May 21, 1996

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

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: PRESSURIZER SAFETY VALVES AND MAIN STEAM SAFETY VALVES LIFT SETTING TOLERANCE CHANGE AND SAFETY LIMIT CURVE CHANGES (TAC NOS. M92349 AND M92350)

Dear Mr. Anderson:

The Commission has issued the enclosed Amendment No. 123 to Facility Operating License No. DPR-42 and Amendment No. 116 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 May 4, 1995, as supplemented by letters dated November 27, 1995, and March 1, 1996.

The amendments revise the Prairie Island TS by changing the pressurizer and main steam safety valve lift setting tolerance from ± 1 percent to ± 3 percent (as found only), revising the safety limit curves, reformatting Section 2, and correcting typographical errors.

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

Docket Nos. 50-282 and 50-306

Enclosures:
1. Amendment No. 123 to DPR-42
2. Amendment No. 116 to DPR-60
3. Safety Evaluation

cc w/encls: See next page

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DATED: May 21, 1996

AMENDMENT NO. 123 TO FACILITY OPERATING LICENSE NO. DPR-42-PRAIRIE ISLAND UNIT 1
AMENDMENT NO. 116 TO FACILITY OPERATING LICENSE NO. DPR-60-PRAIRIE ISLAND UNIT 2

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

May 21, 1996

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

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: PRESSURIZER SAFETY VALVES AND MAIN STEAM SAFETY VALVES LIFT SETTING TOLERANCE CHANGE AND SAFETY LIMIT CURVE CHANGES (TAC NOS. M92349 AND M92350)

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Sincerely,

Janet Kennedy for

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

Docket Nos. 50-282 and 50-306

Enclosures:
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3. Safety Evaluation

cc w/encls: See next page

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 123
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 May 4, 1995, as supplemented November 27, 1995, and March 1, 1996, 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. 123, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance, with full implementation within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Janet Kennedy for

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 21, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 123

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-i	TS-i
TS-viii	TS-viii
TS-x	TS-x
TS-xiii	TS-xiii
TS.2.1-1	TS.2.1-1
TS.2.2-1	--
Figure TS.2.1-1	Figure TS.2.1-1
TS.2.3-2	TS.2.3-2
TS.2.3-3	TS.2.3-3
TS.3.4-1	TS.3.4-1
Table TS.4.1-2A	Table TS.4.1-2A (Page 1 of 2)
--	Table TS.4.1-2A (Page 2 of 2)
TS.6.4-1	--
B.2.1-1	B.2.1-1
B.2.1-2	B.2.1-2
--	B.2.1-3
--	B.2.1-4
--	B.2.1-5
--	Figure B.2.1-1
B.2.2-1	B.2.2-1
B.3.1-2	B.3.1-2
B.3.1-3	B.3.1-3
B.3.4-1	B.3.4-1
--	B.3.4-2

TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

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APPENDIX A TECHNICAL SPECIFICATIONS

LIST OF FIGURES

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTING

2.1 SAFETY LIMITS

A. Reactor Core Safety Limits

In MODES 1 and 2, combination of thermal power (measured in ΔT), pressurizer pressure, and the highest reactor coolant system loop average temperature shall not exceed the limits shown in Figure TS.2.1-1.

B. Reactor Coolant System Pressure Safety Limit

In MODES 1, 2, 3, 4, and 5, the reactor coolant system pressure shall not exceed 2735 psig.

2.2 SAFETY LIMIT VIOLATIONS

A. If SAFETY LIMIT 2.1.A. is violated, restore compliance and be in MODE 3 within 1 hour.

B. If SAFETY LIMIT 2.1.B. is violated:

1. In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

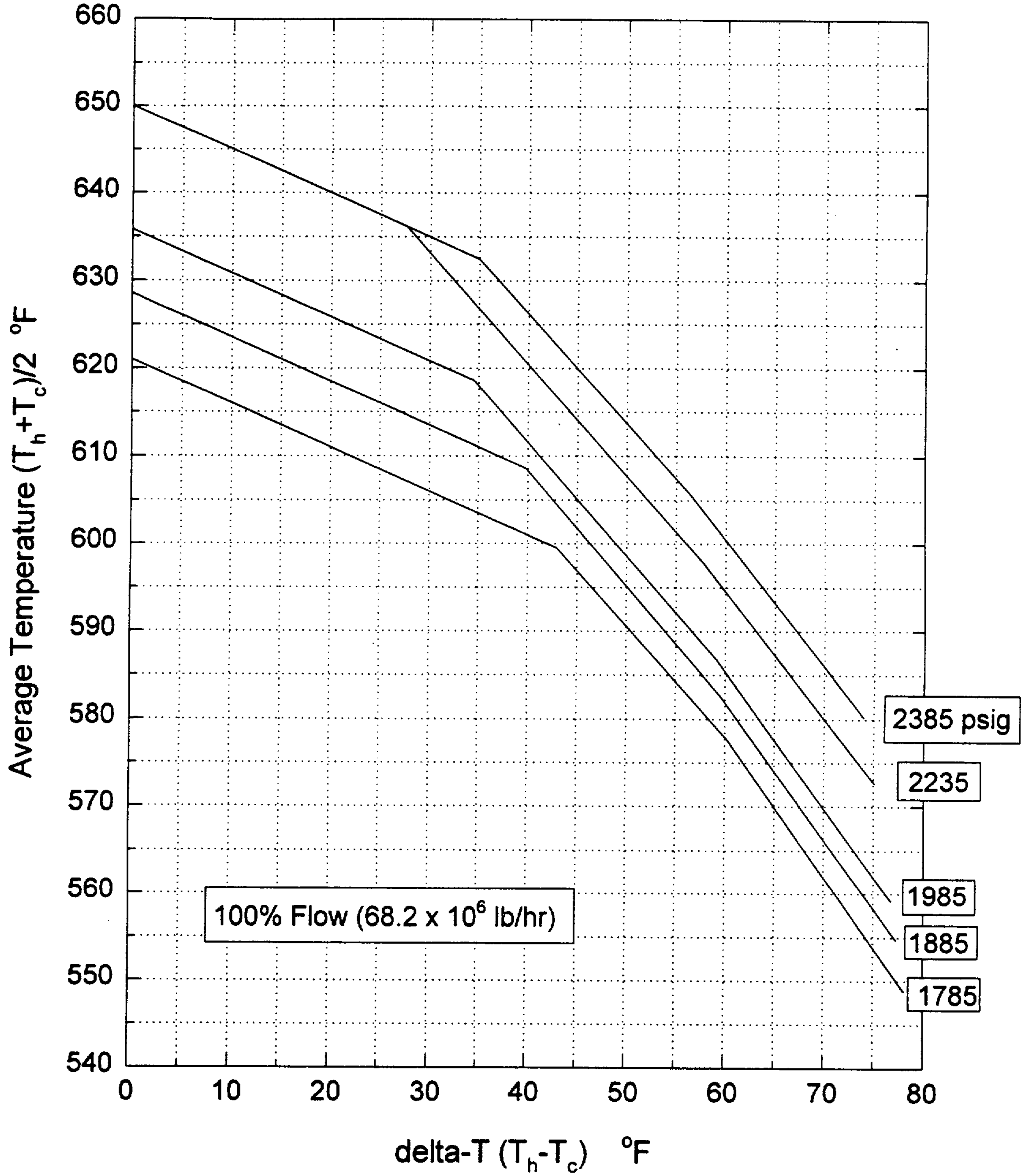
2. In MODE 3, 4, or 5, restore compliance within 5 minutes.

C. If a SAFETY LIMIT is violated, within 1 hour notify the NRC Operations Center in accordance with 10CFR50.72.

D. If a SAFETY LIMIT is violated, within 24 hours notify the Vice President Nuclear Generation, and the Chairman of the Safety Audit Committee or their designated alternates.

E. If a SAFETY LIMIT is violated, within 30 days a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC, the Vice President Nuclear Generation and the Safety Audit Committee.

