Docket File
— Docket No. 50-306

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

December 17, 1980

Docket Nos. 50-282
and 50-306

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

Dear Mr. Mayer:

The Nuclear Regulatory Commission has issued Amendment Nos. 44 and 38 to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, respectively, in response to your application dated May 6, 1980 as supplemented September 19 and December 2, 1980.

The amendments revise the common station Technical Specifications for the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2 with respect to the current logic for actuation of safety injection and with respect to the control rod and power distribution limits. During our review of your proposed amendments we found that certain modifications were necessary to meet our requirements. Your staff has agreed to these modifications and they have been incorporated in these amendments.

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

Sincerely,

R. A. Clark

R. A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 44 to DPR-42
2. Amendment No. 38 to DPR-60
3. Safety Evaluation
4. Notice of Issuance

cc: w/enclosures
See next page

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cc w/enclosures(s) and incoming
dtd: 5/6/80; 9/19/80; 12/2/80

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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. 44
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 6, 1980, as supplemented September 19 and December 2, 1980, 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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 44, 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

R. A. Clark

R. A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: December 17, 1980

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. 38
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 6, 1980, as supplemented September 19 and December 2, 1980, 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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 38 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

R. A. Clark

R. A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 17, 1980

ATTACHMENT TO LICENSE AMENDMENTS NOS. 44 AND 38

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

DOCKET NOS. 50-282 AND 50-306

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The changed areas on the revised pages are reflected by marginal lines.

Remove Pages	Insert Pages
TS iv	TS iv
TS 3.5-2	TS 3.5-2
Table TS 3.5-1	Table TS 3.5-1
Table TS 3.5-3	Table TS 3.5-3
TS 3.10.1	TS 3.10.1
TS 3.10-1A	
TS 3.10-2	TS 3.10-2
TS 3.10-3	TS 3.10-3
TS 3.10-4	TS 3.10-4
TS 3.10-5	TS 3.10-5
TS 3.10-6	TS 3.10-6
TS 3.10-7	TS 3.10-7
TS 3.10-8	TS 3.10-8
TS 3.10-9	TS 3.10-9
TS 3.10-10	TS 3.10-10
TS 3.10-10a	
TS 3.10-11	TS 3.10-11
TS 3.10-12	TS 3.10-12
TS 3.10-13	TS 3.10-13
TS 3.10-13a	
	TS 3.10-14
	TS 3.10-15
	TS 3.10-16
	TS 3.10-17
Figure TS 3.10-7	Figure 3.10-7

APPENDIX A TECHNICAL SPECIFICATIONS

LIST OF FIGURES

Safety Injection

The Safety Injection System is actuated automatically to provide emergency cooling and reduction of reactivity in the event of a loss-of-coolant accident or a steam line break accident.

Safety injection in response to a loss-of-coolant accident (LOCA) is provided by a high containment pressure signal backed up by the low pressurizer pressure signal. These conditions would accompany the depressurization and coolant loss during a LOCA.

Safety injection in response to a steam line break is provided directly by a low steam line pressure signal, backed up by the low pressurizer pressure signal and, in case of a break within the containment, by the high containment pressure signal.

The safety injection of highly borated water will offset the temperature-induced reactivity addition that could otherwise result from cooldown following a steam line break.

Containment Spray

Containment sprays are also actuated by a high containment pressure signal (Hi-Hi) to reduce containment pressure in the event of a loss of coolant or steam line break accident inside the containment.

The containment sprays are actuated at a higher containment pressure (approximately 50% of design containment pressure) than is safety injection (10% of design). Since spurious actuation of containment spray is to be avoided, it is initiated on coincidence of high containment pressure sensed by three sets of one-out-of-two containment pressure signals provided for its actuation.

Containment Isolation

A containment isolation signal is initiated by any signal causing automatic initiation of safety injection or may be initiated manually. The containment isolation system provides the means of isolating the various pipes passing through the containment walls as required to prevent the release of radioactivity to the environment in the event of a loss-of-coolant accident.

TABLE TS.3.5-1

ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT LIMITING SET POINTS

	FUNCTIONAL UNIT	CHANNEL	LIMITING SET POINTS*
1	High Containment Pressure (Hi)	Safety Injection*	$\leq$4 psig
2	High Containment Pressure (Hi-Hi)	a. Containment Spray	$\leq$23 psig
		b. Steam Line Isolation of Both Lines	$\leq$17 psig
3	Pressurizer Low Pressure	Safety Injection*	$\geq$1815 psig
4	Low Steam Line Pressure	Safety Injection* Lead Time Constant Lag Time Constant	$\geq$500 psig $\geq$12 seconds $\leq$2 seconds
5	High Steam Flow in a Steam Line Coincident with Safety Injection and Low T_{avg}	Steam Line Isolation of Affected Line	d/p corresponding to $\leq 0.745 \times 10^6$ lb/hr at 1005 psig $\geq 540^{\circ}F$
6	High-high Steam Flow in a Steam Line Coincident with Safety Injection	Steam Line Isolation of Affected Line	$\leq$d/p corresponding to 4.5×10^6 lb/hr at 735 psig
7	High Pressure Difference Between Shield Building and Containment	Containment Vacuum Breakers	$\leq$0.5 psi
8	High Temperature in Ventilation Ducts	Ventilation System Isolation Dampers	$\leq 120^{\circ}F$

- *Initiates also containment isolation, feedwater line isolation and starting of all containment fans.
- d/p means differential pressure

TABLE TS.3.5-3

INSTRUMENT OPERATING CONDITIONS FOR EMERGENCY COOLING SYSTEM

FUNCTIONAL UNIT	1 MINIMUM OPERABLE CHANNELS	2 MINIMUM DEGREE OF REDUNDANCY	3 PERMISSIBLE BYPASS CONDITIONS	4 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 or 2 CANNOT BE MET
1. SAFETY INJECTION				
a. Manual	2	1		Hot shutdown **
b. High Containment Pressure	2	1		Hot shutdown **
c. Steam Generator Low Steam Pressure/Loop	2	1	primary pressure less than 2000 psig	Hot shutdown **
d. Pressurizer Low Pressure	2	1	primary pressure less than 2000 psig	Hot shutdown **
2. CONTAINMENT SPRAY				
a. Manual	2	--*		Hot shutdown **
b. Hi-Hi Containment Pressure (Containment Spray)				Hot shutdown **
Channel a	2	1		
Channel b	2	1		
Channel c	2	1		
Logic	2	1		

- * - Must actuate 2 switches simultaneously.
- ** - If minimum conditions are not met within 24 hours, steps shall be taken to place the unit in cold shutdown condition.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the limits on core fission power distribution and to the limits on control rod operations.

