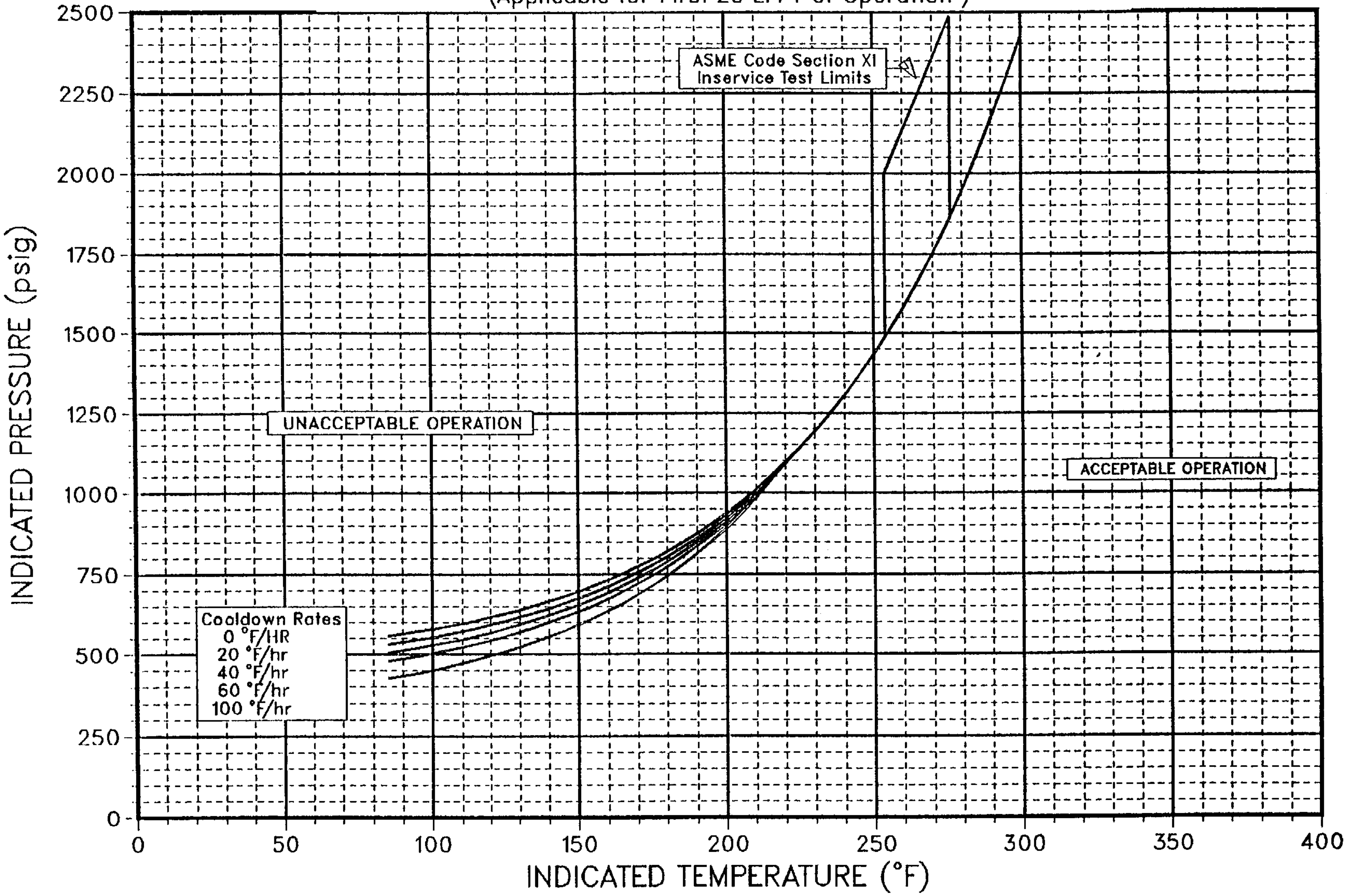


FIGURE TS 3.1-2

UNIT 1 and UNIT 2
REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS
(Applicable for First 20 EFPY of Operation)



Prairie Island Unit 1 - Amendment No. 82, 80, 89
Prairie Island Unit 2 - Amendment No. 76, 73, 82

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 it be heated or maintained above 200°F unless thc following conditions are satisfied except as permitted in Specification 3.3.A.2.