F. If a SAFETY LIMIT is violated, operation of the unit shall not be resumed until authorized by the NRC.



Reactor Core Safety Limits

Figure TS.2.1-1

2.3.A.2.d Cont.

and f (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chamber, with gains to be selected based on measured instrument response during plant startup tests, such that where q_t and q_b are the percent power in the top and bottom halves of the core, respectively, and $q_t + q_b$ is total core power in percent of rated power:

1. for $q_t - q_b$ within -12% and +9%, f (ΔI) = 0, and

2. for each percent that the magnitude of $q_t - q_b$ exceeds +9% the ΔT trip set point shall be automatically reduced by an equivalent of 2.5 percent of RATED THERMAL POWER.

3. for each percent that the magnitude of $q_t - q_b$ exceeds -12%, the T trip set point shall be automatically reduced by an equivalent of 1.5 percent of RATED THERMAL POWER.

e. Overpower Δ T

$$\Delta T_p \leq \Delta T_o \left[K_4 - \frac{K_5 t_3 s T}{1 + t_3 s} - K_6 (T - T') - f(\Delta I)\right]$$

where

ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
T	=	Average temperature, °F
T'	=	567.3°F
K_4	≤	1.10
K_5	=	0.0275 for increasing T; 0 for decreasing T
K_6	=	0.002 for T > T', 0 for T < T'
t_3	=	10 sec
f(ΔI)	=	as defined in d. above

f. Low reactor coolant flow per loop - ≥90% of normal indicated loop flow as measured at loop elbow tap.

2.3.A.2.g. Reactor coolant pump bus undervoltage - ≥75% of normal voltage.

h. Open reactor coolant pump motor breaker.

Reactor coolant pump bus underfrequency - ≥58.2 Hz

i. Power range neutron flux rate.

1. Positive rate - ≤15% of RATED THERMAL POWER with a time constant ≥2 seconds

2. Negative rate - ≤7% of RATED THERMAL POWER with a time constant of ≥2 seconds

3. Other reactor trips

a. High pressurizer water level - ≤90% of narrow range instrument span.

b. Low-low steam generator water level - ≥5% of narrow range instrument span.

c. Turbine Generator trip

1. Turbine stop valve indicators - closed

2. Low auto stop oil pressure - ≥45 psig

d. Safety injection - See Specification 3.5

3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the operating status of the steam and power conversion system.

Objective

To specify minimum conditions of steam-relieving capacity and auxiliary feed- water supply necessary to assure the capability of removing decay heat from the reactor, and to limit the concentration of activity that might be released by steam relief to the atmosphere.

Specification

A. Steam Generator Safety and Power Operated Relief Valves

1. A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 350°F unless the following conditions are satisfied (except as specified in 3.4.A.2 below):

   a. Ten steam generator safety valves shall be OPERABLE with lift settings of 1077, 1093, 1110, 1120 and 1131 psig ± 3% except during testing.

   b. Both steam generator power-operated relief valves for that reactor are OPERABLE.

2. During STARTUP OPERATION or POWER OPERATION, the following condition of inoperability may exist provided STARTUP OPERATION is discontinued until OPERABILITY is restored. If OPERABILITY is not restored within the time specified, be in at least HOT SHUTDOWN within the next 6 hours and reduce reactor coolant system average temperature below 350°F within the following 6 hours.

   a. One steam generator power-operated relief valve may be inoperable for 48 hours.

B. Auxiliary Feedwater System

1. A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 350°F unless the following conditions are satisfied (except as specified in 3.4.B.2 below):

   a. For single unit operation, the turbine-driven pump associated with that reactor plus one motor-driven pump are OPERABLE.

   b. For two-unit operation, all four auxiliary feedwater pumps are OPERABLE.

   c. Valves and piping associated with the above components are OPERABLE except that during STARTUP OPERATION necessary changes may be made in motor-operated valve position. All such changes shall be under direct administrative control.

Table TS.4.1-2A (Page 1 of 2)

MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

Equipment	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 Quarter	7
3. Pressurizer Safety Valves	Verify OPERABLE in accordance with the Inservice Testing Program (± 3%). Following testing, lift settings shall be within ±1%	Per ASME Code, Section XI Inservice Testing Program	-
4. Main Steam Safety Valves	Verify each required lift setpoint in accordance with the Inservice Testing Program (± 3%). Following testing, lift settings shall be within ±1%	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	-

Table TS.4.1-2A (Page 2 of 2)

MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

Equipment	Test	Frequency	FSAR Sect. Reference
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	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 missle ejection incident of 1.0×10^{-5} per year. In no case shall the turbine valve test interval exceed one year.	10

2.1 SAFETY LIMITS

A. Reactor Core Safety Limits

Bases

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating the hot regions of the core within the nucleate boiling regime of heat transfer wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters; thermal power, reactor coolant temperature and pressure have been related to DNB through the W-3 and WRB-1 DNB correlations. The W-3 DNB correlation is used for Exxon fuel. The WRB-1 DNB correlation is used for Westinghouse fuel. The W-3 and WRB-1 DNB correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNB ratio, DNBR, during steady state operation, normal operational transients, and anticipated transients is limited to 1.30 for the Exxon fuel using the W-3 correlation and to 1.17 for the Westinghouse fuel using the WRB-1 correlation. There is a third DNBR limit specifically for the steam line break accident but it does not apply to the safety limit curve calculations. These limits correspond to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The safety limit curves of Figure TS.2.1-1 define the regions of acceptable operation with respect to average temperatures, power, and pressurizer pressure. These boundaries of acceptable operations are limited by the thermal overpower limit (fuel melting), thermal overtemperature limit (cladding damage based on DNB considerations), and the locus of points where the steam generation safety valves open. These limits are used to set the overpower and overtemperature ΔT trip setpoints.

The safety limit curves of Figure TS.2.1-1 comprise the most limiting of the following four criteria:

1) Vessel Exit Temperature < 650°F

This is the design temperature limit. This limit defines the portion of the safety limit curves from 0 ΔT to the first knee for the 2235 and 2385 psig curves. At these pressures, the temperature limit of 650° is more

A. Reactor Core Safety Limits

Bases continued

limiting than the T_{sat} limit. The locus of points is calculated from a heat balance with the minimum RCS flow specified in TS 3.10.J.

2) Vessel Exit Temperature < T_{sat}

This limit ensures that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated water which ensures that the ΔT measured by instrumentation used by the RPS as a measure of core thermal power is proportional to core power. This limit defines the portion of the safety limit curves from 0 ΔT to the first knee for the 1985, 1885 and 1785 psig curves. The locus of points is calculated from a heat balance with the minimum RCS flow specified in TS 3.10.J.

3) Minimum DNBR > 1.3 or 1.17 whichever is applicable

As mentioned before, 1.3 is the DNBR limit for Exxon fuel using the W-3 critical heat flux correlation and 1.17 is the DNBR limit for Westinghouse fuel using the WRB-1 critical heat flux correlation. The locus of points past the first knee at all pressures represents the thermal-hydraulic conditions above which the hot channel has a DNBR less than the limit. The conditions are evaluated using approved DNB methodology. The assumptions used in the calculation include a bypass flow of 6%, an $F_{\Delta H}$ greater than 1.75, and a rod bow penalty of 2.6%. The very shallow knee at full power ΔT occurs because the $F_{\Delta H}$ (hot channel power) is allowed to increase for core power less than RATED THERMAL POWER as described in TS 3.10.B.1.

4) Hot Channel Exit Quality < 15% or 30% whichever is applicable

This limit is typically not the most restrictive because it is generally approached at lower powers where the T_{exit} < T_{sat} or 650°F is more limiting. However, it is considered when the DNB calculations described above are performed using approved DNB methodology. This limit is determined by the range of the channel exit quality for the critical heat flux correlations. The maximum channel exit quality limit is 15% for the W-3 correlation and 30% for the WRB-1 correlation.