Objective

To assure 1) core subcriticality after reactor trip, 2) acceptable core power distributions during power operation, and 3) limited potential reactivity insertions caused by hypothetical control rod ejection.

Specification

A. Shutdown Reactivity

The shutdown margin with allowance for a stuck control rod assembly shall exceed the applicable value shown in Figure TS.3.10-1 under all steady-state operating conditions, except for physics tests, from zero to full power, including effects of axial power distribution. The shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at hot shutdown conditions if all control rod assemblies were tripped, assuming that the highest worth control rod assembly remained fully withdrawn, and assuming no changes in xenon or boron concentration.

B. Power Distribution Limits

1. At all times, except during low power physics testing, measured hot channel factors, F_Q^N and $F_{\Delta H}^N$, as defined below and in the bases, shall meet the following limits:

$$F_Q^N \times 1.03 \times 1.05 \leq (2.21/P) \times K(Z) \times BU\,(E_j)$$

$$F_{\Delta H}^N \times 1.04 \qquad \leq 1.55 \times [1+ 0.2(1-P)]$$

where the following definitions apply:

(a) K(Z) is the axial dependence function shown in Figure TS.3.10-5.

(b) Z is the core height location.

(c) E_j is the maximum pellet exposure in fuel rod j for which the F_Q^N is being measured.

(d) $BU(E_j)$ is the normalized exposure dependence function for Exxon Nuclear Company fuel shown in Figure TS.3.10-7. For Westinghouse fuel, $BU(E_j) = 1.0$

(e) P is the fraction of full power at which the core is operating. In the F_Q^N limit determination when $P \leq .50$, set $P = 0.50$.

(f) F_Q^N or $F_{\Delta H}^N$ is defined as the measured F_Q^N or $F_{\Delta H}^N$, respectively, with the smallest margin or greatest excess of limit

(g) 1.03 is the engineering hot channel factor, F_Q^E, applied to the measured F_Q^N to account for manufacturing tolerance.

(h) 1.05 is applied to the measured F_Q^N to account for measurement uncertainty.

(i) 1.04 is applied to the measured $F_{\Delta H}^N$ to account for measurement uncertainty

2. Hot channel factors, F_Q^N and $F_{\Delta H}^N$, shall be measured and the target flux difference determined, at equilibrium conditions according to the following conditions, whichever occurs first:

   (a) At least once per 31 effective full-power days in conjunction with the target flux difference determination, or

   (b) Upon reaching equilibrium conditions after exceeding the reactor power at which target flux difference was last determined, by 10% or more of rated power.

   F_Q^N (equil) shall meet the following limit for the middle axial 80% of the core:

   $$F_Q^N \text{ (equil)} \times V(Z) \times 1.03 \times 1.05 \leq (2.21/P) \times K(Z) \times BU(E_j)$$

   where V(Z) is defined Figure 3.10-8 and other terms are defined in 3.10.B.1 above.

3. (a) If either measured hot channel factor exceeds its limit specified in 3.10.B.1, reduce reactor power and the high neutron flux trip setpoint by 1% for each percent that the measured F_Q^N or $F_{\Delta H}^N$ exceeds the 3.10.B.1 limit. Then follow 3.10.B.3(c).

   (b) If the measured F_Q^N (equil) exceeds the 3.10.B.2 limits but not the 3.10.B.1 limit, take one of the following actions:

      1. Within 48 hours place the reactor in an equilibrium configuration for which Specification 3.10.B.2 is satisfied, or

      2. Reduce reactor power and the high neutron flux trip setpoint by 1% for each percent that the measured F_Q^N(equil) x 1.03 x 1.05 x V(Z) exceeds the (2.21/P) x K(Z) x $BU(E_j)$ limit.

(c) If subsequent in-core mapping cannot, within a 24 hour period, demonstrate that the hot channel factors are met, the reactor shall be brought to a hot shutdown condition with return to power authorized up to 50% power for the purpose of physics testing. Identify and correct the cause of the out of limit condition prior to increasing thermal power above 50% power. Thermal power may then be increased provided F_Q^N or $F_{\Delta H}^N$ is demonstrated through in-core mapping to be within its limits.

(d) If two successive measurements indicate an increase in the peak pin power $F_{\Delta H}^N$ with exposure, either of the following actions shall be taken:

1. F_Q^N(equil) shall be multiplied by 1.02 x V(Z) x 1.03 x 1.05 for comparison to the limit specified in 3.10.B.2, or

2. F_Q^N(equil) shall be measured at least once per seven effective full power days until two successive maps indicate that the peak pin power, $F_{\Delta H}^N$ is not increasing.

4. Except during physics tests, and except as provided by Specifications 5 through 8 below, the indicated axial flux difference for at least three operable excore channels shall be maintained within a $\pm 5\%$ band about the target flux difference.

5. Above 90 percent of rated thermal power:

If the indicated axial flux difference of two operable excore channels deviates from its target band, within 15 minutes either eliminate such deviation, or reduce thermal power to less than 90 percent of rated thermal power.

6. Between 50 and 90 percent of rated thermal power:

a. The indicated axial flux difference may deviate from its $\pm 5\%$ target band for a maximum of one* hour (cumulative) in any 24-hour period provided that the difference between the indicated axial flux difference about the target flux difference does not exceed the envelope shown in Figure TS.3.10-6.

b. If 6.a is violated for two operable excore channels then the reactor power shall be reduced to less than 50% power and the high neutron flux setpoint reduced to less than 55% of rated power.

*May be extended to 16 hours during incore/excore calibration.

c. A power increase to a level greater than 90 percent of rated power is contingent upon the indicated axial flux difference of at least three operable excore channels being within the target band.

