Docket Nos. 50-282
and 50-306

MAY 18 1978

DISTRIBUTION
Dockets(2)
NRC PDR(2)
Local PDR
ORB#1 Rdg
VStello
BGrimes
CParrish
ASchwencer
MGrotenhuis
OELD
OI&E(5)
BJones(8)
BScharf(10)
JMcGough
BHarless
DEisenhut
ACRS(16)
OT/BC
CMiles
TBAbernathy
JRBuchanan

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

Gentlemen:

In response to your request dated August 31, 1977, the Commission has issued the enclosed Amendment Nos. 29 and 23 to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2, respectively.

The amendments consist of changes to the Technical Specifications that relate to the power distribution limits. During our review of your proposed request, we found that certain changes were necessary to meet our requirements. Your staff has agreed to these changes and the changes have been incorporated in these amendments.

Please note, however, that by Order for Modification of License issued on May , 1978 (the same date as the amendments) you are required (1) to submit a reevaluation of ECCS cooling performance as soon as possible and (2) until further authorization to operate with a total nuclear peaking factor (F_Q) limited to maximum allowable 2.24 if accumulator conditions are modified or to 2.21 if the accumulator conditions are not modified.

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

Sincerely,

Original Signed By
M. Grotenhuis

for A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures and cc's:
See next page

FOR PREVIOUS CONCURRENCES PLEASE SEE ATTACHED YELLOW.

x27433:tsb
retyped
5/16/78

OFFICE➤	ORB#1	OELD	ORB#1			
SURNAME➤	MGrotenhuis	EKetchen	ASchwencer			
DATE➤	5/3/78	5/11/78	5/18/78			

NRC FORM 318 (9-76) NRCM 0240

Docket Nos. 50-282
and 50-306

DISTRIBUTION
Dockets(2)
NRC PDR(2)
Local PDR
ORB#1 Reading
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ATTN: Mr. L. O. Mayer, Manager
Nuclear Support Services
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

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The amendments consist of changes to the Technical Specifications that relate to the power distribution limits. During our review of your proposed request, we found that certain changes were necessary to meet our requirements. Your staff has agreed to these changes and the changes have been incorporated in these amendments.

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

Sincerely,

Ketchen
OK with changes

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

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

cc w/enclosures:
See next page

¶ Please note, however, that by Order for Modification of License issued on May __, 1978 (the same date as the amendments) you are required (1) to submit a reevaluation of ECCS cooling performance as soon as possible and (2) until further authorization to operate with a total nuclear peaking factor (F_Q) limited to 2.24 if maximum allowable accumulator conditions are modified or to 2.21 if the accumulator conditions are not modified

X27433:tsb
4/25/78

OFFICE➤	ORB#1	OELD (sub to comments)	ORB#1
SURNAME➤	MGrotenhuis	Ketchen	ASchwencer
DATE➤	5-3-78 ~~4/ /78~~	5/1/78	4/ /78

E.R.

NRC FORM 318 (9-76) NRCM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

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

Docket Nos. 50-282
and 50-306

MAY 18 1978

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

Gentlemen:

In response to your request dated August 31, 1977, the Commission has issued the enclosed Amendment Nos. 29 and 23 to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2, respectively.

The amendments consist of changes to the Technical Specifications that relate to the power distribution limits. During our review of your proposed request, we found that certain changes were necessary to meet our requirements. Your staff has agreed to these changes and the changes have been incorporated in these amendments.

Please note, however, that by Order for Modification of License issued on May 18, 1978 (the same date as the amendments) you are required (1) to submit a reevaluation of ECCS cooling performance as soon as possible and (2) until further authorization to operate with a total nuclear peaking factor (F_Q) limited to maximum allowable 2.24 if accumulator conditions are modified or to 2.21 if the accumulator conditions are not modified.