Operation above the safety limit curves of Figure TS 2.1-1 is not acceptable. At each pressure the safety limit curve is the most restrictive combination of the four limits discussed above. The area of acceptable operation below the safety limit curves is bounded by the OTΔT trip, the OPΔT trip, and the locus of points where the steam generator (main steam) safety valves open. The ΔT trips are set conservatively with respect to the safety limit curves to protect the core from exceeding the safety limits. The locus of points at which the steam generator safety valves open defines the thermodynamic limit of temperature conditions in the RCS based on the maximum pressure in the steam generators. For this calculation, it is assumed that the pressure in the steam generator is 1195 psig which is 110% of design pressure. It is

A. Reactor Core Safety Limits

Bases continued

required that the steam generator safety valves protect the pressure from exceeding 110% of design pressure so using 1195 psig in the calculations is conservative. Thus, the reactor is protected from violating the safety limits by the physical limit of the ΔT trips and the opening of the steam generator safety valves.

As an example, all the limits for the 2235 psig curve are plotted in Figure B.2.1-1 along with the ΔT trips and the locus of points where the steam generator safety valves open. This plot demonstrates that the ΔT trips and the steam generator safety valves do protect the reactor from exceeding the safety limits. Note, however, that the OTΔT trip locus on that plot is for steady state conditions and that the locus will drop in response to the rate at which the ΔT is increasing. In addition, f(ΔI) increasing will also lower the OTΔT trip locus.

The safety limit curves are plotted with ΔT on the x-axis for the following two reasons: 1.) the full power ΔT is different at different temperatures and pressures because water properties are nonlinear. This makes it difficult to plot the curves at each pressure using the same scale for the percent power axis. 2.) the ΔT trip setpoints which the reactor protection system actually calculates is based on the ΔT, not the percent power.

Except for special tests, POWER OPERATION with only one loop or with natural circulation is not allowed. Safety limits for such special tests will be determined as a part of the test procedure.

The curves are conservative for the following nuclear hot channel factors:

$$F^{N}_{\Delta H} = F^{RTP}_{\Delta H} \,[1 + PFDH(1-P)] \;\; ; \text{ and } F^{N}_{Q} = F^{RTP}_{Q}$$

where:

- F^{RTP}_{Q} is the F_Q limit at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT.

- $F^{RTP}_{\Delta H}$ is the $F_{\Delta H}$ limit at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT.

- PFDH is the Power Factor Multiplier for $F^{N}_{\Delta H}$ specified in the CORE OPERATING LIMITS REPORT

Use of these factors results in more conservative safety limits than would result from power distribution limits in Specification TS.3.10.

A. Reactor Core Safety Limits

Bases continued

This combination of hot channel factors is higher than that calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are covered by Specification 3.10. Adverse power distribution factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits specified in the CORE OPERATING LIMITS REPORT assure that the DNB ratio is always greater at part power than at full power.

The Reactor Control and Protective System is designed to prevent any anticipated combination of transient conditions that would result in a DNB ratio of less than 1.30 for Exxon Nuclear fuel and less than 1.17 for Westinghouse fuel.

B. Reactor Coolant System Pressure Safety Limit

Bases

The reactor coolant system (Reference 1) serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure the reactor coolant system is the primary barrier against the release of fission products. By establishing a system pressure limit, the continued integrity of the reactor coolant system is assured. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME Code, Section III is 110% of design pressure.

The maximum transient pressure allowable in the reactor coolant system piping, valves and fittings under USAS Section B31.1 is 120% of design pressure. Thus, the safety limit of 2735 psig (110% of design pressure) has been established (Reference 2).

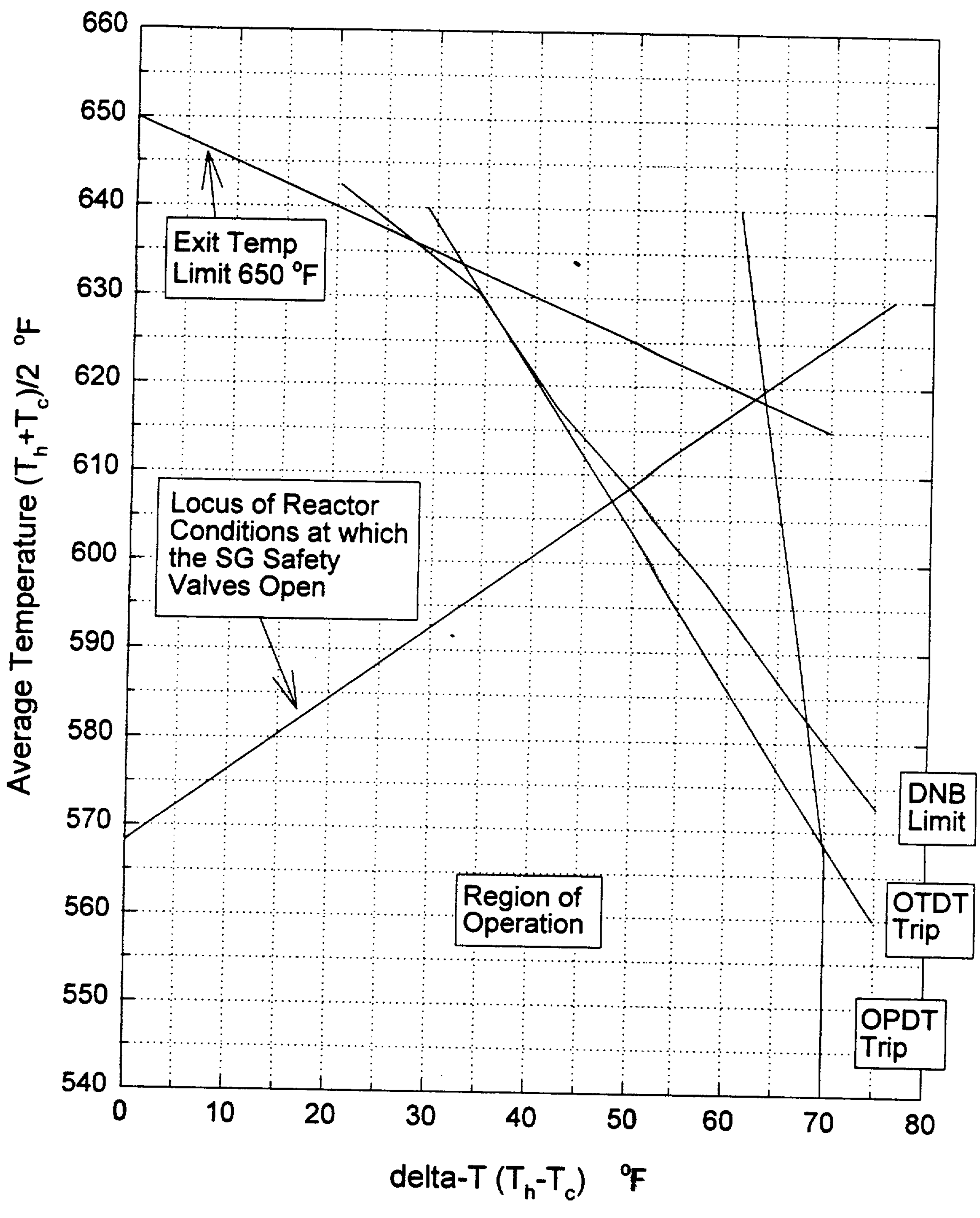
The nominal settings of the power-operated relief valves, the reactor high pressure trip and the safety valves have been established to assure that the pressure never reaches the reactor coolant system pressure safety limit.

In addition, the reactor coolant system safety valves (Reference 3) are sized to prevent system pressure from exceeding the design pressure by more than 10 percent (2735 psig) in accordance with Section III of the ASME Boiler and Pressure Vessel Code, assuming complete loss of load without a direct reactor trip or any other control, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valves settings.

As an assurance of system integrity, the reactor coolant system was hydrotested at 3107 psig prior to initial operation (Reference 4).

References

1. USAR, Section 4.1
2. USAR, Section 4.1.3.1
3. USAR, Section 4.4.3.2
4. USAR, Section 4.1



Origin of Safety Limit Curves at 2235 psig
with delta-T Trips
and Locus of Reactor Conditions at
which the SG Safety Valves Open

Figure B.2.1-1

2.2 SAFETY LIMIT VIOLATIONS

Bases

If the reactor core SAFETY LIMIT 2.1.A is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SAFETY LIMIT is not applicable.