7. Less than 50 percent of rated thermal power:

a. The indicated axial flux difference may deviate from its target band.

b. A power increase to a level greater than 50 percent of rated power is contingent upon the indicated axial flux difference of at least three operable excore channels not being outside the target band for more than one hour (cumulative) out of the preceding 24 hour period

8. In applying 6a and 7b above, penalty deviations outside the ±5% target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each one minute of power operation outside of the target band at thermal power levels equal to or above 50% of rated thermal power, and

b. One-half minute penalty deviation for each one minute of power operation outside of the target band at thermal power levels between 15% and 50% of rated thermal power.

9. If alarms associated with monitoring the indicated axial flux difference deviations from the ±5% target band are not operable, the indicated axial flux difference value for each operable excore channel shall be logged at least once per hour for the first 24 hours and half-hourly thereafter until the alarms are returned to an operable status. For the purpose of applying this specification, logged values of indicated axial flux difference must be assumed to apply during the previous interval between loggings.

C. Quadrant Power Tilt Limits

1. Except for physics tests, if the percentage quadrant power tilt exceeds 2% but is less than 7%, the rod position indication shall be monitored and logged once each shift to verify rod position within each bank assignment and, within two hours, one of the following steps shall be taken:

a. Correct the tilt to less than 2%

b. Restrict core power level so as not to exceed rated power, less 2% for every percent that the quadrant power tilt ratio exceeds 1.0.

2. If the percentage quadrant power tilt exceeds 2% but is less than 7% for a sustained period of more than 24 hours, or if such a tilt recurs intermittently, the reactor shall be brought to the hot shutdown condition. Subsequent operation below 50% of rating, for testing, shall be permitted.

3. Except for physics tests if the quadrant power tilt ratio exceeds 1.07, the reactor shall be brought to the hot shutdown condition. Subsequent operation below 50% of rating, for testing, shall be permitted.

4. If the core is operating above 85% power with one excore nuclear channel out of service, then the core quadrant power balance shall be determined daily and after a 10% power change using either 2 movable detectors or 4 core thermocouples per quadrant, per Specification 3.11.

D. Rod Insertion Limits

1. The shutdown rods shall be fully withdrawn when the reactor is critical or approaching criticality.

2. When the reactor is critical or approaching criticality, the control banks shall be limited in physical insertion; insertion limits are shown in Figure TS.3.10-2, -3 and -4 for normal and abnormal operating conditions.

3. Control bank insertion may be further restricted by specification 3.10.A if, (1) the measured control rod worth of all rods, less the worth of the worst stuck rod, is less than 5.52% reactivity at the beginning of the first cycle or the equivalent value if measured at any other time, or (2) if a rod is inoperable (Specification 3.10.G).

4. Insertion limits do not apply during physics tests or during periodic exercise of individual rods. The shutdown margin shown in Figure TS.3.10-1 must be maintained except for low power physics testing. For this test the reactor may be critical with all but one high worth full-length control rod inserted for a period not to exceed 2 hours per year provided a rod drop test is run on the high worth full-length rod prior to this particular low power physics test.

E. Rod Misalignment Limitations

1. If a full-length rod cluster control assembly (RCCA) is misaligned from its bank by more than 15 inches, the rod will be realigned or the core power peaking factors shall be determined within 2 hours, and Specification 3.10.B applied. If peaking factors are not determined within 2 hours, the high neutron flux trip setpoint shall be reduced to 85 percent of rating.

2. If the misaligned RCCA is not realigned within a total of 8 hours, the RCCA shall be declared inoperable.

F. Inoperable Rod Position Indicator Channels

1. If a rod position indicator (RPI) channel is out of service then

   a. For operation between 50% and 100% of rating, the position of the RCCA shall be checked directly by core instrumentation (excore detector and/or thermocouples and/or movable incore detectors) every shift or subsequent to rod motion exceeding a total of 24 steps, whichever occurs first.

   b. During operation below 50% of rating, no special monitoring is required.

2. The plant shall be brought to the hot shutdown condition should more than one RPI channel per group or more than two RPI channels per bank be found to be inoperable during power operation.

3. If a full length rod having a rod position indicator channel out of service is found to be misaligned from 1.a. above, then apply Specification 3.10.E.

G. Inoperable Rod Limitations

1. An inoperable rod is a rod which (a) does not trip, (b) is declared inoperable under Specification 3.10.E. or 3.10.H. or (c) cannot be moved by its drive mechanism and cannot be corrected within 8 hours.

2. The plant shall be brought to the hot shutdown condition should more than one inoperable full length rod be discovered during power operation.

3. If the inoperable full-length rod is located below the 200 step level and is capable of being tripped, or if the full-length rod is located below the 30 step level whether or not it is capable of being tripped, then the insertion limits in Figure TS.3.10-3 apply.

4. If the inoperable full-length rod cannot be located, or if the inoperable full-length rod is located above the 30 step level and cannot be tripped, then the insertion limits in Figure TS.3.10-4 apply.

5. If reactor operation is continued with one inoperable full-length rod, the potential ejected rod worth and associated transient power distribution peaking factors shall be determined by analysis within 30 days unless the rod is earlier made operable. The analysis shall include due allowance for nonuniform fuel depletion in the neighborhood of the inoperable rod. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, the plant power level shall be reduced to a level consistent with the safety analysis.

H. Rod Drop Time

At operating temperature and full flow, the drop time of each full-length RCCA shall be no greater than 1.8 seconds from loss of stationary gripper coil voltage to dashpot entry. If the time is greater than 1.8 seconds, the rod shall be declared inoperable.

I. Monitor Inoperability Requirements

1. If the rod bank insertion limit monitor is inoperable, or if the rod position deviation monitor is inoperable, individual rod positions shall be logged once per shift, after a load change greater than 10 percent of rated power, and after 30 inches or more of rod motion.

2. If both the rod position deviation monitor and one or both of the quadrant power tilt monitors are inoperable for 2 hours or more, the nuclear overpower trip shall be reset to 93% of rated power in addition to the increased surveillance requirements.

3. If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs and the calculated power tilt shall be logged every two hours after a load change greater than 10% of rated power.

J. DNB Parameters

The following DNB related parameters limits shall be maintained during power operation:

a. Reactor Coolant System Tavg $\leq 564^{\circ}F$

b. Pressurizer Pressure $\geq$2220 psia*

c. Reactor Coolant Flow $\geq$178,000 gpm

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce thermal power to less than 5% of rated thermal power using normal shutdown procedures.