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

   b. Each reactor coolant system accumulator shall be operable 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 at least 700 psig

   c. Two safety injection pumps are operable except that pump control switches in the control room shall meet the requirements of Section 3.1.G whenever the reactor coolant system temperature is less than 310°F*.

   d. Two residual heat removal pumps are operable.

   e. Two residual heat exchangers are operable.

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

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

*Valid until 20 EFPY

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 89 AND 82 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

In response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," the Northern States Power Company (the licensee) requested to revise the pressure/temperature (P/T) limits in the Prairie Island Nuclear Generating Plant (PINGP) Unit Nos. 1 and 2 Technical Specification Section 3.1. The request was documented in a letter from the licensee dated January 12, 1989. The purpose of the revision is to change the effectiveness of the P/T limits from 15 to 20 effective full power years (EFPY). The licensee proposed to use one set of P/T limits for both units. The proposed P/T limits were developed based on the data from actual surveillance capsules. The proposed revision provides up-to-date P/T limits for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H to 10 CFR Part 50; the ASTM Standards and ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); Regulatory Guide 1.99, Revision 2; Standard Review Plan (SRP) Section 5.3.2 and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the Technical Specifications. The P/T limits are among the limiting conditions of operation in the Technical Specifications for all commercial nuclear plants in the U.S. Appendices G and H to 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance. These must be considered in setting P/T limits. An acceptable method in constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G to 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, to test the beltline materials in the surveillance capsules in accordance with Appendix H to 10 CFR Part 50. Appendix H, in turn, refers to the ASTM Standards. These tests define the condition of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees and permittees use the methods in Regulatory Guide (RG) 1.99, Revision 2 to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of the unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and margin to account for uncertainties in the prediction method.

Appendix H to 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, requires that the capsules be installed in the vessel before startup and that they contain test specimens that are made from plate, weld and heat-affected-zone materials of the reactor beltline.

2.0 EVALUATION

We have evaluated the effect of neutron irradiation embrittlement on each beltline material in PINGP-1 and PINGP-2 reactor vessels. The amount of neutron irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. We have determined that the material with the highest ART (most embrittled) at 20 EFPY for both units was the circumferential weld between the intermediate and lower shells in Unit 2.

The licensee has removed three surveillance capsules from PINGP-1 and three capsules from PINGP-2. The results from capsules V, P, and R in Unit 1 were published in Westinghouse Reports WCAP-8916, WCAP-10102, and WCAP-11006, respectively. The results from capsules V, T, and R in Unit 2 were published in Westinghouse Reports WCAP-9212, WCAP-9877 and WCAP-11343, respectively. All surveillance capsules contained Charpy impact specimens and tensile specimens which were made from base metal, weld metal, and heat affected zone (HAZ) metal.

For the limiting beltline material, the weld between the intermediate and lower shells in Unit 2, we have calculated the amount of neutron irradiation embrittlement in accordance with RG 1.99, Rev.2. The ART at 20 EFPY at ¼T was calculated to be 126°F. The ART was determined by the least squares extrapolation method of the Unit 2 surveillance data. The least squares method is described in Section 2.1 of RG 1.99, Rev. 2.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 144.8°F for the limiting weld metal in the beltline of Unit 2. We performed a similar calculation and verified the licensee's ART value to be conservative (see Table 1). Substituting the ART of 144.8°F into equations in SRP 5.3.2, we verified that the proposed P/T limits for heatup, cooldown, criticality, and hydrotest meet the beltline material requirements in Appendix G to 10 CFR Part 50.