The allowed completion time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SAFETY LIMIT is not applicable, and reduces the probability of fuel damage.

If the Reactor Coolant System pressure SAFETY LIMIT 2.1.B is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the Reactor Coolant System pressure SAFETY LIMIT may cause immediate Reactor Coolant System failure and create a potential for radioactive releases in excess of 10CFR100, "Reactor Site Criteria", limits.

The allowable completion time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

If the Reactor Coolant System pressure SAFETY LIMIT 2.1.B is exceeded in MODE 3, 4, or 5, Reactor Coolant System pressure must be restored to within the SAFETY LIMIT value within 5 minutes. Exceeding the Reactor Coolant System pressure SAFETY LIMIT in MODE 3, 4, or 5 is more severe than exceeding this SAFETY LIMIT in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SAFETY LIMIT within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

If either SAFETY LIMIT in 2.1.A or 2.1.B is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10CFR50.72.

If either SAFETY LIMIT in 2.1.A or 2.1.B is violated, the Vice President Nuclear Generation shall be notified within 24 hours. The 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to senior management.

If either SAFETY LIMIT in 2.1.A or 2.1.B is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC and the Vice President Nuclear Generation. This requirement is in accordance with 10CFR50.73.

If either SAFETY LIMIT in 2.1.A or 2.1.B is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

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. These valves are considered OPERABLE at ± 3% of their setpoint of 2485 psig. Following testing the valve lift settings are restored within a nominal ± 1% of their setpoint. 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.

3.1 REACTOR COOLANT SYSTEM

Bases continued

A. Operational Components (continued)

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

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

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

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

3.4 STEAM AND POWER CONVERSION SYSTEMS

Bases

A reactor shutdown from power requires removal of decay heat. Decay heat removal requirements are normally satisfied by the steam bypass to the condenser and by continued feedwater flow to the steam generators. Normal feedwater flow to the steam generators is provided by operation of the turbine-cycle feedwater system.

The ten steam generator safety valves have a total combined rated capability of 7,745,000 lbs/hr. The total full power steam flow is 7,094,000 lbs/hr; therefore, the ten steam generator safety valves will be able to relieve the total steam flow if necessary (Reference 1).These valves are considered OPERABLE at ± 3% of their specified setpoint. Following testing the valve lift settings are restored within a nominal ± 1% of their setpoint.

In the unlikely event of complete loss of offsite electrical power to either or both reactors, continued removal of decay heat would be assured by availability of either the steam-driven auxiliary feedwater pump or the motor-driven auxiliary feedwater pump associated with each reactor, and by steam discharge to the atmosphere through the steam generator safety valves. One auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from one reactor. The motor-driven auxiliary feedwater pump for each reactor can be made available to the other reactor. During STARTUP OPERATIONS, the Auxiliary Feedwater motor-operated injection valves maybe less than full open as necessary to faciliate plant startup.

The minimum amount of water specified for the condensate storage tanks is sufficient to remove the decay heat generated by one reactor in the first 24 hours of shutdown. Essentially unlimited replenishment of the condensate storage supply is available from the intake structures through the cooling water system.

The two steam generator power-operated relief valves located upstream of the main steam isolation valves are required to remove decay heat and cool the reactor down following a steam generator tube rupture event (Reference 3) and following a high energy line rupture outside containment (Reference 2). The steam generator power operated relief valves are provided with manual upstream block valves to permit testing at power and to provide a means of isolation.

In order to assure timely response to a steam generator tube rupture event, a steam generator power operated relief valve is considered operable when it is capable of being remotely operated and when its associated block valve is open.

Isolation dampers are required in ventilation ducts that penetrate those rooms containing equipment needed for a high energy line rupture outside containment.

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 gpm primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

References

1. USAR, Section 11.9.4
2. USAR, Appendix I
3. USAR, Section 14.5.4

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. 116
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 May 4, 1995, as supplemented November 27, 1995, and March 1, 1996, 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. 116, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance, with full implementation within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Janet Kennedy for

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 21, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 116

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-i	TS-i
TS-viii	TS-viii
TS-x	TS-x
TS-xiii	TS-xiii
TS.2.1-1	TS.2.1-1
TS.2.2-1	--
Figure TS.2.1-1	Figure TS.2.1-1
TS.2.3-2	TS.2.3-2
TS.2.3-3	TS.2.3-3
TS.3.4-1	TS.3.4-1
Table TS.4.1-2A	Table TS.4.1-2A (Page 1 of 2)
--	Table TS.4.1-2A (Page 2 of 2)
TS.6.4-1	--
B.2.1-1	B.2.1-1
B.2.1-2	B.2.1-2
--	B.2.1-3
--	B.2.1-4
--	B.2.1-5
--	Figure B.2.1-1
B.2.2-1	B.2.2-1
B.3.1-2	B.3.1-2
B.3.1-3	B.3.1-3
B.3.4-1	B.3.4-1
--	B.3.4-2

TECHNICAL SPECIFICATIONS

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APPENDIX A TECHNICAL SPECIFICATIONS

LIST OF FIGURES

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTING

2.1 SAFETY LIMITS

A. Reactor Core Safety Limits

In MODES 1 and 2, combination of thermal power (measured in ΔT), pressurizer pressure, and the highest reactor coolant system loop average temperature shall not exceed the limits shown in Figure TS.2.1-1.

B. Reactor Coolant System Pressure Safety Limit

In MODES 1, 2, 3, 4, and 5, the reactor coolant system pressure shall not exceed 2735 psig.

2.2 SAFETY LIMIT VIOLATIONS

A. If SAFETY LIMIT 2.1.A. is violated, restore compliance and be in MODE 3 within 1 hour.

B. If SAFETY LIMIT 2.1.B. is violated:

1. In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

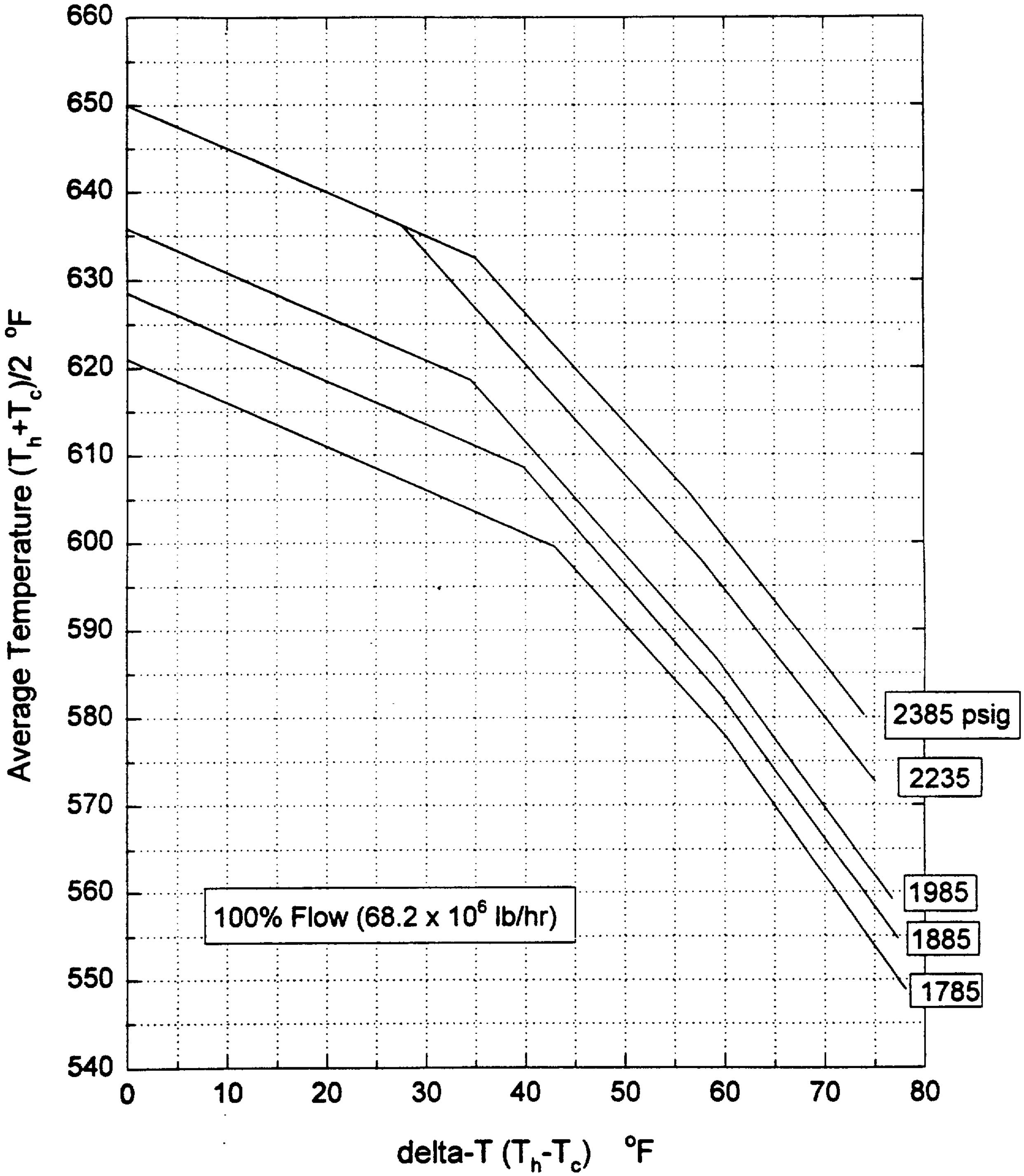
2. In MODE 3, 4, or 5, restore compliance within 5 minutes.