Compliance with a. and b. is demonstrated by verifying that each of the parameters is within its limits at least once each 12 hours.

Compliance with c. is demonstrated by verifying that the parameter is within its limit after each refueling cycle.

Bases

Throughout the 3.10 Technical Specifications, the terms "rod(s)" and "RCCA(s)" are synonomous.

Shutdown Reactivity

Trip shutdown reactivity is provided consistent with plant safety analyses assumptions. One percent shutdown is adequate except for the steam break analysis, which requires more shutdown reactivity due to the more negative moderator temperature coefficient at end of life (when boron concentration is low). Figure TS.3.10-1 is drawn accordingly.

Power Distribution Control

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operations) and II (Incidents of Moderate frequency) events by: (a) maintaining the minimum DNBR in the core $\geq$1.30 during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding

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*Limit not applicable during either a THERMAL POWER ramp increase in excess of (5%) RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of (10%) RATED THERMAL POWER.

mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

During operation, the plant staff compares the measured hot channel factors, F_Q^N and $F_{\Delta H}^N$, (described later) to the limit determined in the transient and LOCA analyses. The limiting $F_Q(Z)$ includes measurement, engineering, and calculational uncertainties. The terms on the right side of the equations in section 3.10.B.1 represent the analytical limits. Those terms on the left side represent the measured hot channel factors corrected for engineering, calculational, and measurement uncertainties.

$F_Q(Z)$, Height Dependent Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods. The maximum value of $F_Q(Z)$ is 2.21/P for the Prairie Island reactors. This value is restricted further by the K(Z) and $BU(E_j)$ functions described below. The product of these three factors is $F_Q(Z)$.

The K(Z) function shown in Figure TS.3.10-5 is a normalized function that limits $F_Q(Z)$ axially for three reasons. The K(Z) specified for the lowest six (6) feet of the core is based on large break LOCA analyses. Above this region the K(Z) value is based on DNBR requirements since the minimum DNBR would be expected in this region of the core, based on power, pressure, and temperature. The K(Z) value in the uppermost region of the core is based on the small break LOCA analyses. $F_Q(Z)$ in the uppermost region is limited to reduce the PCT expected during a small break LOCA since this region of the core is expected to uncover temporarily for some small break LOCA's.

The $BU(E_j)$ function shown in Figure TS.3.10-7 is a normalized function that limits $F_Q(Z)$ based on exposure dependent analyses for the ENC fuel. These analyses consider pin internal pressure uncertainties, fuel swelling, rupture pressures, and flow blockage.

F_Q^N is the measured Nuclear Hot Channel Factor, defined as the maximum local neutron flux in the core divided by the average neutron flux in the core.

V(Z) is an axially dependent function applied to the equilibrium measured F_Q^N to bound F_Q^N's that could be measured at non-equilibrium conditions. This function is based on power distribution control analyses that evaluated the effect of burnable poisons, rod position, axial effects, and xenon worth.

F_Q^E, Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

The 1.05 multiplier accounts for uncertainties associated with measurement of the power distribution with the moveable incore detectors and the use of those measurements to establish the assembly local power distribution.

$F^N_{\Delta H}$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power. $F^N_{\Delta H}$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizational (x-y) power shapes throughout the core. Thus the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F^N_{\Delta H}$.

In the specified limit of $F^N_{\Delta H}$ there is an 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq 1.55/1.08$. The logic behind the larger uncertainty in this case is that:

(a) abnormal perturbations in the radial power shape (e.g. rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F^N_Q,

(b) the operator has a direct influence on F^N_Q through movement of rods, and can limit it to the desired value, while he has no direct control over $F^N_{\Delta H}$ and,

(c) an error in the predictions for radial power shape, which may be detected during startup physics tests can be compensated for in F^N_Q by tighter axial control, but compensation for $F^N_{\Delta H}$ is less readily available.

When a measurement of $F^N_{\Delta H}$ is taken, experimental error must be allowed for and 4 percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system.

Measurements of the hot channel factors are required as part of startup physics tests, at least once each effective full power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design bases including proper fuel loading patterns. The periodic monthly incore mapping provides additional assurance that the nuclear design bases remain inviolate and identify operational anomalies which would otherwise affect these bases.

For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met; these conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15

inches from the bank demand position. An accidental misalignment limit of 13 steps precludes a rod misalignment greater than 15 inches with consideration of maximum instrumentation error.

2. Control rod banks are sequenced with overlapping banks as described in Technical Specification 3.10.

3. The control bank insertion limits are not violated.

4. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

The permitted relaxation in $F^N_{\Delta H}$ and F^N_Q allows for radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factor limits are met. In specification 3.10, F^N_Q is arbitrarily limited for $P \leq 0.5$ (except for low power physics tests).

The procedures for axial power distribution control referred to above are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically control of flux difference is required to limit the difference between the current value of Flux Difference (ΔI) and a reference value which corresponds to the full power equilibrium value of Axial Offset (Axial Offset = ΔI/fractional power). The reference value of flux difference varies with power level and burnup but expressed as axial offset it varies only with burnup.

The technical specifications on power distribution control assure that the $F_Q(Z)$ upper bound envelope of 2.21/P times Figures TS.3.10-5 and TS.3.10-7 is not exceeded and xenon distributions are not developed which at a later time, would cause greater local power peaking even though the flux difference is then within the limits specified by the procedure.

The target (or reference) value of flux difference is determined as follows: At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with the full length rod control rod bank more than 190 steps withdrawn (i.e., normal full power operating position appropriate for the time in life, usually withdrawn farther as burnup proceeds). This value, divided by the fraction of full power at which the core was operating is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium was noted, no allowances for excore detector error are necessary and indicated deviation of ±5 percent ΔI are permitted from the indicated reference value. Figure TS.3.10-6 shows the allowed deviation from the target flux difference as the function of thermal power.