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

Sincerely,

Marshall Grotenhuis
for A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures and cc's:
See next page

Enclosures:
1. Amendment No. 29 to DPR-42
2. Amendment No. 23 to DPR-60
3. Safety Evaluation
4. Notice of Issuance

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Shaw, Pittman Potts & Trowbridge
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Washington, D.C. 20036

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Executive Director
Minnesota Pollution Control Agency
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Roseville, Minnesota 55113

The Environmental Conservation Library
Minneapolis Public Library
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Minneapolis, Minnesota 55401

Bernard M. Cranum
Bureau of Indian Affairs, DOI
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Minneapolis, Minnesota 55402

Mr. John C. Davidson, Chairman
Goodhue County Board of Commissioners
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Red Wing, Minnesota 55066

State Department of Health
ATTN: Secretary & Executive Officer
University Campus
Minneapolis, Minnesota 55440

Chairman, Public Service Commission of Wisconsin
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Madison, Wisconsin 53702

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U.S. Environmental Protection Agency
Federal Activities Branch
Region V Office
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Joclyn F. Olson, Esquire
Special Assistant Attorney General
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Robert L. Nybo, Jr., Chairman
Minnesota-Wisconsin Boundary Area Commission
619 Second Street
Hudson, Wisconsin 54016

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. 29
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 August 31, 1977, 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 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. 29, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, except as limited by the Order for Modification of License issued on May 18, 1978.

3. This license amendment is effective as of the date of its issuance, provided, however, this amendment is subject to the Order for Modification of License issued on May 18, 1978, which specifies Technical Specification limits for maximum allowable total nuclear peaking factor (F_Q) until further authorization by the Commission.

FOR THE NUCLEAR REGULATORY COMMISSION

Marshall Grotenhuis

for A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: MAY 18 1978

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. 23
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 August 31, 1977, 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 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. 23 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, except as limited by the Order for Modification of License issued on May 18 , 1978.

3. This license amendment is effective as of the date of its issuance, provided, however, this amendment is subject to the Order for Modification of License issued on May 18, 1978, which specifies Technical Specification limits for maximum allowable total nuclear peaking factor (F_Q) until further authorization by the Commission.

FOR THE NUCLEAR REGULATORY COMMISSION

for Marshall Grotenhuis

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: MAY 18 1978

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 29 TO FACILITY OPERATING LICENSE NO. DPR-42

AMENDMENT NO. 23 TO FACILITY OPERATING LICENSE NO. DPR-60

DOCKET NOS. 50-282 AND 50-306

Revise Appendix A as follows:

Remove the following pages and replace with identically numbered pages.

TS 3.10-1	TS Figure 3.10-5
TS 3.10-2	TS Figure 3.10-7
TS 3.10-3	
TS 3.10-7A	
TS 3.10-8	
TS 3.10-9	

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 on 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, boron, or part-length rod position.

B. Power Distribution Limits

1. At all times except during low power physics tests, the hot channel factors defined in the basis must meet the following limits

$$F_Q^N(Z) \leq (2.25/P) \times K(Z) \qquad \text{for } P > .5 \ \underline{1/}$$

$$F_Q^N(Z) \leq (4.5) \times K(Z) \qquad \text{for } P \leq .5$$

$$F_{\Delta H}^N \leq 1.55(1 + 0.2(1-P))(1-RBP(BU))$$

where P is the fraction of full power at which the core is operating. K(Z) is the function given in Figure TS.3.10-5 and Z is the core height location of F_Q^N. RBP(BU) is the Rod Bow Penalty as a function of region average burnup as shown on Figure TS.3.10-7, where region is defined as those assemblies with the same loading date (reloads) or enrichments (first cores).

2. Following initial loading and at regular effective full power monthly intervals thereafter, power distribution maps, using the movable detector system, shall be made to confirm that the hot channel factor limits of this specification are satisfied. For the purpose of this comparison,

1/ Until NRC Order dated May 18 , 1978 is terminated or otherwise modified these values of F_Q^N shall be further reduced by 2.21/2.32 if the accumulator conditions conform to Specification 3.3.A.1.b.(2) or by 2.24/2.32 if the accumulator volumes are reduced as provided for in the Order.

a. The measured peaking factor, F_Q^N, shall be increased by five percent to account for measurement error.

b. The measurement of enthalpy rise hot channel factor, $F_{\Delta H}^N$, shall be increased by four percent to account for measurement error.