C. If a SAFETY LIMIT is violated, within 1 hour notify the NRC Operations Center in accordance with 10CFR50.72.

D. If a SAFETY LIMIT is violated, within 24 hours notify the Vice President Nuclear Generation, and the Chairman of the Safety Audit Committee or their designated alternates.

E. If a SAFETY LIMIT is violated, within 30 days a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC, the Vice President Nuclear Generation and the Safety Audit Committee.

F. If a SAFETY LIMIT is violated, operation of the unit shall not be resumed until authorized by the NRC.



Reactor Core Safety Limits

Figure TS.2.1-1

2.3.A.2.d Cont.

and f (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chamber, with gains to be selected based on measured instrument response during plant startup tests, such that where q_t and q_b are the percent power in the top and bottom halves of the core, respectively, and $q_t + q_b$ is total core power in percent of rated power:

1. for $q_t - q_b$ within -12% and +9%, f (ΔI) = 0, and

2. for each percent that the magnitude of $q_t - q_b$ exceeds +9% the ΔT trip set point shall be automatically reduced by an equivalent of 2.5 percent of RATED THERMAL POWER.

3. for each percent that the magnitude of $q_t - q_b$ exceeds -12%, the T trip set point shall be automatically reduced by an equivalent of 1.5 percent of RATED THERMAL POWER.

e. Overpower Δ T

$$\Delta T_p \le \Delta T_o \left[K_4 - \frac{K_5 t_3 s T}{1 + t_3 s} - K_6(T-T') - f(\Delta I)\right]$$

where

ΔT_o = Indicated ΔT at RATED THERMAL POWER
T = Average temperature, °F
T' = 567.3°F
K_4 ≤ 1.10
K_5 = 0.0275 for increasing T; 0 for decreasing T
K_6 = 0.002 for $T > T'$, 0 for $T < T'$
t_3 = 10 sec
$f(\Delta I)$ = as defined in d. above

f. Low reactor coolant flow per loop - ≥90% of normal indicated loop flow as measured at loop elbow tap.

2.3.A.2.g. Reactor coolant pump bus undervoltage - ≥75% of normal voltage.

h. Open reactor coolant pump motor breaker.

Reactor coolant pump bus underfrequency - ≥58.2 Hz

i. Power range neutron flux rate.

1. Positive rate - ≤15% of RATED THERMAL POWER with a time constant ≥2 seconds

2. Negative rate - ≤7% of RATED THERMAL POWER with a time constant of ≥2 seconds

3. Other reactor trips

a. High pressurizer water level - ≤90% of narrow range instrument span.

b. Low-low steam generator water level - ≥5% of narrow range instrument span.

c. Turbine Generator trip

1. Turbine stop valve indicators - closed

2. Low auto stop oil pressure - ≥45 psig

d. Safety injection - See Specification 3.5

3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the operating status of the steam and power conversion system.

Objective

To specify minimum conditions of steam-relieving capacity and auxiliary feed- water supply necessary to assure the capability of removing decay heat from the reactor, and to limit the concentration of activity that might be released by steam relief to the atmosphere.

Specification

A. Steam Generator Safety and Power Operated Relief Valves

1. A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 350°F unless the following conditions are satisfied (except as specified in 3.4.A.2 below):

   a. Ten steam generator safety valves shall be OPERABLE with lift settings of 1077, 1093, 1110, 1120 and 1131 psig ± 3% except during testing.

   b. Both steam generator power-operated relief valves for that reactor are OPERABLE.

2. During STARTUP OPERATION or POWER OPERATION, the following condition of inoperability may exist provided STARTUP OPERATION is discontinued until OPERABILITY is restored. If OPERABILITY is not restored within the time specified, be in at least HOT SHUTDOWN within the next 6 hours and reduce reactor coolant system average temperature below 350°F within the following 6 hours.

   a. One steam generator power-operated relief valve may be inoperable for 48 hours.

B. Auxiliary Feedwater System

1. A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 350°F unless the following conditions are satisfied (except as specified in 3.4.B.2 below):

   a. For single unit operation, the turbine-driven pump associated with that reactor plus one motor-driven pump are OPERABLE.

   b. For two-unit operation, all four auxiliary feedwater pumps are OPERABLE.

   c. Valves and piping associated with the above components are OPERABLE except that during STARTUP OPERATION necessary changes may be made in motor-operated valve position. All such changes shall be under direct administrative control.

Table TS.4.1-2A (Page 1 of 2)

MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

Equipment	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 Quarter	7
3. Pressurizer Safety Valves	Verify OPERABLE in accordance with the Inservice Testing Program (± 3%). Following testing, lift settings shall be within ±1%	Per ASME Code, Section XI Inservice Testing Program	-
4. Main Steam Safety Valves	Verify each required lift setpoint in accordance with the Inservice Testing Program (± 3%). Following testing, lift settings shall be within ±1%	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	-

Table TS.4.1-2A (Page 2 of 2)

MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

Equipment	Test	Frequency	FSAR Sect. Reference
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	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 missle ejection incident of $1.0x10^{-5}$ per year. In no case shall the turbine valve test interval exceed one year.	10

2.1 SAFETY LIMITS

A. Reactor Core Safety Limits

Bases

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating the hot regions of the core within the nucleate boiling regime of heat transfer wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters; thermal power, reactor coolant temperature and pressure have been related to DNB through the W-3 and WRB-1 DNB correlations. The W-3 DNB correlation is used for Exxon fuel. The WRB-1 DNB correlation is used for Westinghouse fuel. The W-3 and WRB-1 DNB correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNB ratio, DNBR, during steady state operation, normal operational transients, and anticipated transients is limited to 1.30 for the Exxon fuel using the W-3 correlation and to 1.17 for the Westinghouse fuel using the WRB-1 correlation. There is a third DNBR limit specifically for the steam line break accident but it does not apply to the safety limit curve calculations. These limits correspond to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The safety limit curves of Figure TS.2.1-1 define the regions of acceptable operation with respect to average temperatures, power, and pressurizer pressure. These boundaries of acceptable operations are limited by the thermal overpower limit (fuel melting), thermal overtemperature limit (cladding damage based on DNB considerations), and the locus of points where the steam generation safety valves open. These limits are used to set the overpower and overtemperature ΔT trip setpoints.