The alarms provided are derived from the plant process computer which determines the one minute averages of the operable excore detector outputs to monitor indicated axial flux difference in the reactor core and alerts the operator when indicated axial flux difference alarm conditions exist. Two types of alarm messages are output. Above a preset (90%) power level, an alarm message is output immediately upon determining a delta flux (as determined from two operable excore channels) exceeding a preset band about a target delta flux value. Below this preset power level, an alarm message is output if the indicated axial flux difference (as determined from two operable excore channels) exceeded its allowable limits for a preset cumulative (usually 1 hour) amount of time in the past 24 hours. For periods during which the alarm on flux difference in inoperable, manual surveillance will be utilized to provide adequate warning of significant variations in expected flux differences. However, every attempt should be made to restore the alarm to an operable condition as soon as possible. Any deviations from the target band during manual logging would be treated as deviations during the entire preceding logging interval and appropriate actions would be taken. This action is necessary to satisfy NRC requirements; however, more frequent readings may be logged to minimize the penalty associated with a deviation from the target band to justify continued operation at the current power. The time that deviations from the target band occur are normally accumulated by the computer above 15% power. Below 15% the probability of exceeding the allowable limits becomes increasingly smaller as it becomes theoretically impossible to deviate from the target band. Between 15-50% power the deviations are more significant and are accumulated at 1/2 of their actual time. Above 50% the deviations are most significant and their time is accumulated on a one for one time basis.

Strict control of the flux difference (and rod position) is not as necessary during part power operation because xenon distribution control at part power is less significant than control at full power. Allowance has been made in predicting the heat flux peaking factors for less strict control at part power. Strict control of the flux difference is not possible during certain physics tests or during required, periodic, excore calibrations which require larger flux differences than permitted. Therefore, the specifications on power distribution control are not applied during physics tests or excore calibrations; this is acceptable due to the low probability of a significant accident occurring during these operations.

In some instances of rapid plant power reduction, automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band, however to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The consequences of being outside the $\pm 5\%$ target band but within the Figure TS.3.10-6 limit for power levels between 50% and 90% has been evaluated and determined to result in acceptable $F_Q(Z)$ values. Therefore, while the deviation exists the power level is limited to 90 percent or lower depending on the indicated axial flux difference. In all cases the ± 5 percent target band is the Limiting Condition for Operation. Only when the target band is violated do the limits under Figure TS.3.10-6 apply.

If, for any reason, the indicated axial flux difference is not controlled within the ± 5 percent band for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50 percent is required to protect against potentially more severe consequences of some accidents.

As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished by using the boron system to position the full length control rods to produce the required indicated flux difference.

For Condition II events the core is protected from overpower and a minimum DNBR of 1.30 by an automatic protection system. Compliance with operating procedures is assumed as a precondition for Condition II transients, however, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

Quadrant Power Tilt Limits

Quadrant power tilt limits are based on the following considerations. Frequent power tilts are not anticipated during normal operation since this phenomenon is caused by some asymmetric perturbation, e.g. rod misalignment, x-y xenon transient, or inlet temperature mismatch. A dropped or misaligned rod will easily be detected by the Rod Position Indication System or core instrumentation per Specification 3.10.F, and core limits protected per Specification 3.10.E. A quadrant tilt by some other means (x-y xenon transient, etc.) would not appear instantaneously, but would build up over several hours and the quadrant tilt limits are set to protect against this situation. They also serve as a backup protection against the dropped or misaligned rod.

Operational experience shows that normal power tilts are less than 1.01. Thus, sufficient time is available to recognize the presence of a tilt and correct the cause before a severe tilt could build up. During start-up and power escalation, however, a large tilt could be initiated. Therefore, the Technical Specification has been written so as to prevent escalation above 50 percent power if a large tilt is present.

The numerical limits are set to be commensurate with design and safety limits for DNB protection and linear heat generation rate as described below.

The percentage quadrant power tilt of 2% at which remedial and corrective action is required has been set so as to provide DNB and linear heat generation rate protection with x-y power tilts. Analyses have shown that percentage increases in the x-y power peaking factor are less than or equal to twice the increase in the indicated quadrant power tilt.

An increase in F_Q^N is not likely to occur with tilts up to 3% because misaligned control rods producing such tilts do not extend to the unrodded plane, where the maximum F_Q^N occurs.

Therefore, a limiting power tilt of 3 percent can be tolerated. However, a measurement uncertainty is associated with the indicated quadrant power tilt. Thus, allowing for a low measurement of power tilt, the action level of indicated tilt has been set at 2 percent. An alarm is set to alert the operator to an indicated tilt of 2 percent or greater for which action is required. To avoid unnecessary power changes, the operator is allowed two hours in which to verify the actual tilt with in-core mappings or to determine and correct the cause of the tilt.

Should this action not be taken, the margin for uncertainty in F_Q^N is reinstated by reducing the power by 2 percent for each percent of tilt above 1.0, in accordance with the relationship described above, or as required by the restriction on peaking factors.

The upper limit on the quadrant tilt at which hot shutdown is required has been set so as to provide protection against excessive linear heat generation rate. The ratio of overpower to normal operation is approximately 1.15. Since the x-y component of F_Q^N is bounded by the above described relation with indicated quadrant tilt, the overpower linear heat generation rate can be avoided if the indicated tilt is restricted below 7 percent.

Rod Insertion Limits

Rod insertion limits are used to assure adequate trip reactivity, to assure meeting power distribution limits, and to limit the consequences of a hypothetical rod ejection accident. The available control rod reactivity (or excess beyond needs) decreases with decreasing boron concentration. The negative reactivity required to reduce the core power level from full power to zero power is largest when the boron concentration is low since the power defect increases with core burnup.

The intent of the test to measure control rod worth and shutdown margin (Specification 3.10 D.) is to measure the worth of all rods less the worth of the the most reactive rod. The measurement would be anticipated as part of the initial startup program and infrequently over the life of the plant, to be associated primarily with determinations of special interest such as end of life cooldown, or startup of fuel cycles which deviate from normal equilibrium conditions in terms of fuel loading patterns and anticipated control bank worths. These measurements will augment the normal fuel cycle design calculations and place the knowledge of shutdown capability on a firm experimental as well as analytical basis.

An evaluation has been made of anticipated transients and postulated accidents, assuming that they occur during the portion of this test when the reactor is critical with all but one full-length control rod fully inserted. Further, the withdrawn full-length rod is assumed not to trip. As a result of this evaluation, it has been determined that for a steam line break upstream of the flow restrictor, the possibility of core DNB exists. However, even if core damage does result, any core fission product release would be low because of the low fission product inventory during initial startup physics testing; and further, would be contained within the reactor coolant system.