If either measured hot channel factor exceeds its limit specified under 3.10.B.1, the reactor power and high neutron flux trip setpoint shall be reduced so as not to exceed a fraction of rated power equal to the ratio of the F_Q^N or $F_{\Delta H}^N$ limit to measured value, whichever is less. 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 $F_Q(Z)$ is demonstrated through in-core mapping to be within its limits.

3. The reference equilibrium indicated axial flux difference for each excore channel as a function of power level (called the target flux difference) shall be measured at least once per equivalent full power quarter. The target differences must be updated monthly. This may be done either by using the measured value for that month or by linear interpolation using the most recent measured value and a value of 0 percent at the end of the cycle life.

4. Except during physics tests, and except as provided by Item 5 through 8 below, the indicated axial flux difference for at least the number of operable excore channels required by TS.3.5 shall be maintained within a ±5% band about their target flux differences (defines the target band on axial flux difference).

5. At a power level greater than 90 percent of rated power, if the indicated axial flux difference of two operable excore channels deviates from its target band, either such deviation shall be eliminated, or the reactor power shall be reduced to a level no greater than 90 percent of rated power.

6. At a power level no greater than 90 percent of rated power,

   a. The indicated axial flux difference may deviate from its ±5% target band for a maximum of one* hour (cumulative) in any 24 hour period provided the flux difference does not exceed an envelope bounded by -11 percent and +11 percent at 90% power and increasing by -1 percent and +1 percent for each 2 percent of rated power below 90% power as shown by Figure TS.3.10-6.

   b. If 6.a is violated for two operable excore channels then the reactor power shll be reduced to no greater than 50% power and the high neutron flux setpont reduced to no greater than 55 percent of rated values.

*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 the number of operable excore channels required by TS.3.5 being within their target bands.

7. At a power level no greater than 50 percent of rated 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 the number of operable excore channels required by TS.3.5 not being outside their target bands for more than one hour (cumulative) out of the preceding 24 hour period

8. For the purpose of determining penalties associated with deviations from the ±5% target band, time for use in applying 6.a and 7.b above shall be accumulated in the following manner:

a. For deviations at, or below 50% power, time shall be accumulated such that a 1 minute actual deviation equals a 1/2 minute accumulative penalty in applying Specifications 6.a and 7.b above

b. For deviations above 50% power, time shall be accumulated on a 1 for 1 basis in applying Specifications 6.a and 7.b above.

9. If for any reason the indicated axial flux difference alarms associated with monitoring 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 such time as 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 of quadrant power tilt above 1.0.

$F_Q(Z)$, 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. F_Q is the product of F_Q^N and F_Q^E.

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.

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

$F_{\Delta H}^N$, 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.

It should be noted that $F_{\Delta H}^N$ 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 horizontal (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 $F_{\Delta H}^N$.

An upper bound envelope of 2.25 times the normalized peaking factor axial dependence of Figure TS.3.10-5 has been determined from extensive analyses considering all operating maneuvers consistent with the technical specifications on power distribution control as given in Section 3.10. The results of the loss of coolant accident analyses based on this upper bound envelope indicate a peak clad temperature of 2187.4 °F corresponding to a 12.6 °F margin to the 2200 °F limit.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for experimental error for a full core map taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance.

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_Q, (b) the operator has a direct influence on F_Q through movement of rods, and can limit it to the desired value, 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_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. The penalties applied to $F^N_{\Delta H}$ to account for rod bow as a function of burnup are consistent with those described in the NRC safety evaluation report, "Interim Safety Evaluation Report on Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors," Revision 1, February 1977.

Measurements of the hot channel factors are required as part of startup physics tests, at least each 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. The part length control rods are not inserted.