The safety limit curves of Figure TS.2.1-1 comprise the most limiting of the following four criteria:

1) Vessel Exit Temperature < 650°F

This is the design temperature limit. This limit defines the portion of the safety limit curves from 0 ΔT to the first knee for the 2235 and 2385 psig curves. At these pressures, the temperature limit of 650° is more

A. Reactor Core Safety Limits

Bases continued

limiting than the T_{sat} limit. The locus of points is calculated from a heat balance with the minimum RCS flow specified in TS 3.10.J.

2) Vessel Exit Temperature < T_{sat}

This limit ensures that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated water which ensures that the ΔT measured by instrumentation used by the RPS as a measure of core thermal power is proportional to core power. This limit defines the portion of the safety limit curves from 0 ΔT to the first knee for the 1985, 1885 and 1785 psig curves. The locus of points is calculated from a heat balance with the minimum RCS flow specified in TS 3.10.J.

3) Minimum DNBR > 1.3 or 1.17 whichever is applicable

As mentioned before, 1.3 is the DNBR limit for Exxon fuel using the W-3 critical heat flux correlation and 1.17 is the DNBR limit for Westinghouse fuel using the WRB-1 critical heat flux correlation. The locus of points past the first knee at all pressures represents the thermal-hydraulic conditions above which the hot channel has a DNBR less than the limit. The conditions are evaluated using approved DNB methodology. The assumptions used in the calculation include a bypass flow of 6%, an $F_{\Delta H}$ greater than 1.75, and a rod bow penalty of 2.6%. The very shallow knee at full power ΔT occurs because the $F_{\Delta H}$ (hot channel power) is allowed to increase for core power less than RATED THERMAL POWER as described in TS 3.10.B.1.

4) Hot Channel Exit Quality < 15% or 30% whichever is applicable

This limit is typically not the most restrictive because it is generally approached at lower powers where the T_{exit} < T_{sat} or 650°F is more limiting. However, it is considered when the DNB calculations described above are performed using approved DNB methodology. This limit is determined by the range of the channel exit quality for the critical heat flux correlations. The maximum channel exit quality limit is 15% for the W-3 correlation and 30% for the WRB-1 correlation.

Operation above the safety limit curves of Figure TS 2.1-1 is not acceptable. At each pressure the safety limit curve is the most restrictive combination of the four limits discussed above. The area of acceptable operation below the safety limit curves is bounded by the OTΔT trip, the OPΔT trip, and the locus of points where the steam generator (main steam) safety valves open. The ΔT trips are set conservatively with respect to the safety limit curves to protect the core from exceeding the safety limits. The locus of points at which the steam generator safety valves open defines the thermodynamic limit of temperature conditions in the RCS based on the maximum pressure in the steam generators. For this calculation, it is assumed that the pressure in the steam generator is 1195 psig which is 110% of design pressure. It is

A. Reactor Core Safety Limits

Bases continued

required that the steam generator safety valves protect the pressure from exceeding 110% of design pressure so using 1195 psig in the calculations is conservative. Thus, the reactor is protected from violating the safety limits by the physical limit of the ΔT trips and the opening of the steam generator safety valves.

As an example, all the limits for the 2235 psig curve are plotted in Figure B.2.1-1 along with the ΔT trips and the locus of points where the steam generator safety valves open. This plot demonstrates that the ΔT trips and the steam generator safety valves do protect the reactor from exceeding the safety limits. Note, however, that the OTΔT trip locus on that plot is for steady state conditions and that the locus will drop in response to the rate at which the ΔT is increasing. In addition, f(ΔI) increasing will also lower the OTΔT trip locus.

The safety limit curves are plotted with ΔT on the x-axis for the following two reasons: 1.) the full power ΔT is different at different temperatures and pressures because water properties are nonlinear. This makes it difficult to plot the curves at each pressure using the same scale for the percent power axis. 2.) the ΔT trip setpoints which the reactor protection system actually calculates is based on the ΔT, not the percent power.

Except for special tests, POWER OPERATION with only one loop or with natural circulation is not allowed. Safety limits for such special tests will be determined as a part of the test procedure.

The curves are conservative for the following nuclear hot channel factors:

$$F^N_{\Delta H} = F^{RTP}_{\Delta H}\,[1 + PFDH(1-P)] \;;\; \text{and } F^N_Q = F^{RTP}_Q$$

where:

- F^{RTP}_Q is the F_Q limit at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT.

- $F^{RTP}_{\Delta H}$ is the $F_{\Delta H}$ limit at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT.

- PFDH is the Power Factor Multiplier for $F^N_{\Delta H}$ specified in the CORE OPERATING LIMITS REPORT

Use of these factors results in more conservative safety limits than would result from power distribution limits in Specification TS.3.10.

A. <u>Reactor Core Safety Limits</u>

<u>Bases</u> continued

This combination of hot channel factors is higher than that calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are covered by Specification 3.10. Adverse power distribution factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits specified in the CORE OPERATING LIMITS REPORT assure that the DNB ratio is always greater at part power than at full power.

The Reactor Control and Protective System is designed to prevent any anticipated combination of transient conditions that would result in a DNB ratio of less than 1.30 for Exxon Nuclear fuel and less than 1.17 for Westinghouse fuel.

B. Reactor Coolant System Pressure Safety Limit

Bases

The reactor coolant system (Reference 1) serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure the reactor coolant system is the primary barrier against the release of fission products. By establishing a system pressure limit, the continued integrity of the reactor coolant system is assured. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME Code, Section III is 110% of design pressure.

The maximum transient pressure allowable in the reactor coolant system piping, valves and fittings under USAS Section B31.1 is 120% of design pressure. Thus, the safety limit of 2735 psig (110% of design pressure) has been established (Reference 2).

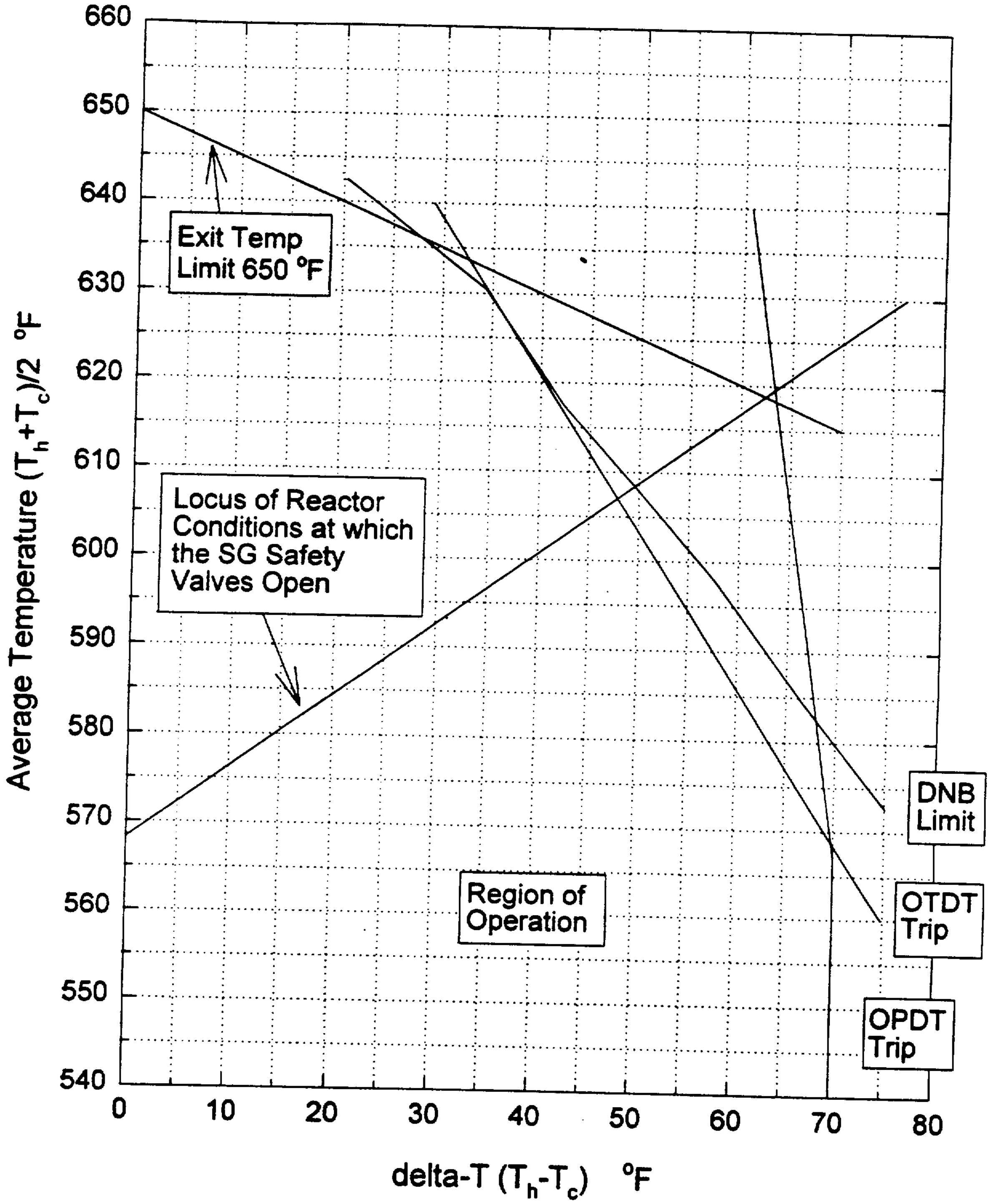
The nominal settings of the power-operated relief valves, the reactor high pressure trip and the safety valves have been established to assure that the pressure never reaches the reactor coolant system pressure safety limit.