Thus, for the initial startup physics tests, this test will not endanger the health and safety of the public even in the event of highly improbable accidents coupled with the failure of the withdrawn control rod to trip. To perform this test later in life is equally valuable, as stated above. Therefore, this specification has been written to further minimize the likelihood of any hypothesized event during the performance of these tests later in life. This is accomplished by limiting to two hours per year the time the reactor can be in this type of configuration, and requiring that a rod drop test is performed on the rod to be measured prior to performance of test.

Operation with abnormal rod configuration during low power and zero power testing is permitted because of the brief period of the test and because special precautions are taken during the test.

Rod Misalignment Limitation

Rod misalignment requirements are specified to ensure that power distributions more severe than those assumed in the safety analyses do not occur.

Inoperable Rod Position Indicator Channels

The rod position indicator channel is sufficiently accurate to detect a rod ±7 inches away from its demand position. A misalignment less than 15 inches does not lead to over-limit power peaking factors. If the rod position indicator channel is not operable, the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors, and/or core thermocouples, and/or movable incore detectors, will be used to verify power distribution symmetry. These indirect measurements do not have the same resolution if the bank is near either end of the core, because a 15-inch misalignment would have no effect on power distributions. Therefore, it is necessary to apply the indirect checks following significant rod motion.

Inoperable Rod Limitations

One inoperable control rod is acceptable provided that the power distribution limits are met, trip shutdown capability is available, and provided the potential hypothetical ejection of the inoperable rod is not worse than the cases analyzed in the safety analysis report. The rod ejection accident for an isolated fully-inserted rod will be worse if the residence time of the rod is long enough to cause significant non-uniform fuel depletion. The four-week period is short compared with the time interval required to achieve a significant non-uniform fuel depletion.

Rod Drop Time

The required drop time to dashpot entry is consistent with the safety analysis.

Monitor Inoperability Requirements

If either the rod bank insertion limit monitor or rod position deviation monitor are inoperable, additional surveillance is required to ensure adequate shutdown margin is maintained.

If the rod position deviation monitor and quadrant power tilt monitor(s) are inoperable, the overpower reactor trip setpoint is reduced (and also power) to ensure that adequate core protection is provided in the event that unsatisfactory conditions arise that could affect radial power distribution.

Increased surveillance is required, if the quadrant power tilt monitors are inoperable and a load change occurs, in order to confirm satisfactory power distribution behavior. The automatic alarm functions related to quadrant power tilt must be considered incapable of alerting the operator to unsatisfactory power distribution conditions.

DNB Parameters

The RCS flow rate, T_{avg}, and Pressurizer Pressure requirements are based on transient analyses assumptions. The flow rate shall be verified by calorimetric flow data and/or elbow taps. Elbow taps are used in the reactor coolant system as an instrument device that indicates the status of the reactor coolant flow. The basic function of this device is to provide information as to whether or not a reduction in flow rate has occurred. If a reduction in flow rate is indicated below the specification value indicated, shutdown is required to investigate adequacy of core cooling during operation.

For fuel regions with high burnups, the depletion of fissile nuclides and build-up of fission products greatly reduces power production capability. These combined burnup effects reduce $F^N_{\Delta H}$ sufficiently to cover residual rod bow penalties beyond a region average burnup of 40,000 MWD/MTU.