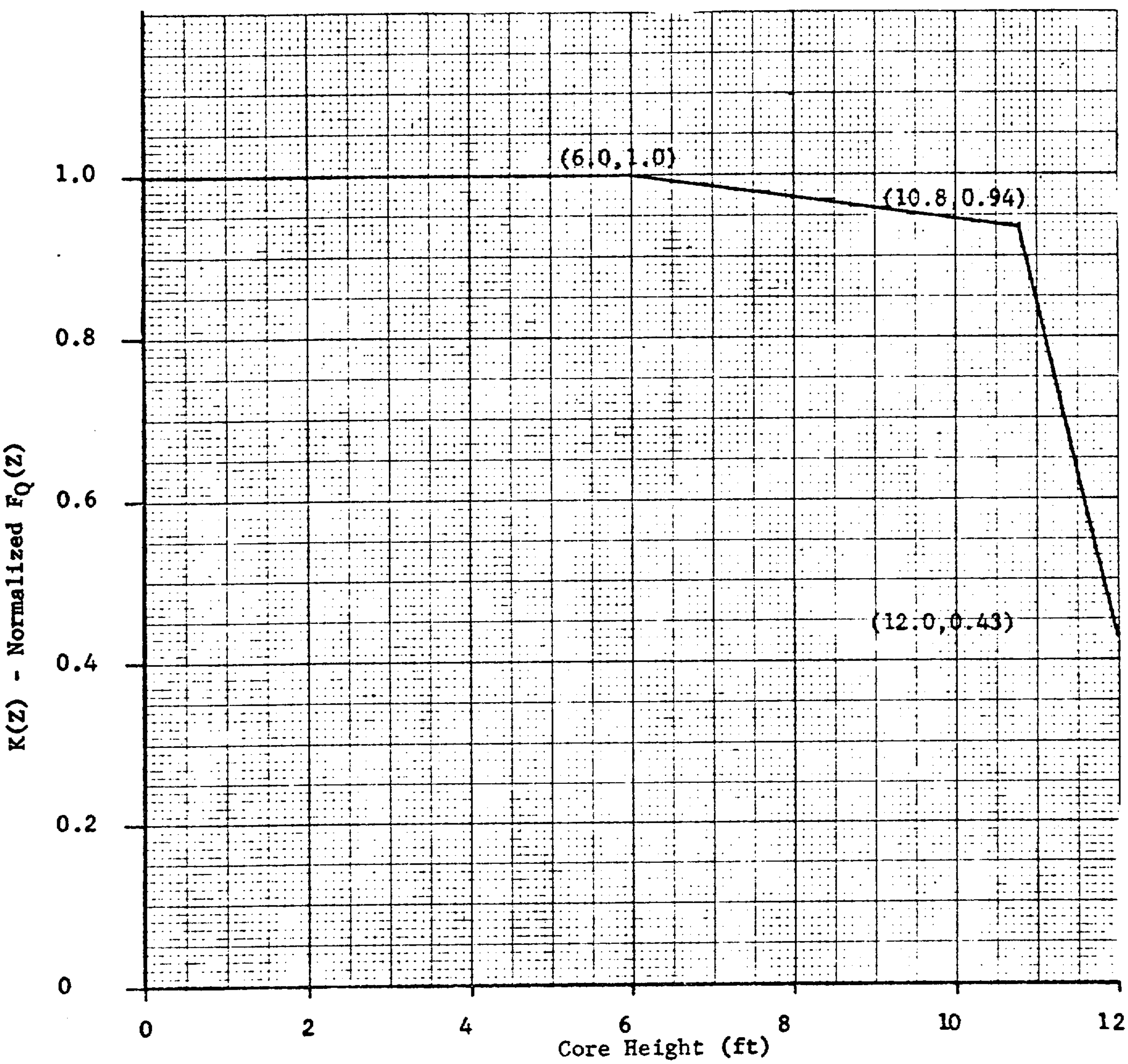
5. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure 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 5 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 $= \Delta I$/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^N_Q upper bound envelope of 2.25 times Figure TS.3.10-5 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 part length rods withdrawn from the core and 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. During periods where extensive load following is required, it may be impractical to establish the required core conditions for measuring the target flux difference every month. For this reason, the specification provides two methods for updating the target flux difference. Figure TS.3.10-6 shows a typical construction of the target flux difference band at BOL.