In addition to beltline materials, Appendix G to 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions that are 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. Based on the flange reference temperature of 4°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires the predicted Charpy USE at end of life to be above 50 ft-lb. Based on data from a surveillance capsule withdrawn at 8.56 EFPY, the measured Charpy USE is 75 ft-lb for the intermediate to lower shell weld metal. This is a 4.5% reduction from the unirradiated value of 78.5 ft-lb. Using the method in RG 1.99, Revision 2, the predicted Charpy USE of the weld metal at end of life will be below 50 ft-lb. The staff will monitor the weld metal Charpy USE from future surveillance capsules. The surveillance capsule data will provide early warning of the decrease in Charpy USE, because the surveillance capsule lead factors are greater than 1.0. Furthermore, since capsule R was withdrawn after 8.56 EFPY and had a lead factor of 2.93, the Charpy USE will be greater than 75 ft-lb for about 25 years of reactor operation.

Based on the above evaluation, the staff concludes that the proposed P/T limits on the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 20 EFPY, because the limits conform to requirements of Appendices G and H to 10 CFR Part 50. The licensee's submittal also satisfies Generic Letter 88-11, because the licensee used the method in RG 1.99, Rev. 2, to calculate the ART. Hence, the proposed P/T limits that would be incorporated into the PINGP-1 and PINGP-2 Technical Specifications are acceptable.

The licensee has also proposed several miscellaneous administrative changes that serve to clarify the TS requirements. The evaluation of these proposed changes are as follows:

1. The TS 3.1.B.3 deals with the pressurization requirements of the steam generator when the temperature is below 70°F. The licensee proposes the replace word "vessel" with the phrase "steam generator shell." This change serves to clarify the requirement and does not affect the intent of the requirement. The staff finds the proposed change acceptable.

2. The description of the steam generator pressure/temperature limitation and the pressurizer temperature limits are being reinserted in the bases section. The descriptions were inadvertantly omitted during the previous revision to the heatup and cooldown curve (i.e., Amendments Nos. 80 and 73 dated November 14, 1986). This is considered an editorial change having no affect on the heatup and cooldown requirement and therefore is acceptable.

3. The licensee is proposing several changes to TS.3.1.G which serve to enhance the requirements concerning the RCS temperature below the minimum pressuirzation temperature (MPT). The proposed changes consist of specifying the temperature value (310°F) for MPT.

Other proposed changes involve actions to assure that the safety injection pumps are not operated at RCS temperature below 200°F and would be operated for maintaining cooling capability and inventory control only when residual heat removal system is inoperable. These proposed changes are considered administrative in nature, clarifying the requirement and therefore are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32 and 51.35, an environmental assessment and finding of no significant impact was published in the Federal Register on August 28, 1989 (54 FR 35542). Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

4.0 CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: B. Elliot
D. DIanni

Dated: August 28, 1989

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

DOCKET NOS. 50-282 AND 50-306

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment Nos. 89 and 82 Facility to Operating License Nos. DPR-42 and DPR-60 issued to Northern States Power Company (licensee) which revised the Technical Specifications for operation of the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2 located in Goodhue County, Minnesota.

The amendment is effective as of the date of issuance.

The amendment revised the Technical Specifications by modifying the pressure-temperature limits taking into consideration the irradiation effects of the embrittlement of the reactor vessel material up to 20 effective full power years. Other modifications are administrative in nature serving to clarify the existing requirements.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on June 8, 1989 (54 FR 24609). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of this amendment will not have a significant effect on the quality of the human environment.

For further details with respect to the action see (1) the application for amendment dated January 12, 1989, (2) Amendment Nos. 89 and 82 to Facility Operating License Nos. DPR-42 and DPR-60, (3) the Commission's related Safety Evaluation and (4) the Commission's Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street NW, Washington, D.C., and at the Technology and Science Department, Minneapolis Public Library, 200 Nicollet Mall, Minneapolis, Minnesota 55401. A copy of items (2), (3), and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Projects III, IV, V and Special Projects.

Dated at Rockville, Maryland this August 28, 1989

FOR THE NUCLEAR REGULATORY COMMISSION

Dominic C. DiIanni

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