FIGURE TS.3.10-7

Normalized Exposure Dependent Function $BU(E_j)$ for Exxon Nuclear Company Fuel

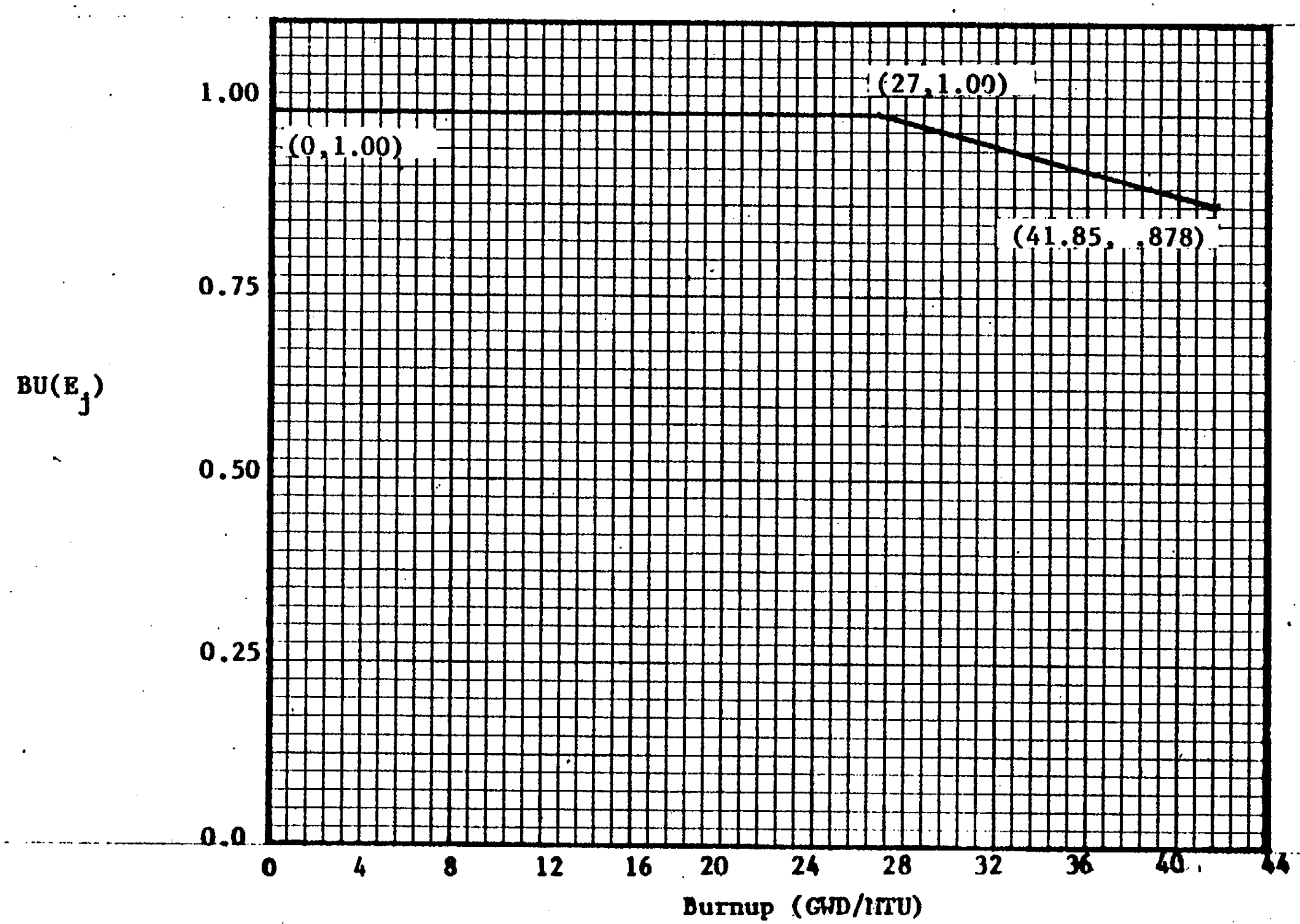

FIGURE TS.3.10-7

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 44 TO FACILITY OPERATING LICENSE NO. DPR-42

AMENDMENT NO. 38 TO FACILITY OPERATING LICENSE NO. DPR-60

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NOS. 1 AND 2

DOCKET NOS. 50-282 AND 50-306

Introduction

By letter dated May 6, 1980 as supplemented by letters dated September 19 and December 2, 1980 Northern States Power Company (the licensee) requested amendments to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2 (PINGP). The proposed amendments would change the Technical Specifications to include editorial changes to reflect the modification to the logic for actuation of safety injection and to include changes in the control rod and power distribution limits.

Discussion

The Commission's evaluation of the changes to the logic for actuation of safety injection is set forth in the Safety Evaluation accompanying Amendment Nos. 36 and 30 to Facility Operating License Nos. DPR-42 and DPR-60, issued on May 1, 1979. The physical modifications authorized by those amendments were completed by the licensee on May 2, 1980 for Unit 1 and on May 8, 1980 for Unit 2. Accordingly the editorial changes to the technical specifications to reflect the presently existing actuation logic may now be made.

The proposed amendments would change the power distribution control requirements involving rod bow penalities, exposure dependence of the peaking factor F_Q^N and an updating of the power distribution control phase 2 methodology. Editorial changes would be made to incorporate consistent terminology, to eliminate reference to part length control rods which were removed from the reactor at an earlier time, to regroup the limiting conditions for operation and the bases and inclusion of more detailed information that is useful to the operator.

Evaluation

SAFETY INJECTION ACTUATION LOGIC

The existing pages TS 3.5-2 and Tables 3.5-1 and 3.5-3 refer to the original safety injection (SI) actuation logic which required coincident low pressurizer pressure and low pressurizer level. This logic has since been modified to a 2-out-of 3 low pressurizer pressure only logic consistent with the previously issued License Amendments Nos. 36 and 30 to License Nos. DPR-42 and DPR-60. Consistent with a previous commitment by the licensee, upon completing the physical modifications to the logic, the licensee's proposed amendment would eliminate editorial references to the previous coincident pressure and level logic. We find these changes to be acceptable.

PART LENGTH RODS

Throughout Section 3.10 of the TS references to the part length rods would be deleted by the proposed amendment. The staff set forth its evaluation and approved the removal of the part length rods from the reactor in Amendment Nos. 32 and 36 to License Nos. DPR-42 and DPR-60. Therefore, deletion of the remaining references to the part length rods is acceptable.

$F^N_{\Delta H}$ Penalty Associated with Rod Bow

The proposed TS revision would delete the multiplier (1-RBP (BU)) for the Westinghouse fuel. The deletion of this multiplier is proposed to be based on a combination of staff positions set forth in two documents, references 1 and 10.

Reference 1 is the staff's evaluation of information presented by Westinghouse as the basis for calculating critical heat flux on bowed rods. The staff's position as a result of that review stated the following:

> "The letter report on CHF with partial and bow provides an acceptable data base for CHF on rods bowed to 85% of the bow necessary for contact. Further, the relation for bow penalty as a function of gap closure, given in Figure 4 is an acceptable bow penalty for use on Westinghouse fuel design."

Figure 4 of reference 1 describes the rod bow on DNBR as a function of fractional gap closure between bowed rods. The staff's position, as presented in reference 16, describes the fractional gap closure as a function of burnup.

Reference 16, in conjunction with reference 1, defines the staff's present position on rod bow DNBR penalty as a function of burnup.

Reference 10 dated February 16, 1973 reports the staff's evaluation of the penalty required to be imposed on DNBR to account for the effects of rod bow. Reference 10 relates the amount of penalty to burnup since the amount of rod bow is a function of burnup. Reference 10 also discusses the offsetting of the total penalty required on DNBR by the application of certain well-known generic thermal margin credits. After these factors are taken into account, Reference 10 stipulates that reductions in $F_{\Delta H}$ are found necessary to account for any remaining penalty on DNBR.

The present TS penalty on $F_{\Delta H}$ is a function of the amount of burnup experienced by the fuel. The removal of the penalty on $F_{\Delta H}$ must therefore be based upon the showing that sufficient margins exist to offset the effects of rod bow for all burnup of significant concern.

Certain well known DNBR design margin credits, as calculated by Westinghouse and verified by the licensee, to be applicable to Prairie Island, are as follows:

Credit	Margin
1.24 DNBR vs 1.30 DNBR	4.8%
Pitch reduction	3.3%
TDC (.019 vs .038)	3.0%
Axial heat flux densification spike effect on DNB	7.0%
TOTAL	18.1%

The staff has determined that these design thermal margin credits will offset the effects of rod bow on DNBR up until a region average burnup of 35,000 MWD/Mtu. We conclude that the depletion of fissile nuclides combined with the buildup of fission products in fuel experiencing more than 35,000 MWD/Mtu will result in a sufficiently low peaking factor F_Q, that $F_{\Delta H}$ will not be limiting for this fuel even with the application of the specified rod bow penalty factor. Accordingly, we find that deletion of the rod bow penalty factor term (1-RBP(BU)) from the equation for $F_{\Delta H}$ is acceptable.