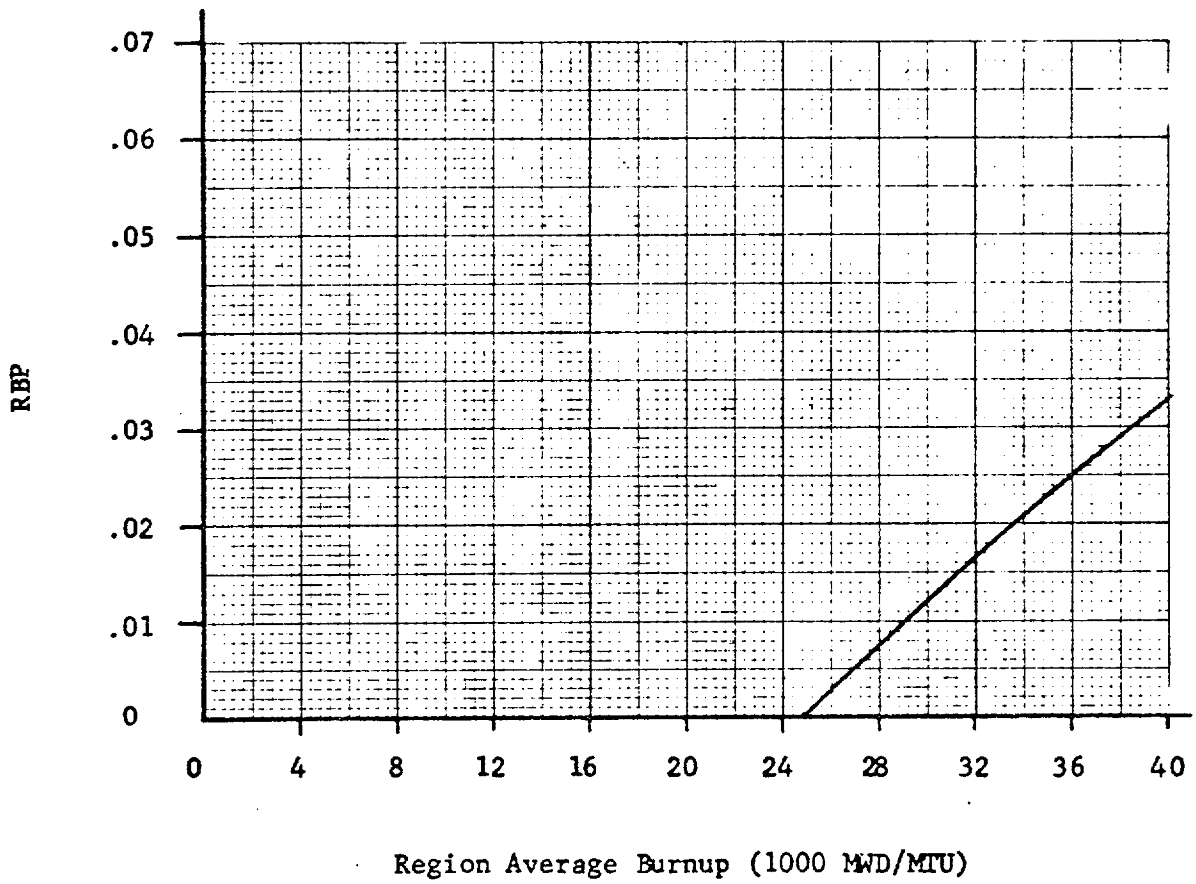


HOT CHANNEL FACTOR NORMALIZED

OPERATING ENVELOPE

FOR F_Q = 2.32

ROD BOW PENALTY (RBP) FRACTION
VERSUS REGION AVERAGE BURNUP



SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 29 TO FACILITY OPERATING LICENSE NO. DPR-42

AMENDMENT NO. 23 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 August 31, 1977, Northern States Power Company (NSP) 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 Specification that relates to power distribution limits. During our review of the proposed request, we found that certain modifications were necessary to meet our requirements. These modifications were discussed with the licensee's staff and they have agreed to the modifications.

Also, during the period of our review of these requests we were notified by Westinghouse Electric Corporation on March 23, 1978 of an ECCS model error which could result in an increase in calculated peak clad temperature in excess of 2200^{o}F unless the allowable peaking factor were reduced somewhat. This matter was further addressed in a Westinghouse letter to the NRC staff on April 7, 1978, in a licensee letter to the NRC staff on April 10, 1978 and culminated in an NRC Order for Modification of License to the licensee on May 18, 1978. The Order for Modification of License requires the licensee (1) to submit as soon as possible a reevaluation of ECCS cooling performance calculated in accordance with the Westinghouse Evaluation Model approved by the NRC staff and corrected for the errors described in the Order and (2) until further authorization by the Commission limits the Technical Specification limit for total nuclear peaking factor (F_Q) for the facilities to maximum allowable 2.24 if the accumulator conditions are modified as specified in the licensee's letter dated April 10, 1978, or to 2.21 if the accumulator conditions are not modified.

Discussion

The amendment request included nine different proposed changes. Of these, using the same numbering system as NSP used in the application, Change 1 is a request for a change in the hot channel factor, Change 4 is a request for a change in the axial flux deviation time during incore/excore calibration, Changes 7, 8 and 9 provide bases and figures supporting Item 1 above. The remainder are miscellaneous changes for the purpose of clarifying the Technical Specifications.

Evaluation:

Item 1 requested the hot channel factor of Specification 3.10.B.1 to be changed to:

"$F_Q^N(Z) \leq (2.25/P) \times K(Z)$ for $P > .5$"

"$F_Q^N(Z) \leq (4.5) \times K(Z)$ for $P \leq .5$"

"$F_{\Delta H}^N \leq 1.55\ (1 + 0.2(1-P))(1-RBP(BU))$"

RBP(BU) is a rod bow penalty which varies as a function of burnup as shown in proposed Technical Specification Figure 3.10-7. In addition we conclude that the following definition should be included in Specification 3.10.B.1:

> "RBP(BU) is the Rod Bow Penalty as a function of region average burnup as shown on Figure TS.3.10-7, where region is defined as those assemblies with the same loading date (reloads) or enrichments (first cores)."

It should be noted that as defined in the Technical Specification on page 3.10-7A, $F_Q = F_Q^E \times F_Q^N$, where F_Q^E is a factor of 1.03. Since the value of F_Q^N in the current Technical Specification at full power and maximum K(Z) is 2.09, then F_Q is correspondingly 2.15. Approval of the changes requested in Item 1 would permit an F_Q^N of 2.25 and an F_Q of 2.32.*

The increase in the allowable $F_Q^N(Z)$ would make the Technical Specification limit consistent with the value obtained in the loss of coolant accident analysis for Prairie Island, which the licensee submitted to us on January 20, 1977.(1) The change to the $F_{\Delta H}^N$ limit is needed to reflect the effects of fuel burnup on this parameter resulting from the rod bow

*An Order for Modification of License issued on May 18, 1978 for these two facilities limits the total peaking factor (F_Q) to 2.24 if accumulator conditions are modified as specified by the Licensee by letter of April 10, 1978, and 2.21 if accumulator conditions are not modified.

penalty factor. The additional definition is needed to clarify the term RBP(BU) used in the equation to determine $F^N_{\Delta H}$. The NSP safety evaluation for the proposed F^N_Q limit was contained in the January 20, 1977 report.(1) In a safety evaluation transmitted to NSP on March 4, 1977 (2) we concluded that the NSP analysis was conservative relative to the 10 CFR 50.46 criteria.

Because the new values for F^N_Q and $F_{\Delta H}$ proposed by NSP are based on previously approved submittals, these values are acceptable to the staff except that the value of F^N_Q is further limited by conditions imposed in our May , 1978 Order dealing with the recently discovered Westinghouse ECCS model error.

Item 4 requested the asterisked item in TS 3.10.B.6.a be changed to read:

> "*May be extended to 16 hours during incore/excore calibrations."

The present Technical Specification states that the axial flux difference may deviate from its $\pm 5\%$ target band for a maximum of 12 hours during incore/excore calibration. The requested change to increase this period to 16 hours will provide for additional flexibility during incore/excore calibrations. Also, an allowance of 16 hours is consistent with the Westinghouse Standard Technical Specifications. We therefore find that the requested change is acceptable.

Items 7, 8 and 9 provide the bases for the changes in the Technical Specifications and submitted additional figures needed for the changes discussed above. Proposed Figure TS 3.10-5 is a graphical representation of the function K(Z) used in the calculation of F^N_Q consistent with the new approved LOCA analysis.(1) Proposed new Figure TS 3.10-7 is a graphical representation of the rod bow penalty function RBP(BU) proposed for use in calculating $F_{\Delta H}$. The function RBP(BU) is consistent with our evaluation(8). The present Figure TS 3.10-7 has been proposed for deletion since it does not constitute a limiting condition for operation. We agree that this figure is informational in nature and that it can be eliminated with no technical impact. We find that the changes requested in Items 7, 8 and 9 are therefore acceptable.

We have reviewed Items 2, 3 5 and 6 and find that they are all minor technical clarifications of the Technical Specifications, are consistent with the NRC staff's intent and are, 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 the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: MAY 18 1978

References

1. "Major Reactor Coolant System Pipe Rupture - October 1975 Evaluation Model", Northern States Power Company, January 29, 1977.

2. Letter from D. L. Ziemann, U.S. NRC, to L. O. Mayer, Northern States Power Company, March 4, 1977.

3. Letter from Edson G. Case, U.S. NRC to L. O. Mayer, Northern States Power Company, December 16, 1977.

4. Letter from L. O. Mayer, NSP to Director NRR dated January 16, 1978.

5. Letter from A. Schwencer, NRC to L. O. Mayer NSP dated February 10, 1978

6. Letter from L. O. Mayer, NSP to Director NRR dated February 24, 1978.

7. Letter from L. O. Mayer, NSP to Director NRR dated March 17, 1978.

8. Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors, U. S. NRC Staff, February 1977.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-282 AND 50-306

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 29 and 23 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 will become effective as of January 1, 1978.

The amendments revised the Technical Specifications for the facilities relating to the power distribution limits. These amendments, which revised the Technical Specifications, are subject to the Order for Modification of License of May 18, 1978. That Order limits the total nuclear peaking factor (F_Q) during operation of the facilities until otherwise further authorized by the Commission and requires the licensee to submit as soon as possible a reevaluation of ECCS cooling performance calculated in accordance with the Westinghouse Evaluation Model, approved by the NRC staff and corrected for the errors described in the Order for Modification of License.

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 the amendments.

For further details with respect to this action, see (1) the application for amendments dated August 31, 1977, (2) Amendment Nos. 29 and 23 to License Nos. DPR-42 and DPR-60, respectively, and (3) the Commission's related Safety Evaluation and (4) the Order for Modification of License dated May 18, 1978, and the related Safety Evaluation referenced in the Order for Modification and dated May 18, 1978. 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 of the Minneapolis Public 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 18th day of May 1978

FOR THE NUCLEAR REGULATORY COMMISSION

Marshall Grotenhuis

Marshall Grotenhuis, Acting Chief
Operating Reactors Branch #1
Division of Operating Reactors