In addition, the reactor coolant system safety valves (Reference 3) are sized to prevent system pressure from exceeding the design pressure by more than 10 percent (2735 psig) in accordance with Section III of the ASME Boiler and Pressure Vessel Code, assuming complete loss of load without a direct reactor trip or any other control, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valves settings.

As an assurance of system integrity, the reactor coolant system was hydrotested at 3107 psig prior to initial operation (Reference 4).

References

1. USAR, Section 4.1
2. USAR, Section 4.1.3.1
3. USAR, Section 4.4.3.2
4. USAR, Section 4.1



Origin of Safety Limit Curves at 2235 psig
with delta-T Trips
and Locus of Reactor Conditions at
which the SG Safety Valves Open

Figure B.2.1-1

2.2 SAFETY LIMIT VIOLATIONS

Bases

If the reactor core SAFETY LIMIT 2.1.A is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SAFETY LIMIT is not applicable.

The allowed completion time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SAFETY LIMIT is not applicable, and reduces the probability of fuel damage.

If the Reactor Coolant System pressure SAFETY LIMIT 2.1.B is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the Reactor Coolant System pressure SAFETY LIMIT may cause immediate Reactor Coolant System failure and create a potential for radioactive releases in excess of 10CFR100, "Reactor Site Criteria", limits.

The allowable completion time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

If the Reactor Coolant System pressure SAFETY LIMIT 2.1.B is exceeded in MODE 3, 4, or 5, Reactor Coolant System pressure must be restored to within the SAFETY LIMIT value within 5 minutes. Exceeding the Reactor Coolant System pressure SAFETY LIMIT in MODE 3, 4, or 5 is more severe than exceeding this SAFETY LIMIT in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SAFETY LIMIT within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

If either SAFETY LIMIT in 2.1.A or 2.1.B is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10CFR50.72.

If either SAFETY LIMIT in 2.1.A or 2.1.B is violated, the Vice President Nuclear Generation shall be notified within 24 hours. The 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to senior management.

If either SAFETY LIMIT in 2.1.A or 2.1.B is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC and the Vice President Nuclear Generation. This requirement is in accordance with 10CFR50.73.

If either SAFETY LIMIT in 2.1.A or 2.1.B is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

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. These valves are considered OPERABLE at ± 3% of their setpoint of 2485 psig. Following testing the valve lift settings are restored within a nominal ± 1% of their setpoint. 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.

3.1 REACTOR COOLANT SYSTEM

Bases continued

A. Operational Components (continued)

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

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

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

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

3.4 STEAM AND POWER CONVERSION SYSTEMS

Bases

A reactor shutdown from power requires removal of decay heat. Decay heat removal requirements are normally satisfied by the steam bypass to the condenser and by continued feedwater flow to the steam generators. Normal feedwater flow to the steam generators is provided by operation of the turbine-cycle feedwater system.

The ten steam generator safety valves have a total combined rated capability of 7,745,000 lbs/hr. The total full power steam flow is 7,094,000 lbs/hr; therefore, the ten steam generator safety valves will be able to relieve the total steam flow if necessary (Reference 1).These valves are considered OPERABLE at ± 3% of their specified setpoint. Following testing the valve lift settings are restored within a nominal ± 1% of their setpoint.

In the unlikely event of complete loss of offsite electrical power to either or both reactors, continued removal of decay heat would be assured by availability of either the steam-driven auxiliary feedwater pump or the motor-driven auxiliary feedwater pump associated with each reactor, and by steam discharge to the atmosphere through the steam generator safety valves. One auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from one reactor. The motor-driven auxiliary feedwater pump for each reactor can be made available to the other reactor. During STARTUP OPERATIONS, the Auxiliary Feedwater motor-operated injection valves maybe less than full open as necessary to faciliate plant startup.

The minimum amount of water specified for the condensate storage tanks is sufficient to remove the decay heat generated by one reactor in the first 24 hours of shutdown. Essentially unlimited replenishment of the condensate storage supply is available from the intake structures through the cooling water system.

The two steam generator power-operated relief valves located upstream of the main steam isolation valves are required to remove decay heat and cool the reactor down following a steam generator tube rupture event (Reference 3) and following a high energy line rupture outside containment (Reference 2). The steam generator power operated relief valves are provided with manual upstream block valves to permit testing at power and to provide a means of isolation.

In order to assure timely response to a steam generator tube rupture event, a steam generator power operated relief valve is considered operable when it is capable of being remotely operated and when its associated block valve is open.

Isolation dampers are required in ventilation ducts that penetrate those rooms containing equipment needed for a high energy line rupture outside containment.

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 gpm primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

References

1. USAR, Section 11.9.4
2. USAR, Appendix I
3. USAR, Section 14.5.4

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 123 AND 116 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 application dated May 4, 1995, as supplemented November 27, 1995, and March 1, 1996, Northern States Power Company, the licensee for the Prairie Island Nuclear Generating Plant (PI), requested a Technical Specification (TS) change that would raise the as-found tolerance of the pressurizer and main steam safety valves (MSSVs) from ± 1 percent to ± 3 percent. The TS change requested does not affect the required as-left setpoint tolerance after testing (± 1 percent). The increase in the acceptable safety valve setpoint tolerances affects TS 3.4.a.1.a. In addition, changes to the safety limit curves were proposed to accommodate this and other changes. Since changes were requested in Section 2 of the TS, the licensee also proposed changes to the section format to conform to the standard TS and to correct typographical errors.

The November 27, 1995, and March 1, 1996, letters provided clarifying information in response to NRC staff questions. This information was within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the Reactor Coolant System (RCS). The safety valves are designed to prevent the system pressure from exceeding the system safety limit. At PI there are two pressurizer safety valves set at 2485 psig. There are a total of 10 MSSVs on the 2 main steam headers which provide overpressure protection for the main steam system. PI safety valves have been found outside the ± 1 percent tolerance following operation and the licensee performed an evaluation to support changing the acceptable as-found setpoint to ± 3 percent. Other licensees have encountered similar difficulties due to setpoint drift.

The safety limit curves of Figure TS.2.1-1 define the regions of acceptable operation with respect to average temperatures, power, and pressurizer pressure. These boundaries of acceptable operations are limited by the thermal overpower limit (fuel melting), thermal overtemperature limit (cladding damage based on departure from nucleate boiling (DNB) considerations), and the locus of points where the steam generator safety valves open. These limits are used to set certain reactor trip setpoints.

3.0 EVALUATION

3.1 Safety Limits Curve and TS Section 2

The proposed changes in this area would adjust for the impact of changing the lift setting tolerances for the pressurizer safety valves and MSSVs. The proposed changes would also remove the curve showing the locus of points at which the MSSVs open and lower the DNB limit part of the curves.

As previously stated, TS Figure 2.1-1 defines the regions of acceptable operation with respect to average RCS temperature, reactor thermal power, and RCS pressure for which the minimum DNB ratio is not less than the safety analysis limit (1.30 for Exxon fuel and 1.17 for Westinghouse fuel). The figure also shows the locus of points at which steam generator safety valves open. The licensee has requested removal of the curve showing the locus of points at which the safety valves open. Since this locus curve is not a safety limit but, rather, represents a limiting condition governed by plant equipment which prevents plant operating conditions from approaching the safety limits, the staff concurs with its removal from Figure 2.1-1. Figure B.2.1-1 has instead been incorporated into TS Bases 2.1 along with the ΔT trips and the locus of points where the steam generator safety valves open to demonstrate that the ΔT trips and the steam generator safety valves do protect the reactor from exceeding the safety limits.

The locus of points past the first knee in Figure 2.1-1 for all pressures represents the thermal-hydraulic conditions above which the hot channel has a DNBR less than the limit. The licensee has proposed to lower this portion of the curves because the bypass flow fraction was increased to 6 percent when the fuel thimble plugs were removed, the $F^N_{\Delta H}$ limit was increased to 1.75 with the loss-of-coolant accident (LOCA) analysis performed for Unit 2 Cycle 16, and a rod bow penalty of 2.6 percent was added to the DNB calculation. The staff finds this proposed change acceptable since it was evaluated using approved DNB methodology and provides appropriate safety margins in the curves of Figure 2.1-1.

The licensee has also proposed changing the x-axis of Figure 2.1-1 from "% Rated Core Power" to "delta-T(T_h-T_c) °F". The staff finds this proposed change acceptable for the following reasons. The full power ΔT is different at different temperatures and pressures because water properties are nonlinear. This makes it difficult to plot the curves at each pressure using the same scale for the percent power axis. In addition, the ΔT trip setpoints which the reactor protection system actually calculates is based on the ΔT, not the percent power. Therefore, the proposed x-axis is a more appropriate variable.

An additional proposed change would remove the curve at 1685 psig and add curves at 1785 and 1885 psig to Figure 2.1-1. Since 1685 psig is a much lower pressure than would ever be achieved in plant operation due to the TS limits of the low pressurizer pressure trip setpoint, its elimination is acceptable.

The addition of the curves at 1785 and 1885 psig ensures that the entire range of pressures allowed by TS 2.3.A.2.c is bounded. This change is, therefore, acceptable.

The remaining proposed changes to TS Section 2 combine the specifications of Sections 2.1 and 2.2 into one section, correct typographical errors, and delete the Safety Limit Violation specification of 6.4 in the Administrative Controls Section of the TS and incorporate it into TS 2.2 in conformance with the Standard Technical Specifications. The Table of Contents has also been revised to reflect the above changes. These proposed changes are administrative and only serve to correct typographical errors and relocate certain specifications to make the TS more clear. The staff has reviewed these administrative changes and finds them acceptable.

The proposed changes to Bases Section 2 support the changes made to TS Section 2 and are, therefore, acceptable.

3.2 <u>Safety Valve Setpoint Tolerance</u>

The pressurizer safety valves (PSVs), in conjunction with other safety systems, provide overpressure protection for the RCS. The valves are designed to prevent the RCS from exceeding 110 percent of the design pressure, which is the safety limit, for the most limiting overpressure transient. For Prairie Island, 110 percent of the design pressure is 2735 psig. Prairie Island has two PSVs which are set at 2485 psig.

As a result of increasing the acceptable tolerance of the pressurizer safety valves to +3 percent, the RCS may reach a higher peak pressure than the previously calculated value. The licensee has performed calculations to show that the increased pressure is acceptable. The most severe overpressure transient for the RCS at Prairie Island is a loss of external load or a turbine trip. The licensee recalculated the maximum RCS pressure for the turbine trip using the following conservative assumptions:

1. No credit is taken for the anticipatory turbine trip. The reactor trip was modeled to occur later in the transient as a result of high RCS pressure.

2. No credit is taken for turbine bypass to the condenser.

3. No credit is taken for the pressurizer sprays to reduce pressure.

4. No credit is taken for the pressurizer power-operated relief valves (PORVs).

5. No credit is taken for the main steam PORVs.

6. The transient occurs at 30 psi higher than normal operating RCS pressure and at 102 percent of rated power due to instrument uncertainty.

The pressurizer and MSSVs all were modeled to open at 3 percent higher than their setpoint with a 3 percent accumulation. The results show that the peak RCS pressure for this limiting transient is 2560 psig which is lower than the maximum allowable peak pressure of 2735 psig and is therefore acceptable.

With regard to the -3 percent tolerance, the plant has been analyzed assuming a pressurizer PORV opens. The PORV setpoint is below the -3 percent tolerance of the PSVs. Therefore, for conditions under which the PORV is modeled to open below the lower tolerance (-3 percent), no further analysis is required. Additionally, the reactor trip setpoint is below the -3 percent tolerance of the PSVs to preclude the valve opening prior to the reactor trip on high RCS pressure. No LOCAs were reanalyzed because RCS pressure continually drops and the PSVs will not be challenged following a LOCA.

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against RCS overpressurization by providing a means to remove energy from the RCS. The MSSVs are required to limit the secondary system pressure to 110 percent of the design pressure. The design pressure for the Prairie Island secondary system is 1085 psig and the safety limit, 110 percent of the design pressure, is 1195 psig. Additionally, the MSSV design capacity is to relieve 110 percent of rated steam flow at 110 percent steam generator design pressure. The modification does not physically affect the valves or the relieving capacity. Prairie Island has five MSSVs on each steam line with setpoints of 1077 psig, 1093 psig, 1110 psig, 1120 psig, and 1131 psig.

As a result of increasing the acceptable tolerance of the MSSVs to +3 percent the secondary system may reach a higher pressure than the previously analyzed value. The licensee has performed calculations to show that the increased pressure is acceptable. The most severe overpressure transient for the secondary system at Prairie Island is also loss of external load or a turbine trip. The licensee recalculated the maximum pressure for the turbine trip using the same conservative assumptions as in the case for the peak RCS analysis with the exception of the RCS pressurizer pressure controller which is modeled to continue to operate and slow RCS pressure increase and delay the reactor trip. This maximizes the secondary system pressure and is acceptable.

The pressurizer and MSSVs all were modeled to open at 3 percent higher than their setpoint with a 3 percent accumulation. The results show that the peak main steam pressure for this limiting transient is 1153 psig which is lower than the 1195 psig safety limit and therefore acceptable.

The licensee reanalyzed the small break LOCA (SBLOCA) because the secondary system is relied on to remove some of the decay heat from the primary RCS following the SBLOCA. The results for the SBLOCA reanalysis show that the emergency core cooling system acceptance criteria continue to be met and the results are acceptable. For the large break LOCAs no analysis was required because the secondary system is not relied on to remove any decay heat from the primary RCS. Additionally, the licensee analyzed the effect of the modification on the auxiliary feedwater flow and determined that safety valves will maintain steam generator pressure low enough to assure minimum required auxiliary feedwater flow. The staff finds these analyses acceptable.

The reanalyses necessitated changes to TS Table 4.1-2A, Minimum Frequencies for Equipment Tests. The licensee proposed changes to items 3 and 4 of TS Table 4.1-2A to reflect the changes in pressurizer and MSSV setpoints. In addition, TS Table 4.1-2A has been reformatted and the footnote for item 11 has been relocated to become part of the Table. These changes were reviewed by the staff and are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Minnesota State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements 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 (60 FR 47621). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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