F_Q^N Limit Penalty Associated with Rod Bow

The licensee proposes to delete the multiplier $(1 - 2.35 \times 10^{-6} (BU - 2.8 \times 10^3))$ on F_Q. This multiplier is a penalty on F_Q^N for fuel rod bowing and was applied only to Exxon fuel. Our review of the Exxon bowing models is not complete. These models would impose no penalty on F_Q^N for fuel rod bowing. There is no penalty on F_Q^N for Westinghouse fuel. Since we expect less bowing of the Exxon fuel, primarily because it has thicker cladding, we have reasonable assurance that a bowing penalty on F_Q need not be applied to the Exxon fuel. We therefore find it acceptable to remove this penalty.

Exposure Dependence of F_Q^N

The licensee proposes to add an exposure dependent function which limits F_Q^N toward the top of the core at high exposures. This was done to conform with the NRC high fission gas release model. We reviewed the licensee's application of this model and agree that he has performed a conservative analysis. We therefore find the proposed exposure dependent function limiting F_Q^N acceptable. This function should not affect operation of the Prairie Island units since the F_Q^N limit decrease is not expected until late in the second cycle or early in the third cycle when F_Q^N values will be lower.

Power Distribution Control, Phase II

The licensee proposed several changes which reflect developments in PDC-II.[14] We earlier approved PDC-II for use in the Prairie Island reactors.[15] The current changes reflect developments[3] in our review of PDC-II since this approval. In particular, the form of the F_Q^N and $F_{\Delta H}^N$ limits are specified in a format designed to show that a measured value is being compared to an analytically determined limit. This change does not alter the values of the parameters affected, and clarifies the specification. It is therefore acceptable. Another change involves removal of stringent requirements on reactor operation to ensure that the F_Q^N measurement is performed at equilibrium conditions. Our review of the PDC-II indicates this requirement is no longer necessary because the target flux difference incorporates the effect of slight variations in F_Q^N. Also included in the requested changes is a specification to handle the case in which $F_{\Delta Q}^N$ increases (unexpectedly between F_Q^N measurements). This is part of the standard PDC-II specifications.[3] The licensee had commited to these procedures earlier, however they had not been placed into the Technical Specifications. This change is acceptable because it provides conservative assessment of possible increasing F_Q^N between measurements.

Editorial Changes

The remaining changes requested by the licensee are all editorial. They clarify the specification by word changes or regrouping, remove references to part length rods which have been removed from the reactor, or change wording to conform with the Standard Technical Specifications. These changes have no significant safety impact and are acceptable
The requested changes to the Technical Specification bases are all technically correct and therefore acceptable.

Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR § 51.5(d) (4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

Conclusion

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

Date: December 17, 1980

References

1. Letter, J F Stolz (NRC) to T M Anderson (Westinghouse) dated April 5, 1979 (Staff review of WCAP-8691)

2. Letter, L O Mayer (NSP) to Director of Nuclear Reactor Regulation (NRC) dated March 30, 1979

3. Exxon Nuclear Company Topical Report, "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors Phase II", XN-NF-77-57, Supplement 1, June 1979

4. Exxon Nuclear Company Topical Report, "Fuel Rod Bowing for Exxon Nuclear Company Reload Fuel," XN-NF-79-43, May 1979

5. Letter, G Lear (NRC) to W Nechodom (ENC), dated June 7, 1978

6. Exxon Nuclear Company Report, "Generic Mechanical and Thermal Hydraulic Design for Exxon Nuclear 14 x 14 Reload Fuel Assemblies with Zircaloy Guide Tubes for Westinghouse 2-Loop Pressurized Water Reactors, XN-NF-78-34 [P], November 1978 (proprietary version), and XN-NF-78-34 [NP], January 1979 (non-proprietary version)

7. Exxon Nuclear Company Report, "ECCS Large Break Spectrum Analysis for Prairie Island Unit 1 Using ENC WREM-IIA PWR Evaluation Model," XN-NF-78-46, November 1978

8. Exxon Nuclear Company Topical Report, "Computational Procedures for Evaluating Fuel Rod Bowing (AXIBOW)", XN-NF-75-32, Supplement 1, July 1979

9. Exxon Nuclear Company Report, "Exposure Sensitivity Study for ENC XN-1 Reload Fuel at Prairie Island Unit 1 Using the ENC-WREM-IIA PWR Evaluation Model", XN-NF-79-18 (P), April 1980 (proprietary); XN-NF-79-18(NP), April 1980 (non-proprietary)

10. Interim Safety Evaluation Report on Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors - Revision 1, USNRC, February 16, 1977.

11. Letter, L O Mayer (NSP) to Director of Nuclear Reactor Regulation (NRC), dated October 30, 1979.

12. NUREG-0452, "Westinghouse Pressurized Water Reactor Standardized Technical Specifications".

13. Amendments No. 32 (DPR-42) and 26 (DPR-60), dated November 1, 1978.

References (Cont'd)

14. XN-NF-77-57, Exxon Nuclear Company, "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors, Phase 2," January 1978.

15. U. S. Nuclear Regulatory Commission, "Amendment Nos. 35 and 29 to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, Respectively," April 30, 1979.

16. Memorandum, R. O. Meyer, NRC, to D. F. Ross, NRC, March 2, 1978, Revised Coefficients for Interim Rod Bowing Analysis.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-282 AND 50-306

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 44 and 38 to Facility Operating License Nos. DPR 42 and DPR-60 issued to the Northern States Power Company (the licensee), which revised Technical Specifications for operation of Unit Nos. 1 and 2 of the Prairie Island Nuclear Generating Plant (the facilities) located in Goodhue County, Minnesota. The amendments are effective as of the date of issuance.

The amendments revise the Technical Specifications with respect to the current logic for actuation of safety injection and with respect to control rod and power distribution limits.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated May 6, 1980 as supplemented September 19 and December 2, 1980, (2) Amendment Nos. 44 and 38 to License Nos. DPR-42 and DPR-60 respectively, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Environmental Conservation Library, 300 Nicollet Mall, Minneapolis, Minnesota 55401. A single copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 17th day of December, 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

R. A. Clark

R. A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing