10-4-76

DISTRIBUTION
Docket (2)
NRC PDR (2)
Local PDR
ORB #2 Reading
KRGoller
TJCarter
RMDiggs
OELD - GLewis
MGrotenhuis
EAReeves
OI&E (5)
BJones (8)
JMcGough
ACRS (16)
OPA (CMiles)
VStello
DRoss
TBabernathy
JRBuchanan

Docket Nos. 50-282
50-306

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 February 2, 1976 as supported by filings dated April 14, April 17, July 9 and October 21, 1975 and January 7 and March 1, 1976, and consistent with your letter of August 17, 1976 and the Commission Order of August 27, 1976, the Commission has issued the enclosed Amendment Nos. 16 and 10 to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2.

The amendments revise the Technical Specifications for the facilities to authorize operation of the facilities with modified operating limits based upon an evaluation of emergency core cooling system performance calculated in accordance with an acceptable evaluation model that conforms with the requirements of 10 CFR Part 50 of the Commission's regulations with the following exceptions. The analysis of the single failure criterion will be the subject of a separate action at a later date after receipt and review of the additional information requested in our letter to you dated April 12, 1976. Your ECCS analysis corrected according to the Commission Order of August 27, 1976 must be evaluated. During our review of the proposed changes, we found that certain modifications to the proposal were necessary to meet NRC requirements. These changes were discussed with your staff. Your staff has agreed with these changes and the changes have been incorporated into the amendments.

Copies of the related Safety Evaluation, Negative Declaration, Environmental Impact Appraisal and the Federal Register Notice also are enclosed.

Sincerely,

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

See attached yellow for previous concurrence

OFFICE ➤	Enclosures and cc:	DOR:ORB #2	DOR:ORB #2	OELD	DOR:ORB #2	DOR:AD/OR
SURNAME ➤	See next page	MGrotenhuis:ah	RMDiggs	G. Lewis	DLZiemann	KRGoller
DATE ➤		9/30/76	9/ /76	9/30/76	10/4/76	10/4/76

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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Operating Reactors Branch #2
Division of Operating Reactors

Enclosures and cc:
See next page

OFFICE ➤	DOR:ORB #2	DOR:ORB #2	OELD	DOR:ORB #2	DOR:AD/OR
SURNAME ➤	RMDiggs	MGrotenhuis:ah	G. Lewis	DLZiemann	KRGoller
DATE ➤	8/6/76	8/6/76	8/6/76	8/ /76	8/ /76

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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

October 4, 1976

Docket Nos. 50-282
50-306

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 February 2, 1976 as supported by filings dated April 14, April 17, July 9 and October 21, 1975 and January 7 and March 1, 1976, and consistent with your letter of August 17, 1976 and the Commission Order of August 27, 1976, the Commission has issued the enclosed Amendment Nos. 16 and 10 to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2.

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Copies of the related Safety Evaluation, Negative Declaration, Environmental Impact Appraisal and the Federal Register Notice also are enclosed.

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Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures and cc:
See next page

Enclosures:
1. Amendment No. 16 to License DPR-42
2. Amendment No. 10 to License DPR-60
3. Safety Evaluation
4. Negative Declaration
5. Environmental Impact Appraisal
6. Federal Register Notice

cc w/enclosures:
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Shaw, Pittman, Potts and
Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

Mr. Steve J. Gadler
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Sandra S. Gardebring, Esquire
Special Assistant Attorney General
Minnesota Pollution Control Agency
1935 W. County Road B2
Roseville, Minnesota 55113

The Environmental Conservation
Library
Minneapolis Public Library
300 Nicollet Mall
Minneapolis, Minnesota 55401

Mr. Bernard Cranum
Bureau of Indian Affairs, DOI
831 Second Avenue South
Minneapolis, Minnesota 55402

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

cc w/enclosures and cy of NSPCo
filings dtd. ~~4/17/75~~; 7/9/75;
10/21/75; 1/7/76*; 2/2/76 and
3/1/76:
Mr. Norman M. Clapp, Chairman
Public Service Commission of
Wisconsin
Hill Farms State Office Building
Madison, Wisconsin 53702

Warren H. Lawson, M. D.
Secretary and Executive Officer
State Department of Health
University Campus
Minneapolis, Minnesota 55440

* NON-PROPRIETARY VERSION

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. 16
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 February 2, 1976 (as supported by filings dated April 17, July 9 and October 21, 1975, and January 7 and March 1, 1976) 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.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 4, 1976

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. 10
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 February 2, 1976 (as supported by filings dated April 17, July 9 and October 21, 1975, and January 7 and March 1, 1976) 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.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 4, 1976

ATTACHMENT TO LICENSE AMENDMENT NOS. 16 AND 10

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

DOCKET NOS. 50-282 AND 50-306

Replace the following pages of the Appendix A portion of the Technical Specifications with the attached revised pages bearing the same numbers except as otherwise noted. Changed areas on these pages are shown by marginal lines:

TS 3.10-1
TS 3.10-2
TS 3.10-3
TS 3.10-4
TS 3.10-7
TS 3.10-8
TS 3.10-9
TS 3.10-10
TS 3.10-10a (new page)
TS 3.10-11 (correction of misspelled word only in first line)
TS 3.10-5
TS 3.10-6 (new figure)
TS 3.10-7 (new figure)

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

$$F_Q^N(Z) \leq (2.09/P) \times K(Z) \qquad \text{for } P > .5$$

$$F_Q^N(Z) \leq (4.18) \times K(Z) \qquad \text{for } P \leq .5$$

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

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.

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,

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 only for the purpose of physics testing.

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 Items 5 through 8 below, the indicated axial flux difference for at least the number of operable 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 shall be reduced to no greater than 50% power and the high neutron flux setpoint reduced to no greater than 55 percent of rated values.

* May be extended to twelve 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 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 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.

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. Except during low power physics testing, operation with part length rods shall be restricted such that the part length rod bank is not inserted in the reactor core at any time the reactor is critical.

3. 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.

4. 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).

5. 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 the low power margin. For this test the reactor may be critical with all but one high worth full-length control rod inserted and all part-length rods fully withdrawn 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.

J. Quadrant Power Tilt Monitor

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.

Basis

Design criteria have been chosen for Condition I and II events which are consistent with the fuel integrity analyses of Section 3.2 of the FSAR. These relate to fission gas release, pellet temperature and cladding mechanical properties. Also the minimum DNBR in the core must not be less than 1.30 in normal operation or in short term transients.[2]

In addition to conditions imposed for Condition I and II events, the peak linear power density must not exceed the limiting Kw/ft values which result from the large break loss of coolant accident analysis based on the Final Acceptance Criteria (FAC) limit of 2200°F. This is required to meet the initial conditions assumed for loss of coolant accident. To aid in specifying the limits on power distribution the following hot channel factors are defined.

$F_Q(Z)$, 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 2.15 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 2155°F corresponding to a 45°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.

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 =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 upper bound envelope of 2.15 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 $\pm$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 and Figure TS.3.10-7 shows the typical variation of the full power value with burnup.

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 shall be treated as deviations during the ntire preceding logging interval and appropriate actions shall 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 theorectically 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. This is because xenon distribution control at part power is not as significant as the control at full power and 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 occuring 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 instantaneous consequences of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for axial flux defference in the range of +14 to -14 percent (+11 percent to -11 percent indicated) increasing by ±1 percent for each 2 percent decrease in rated power. 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 Specification TS.3.10.B.6.a apply. Figure TS.3.10-2 shows a typical monthly target band near beginning of life and the appropriate boundaries and alarms if the target band is exceeded.

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, without part-length rods, 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 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 startup 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.

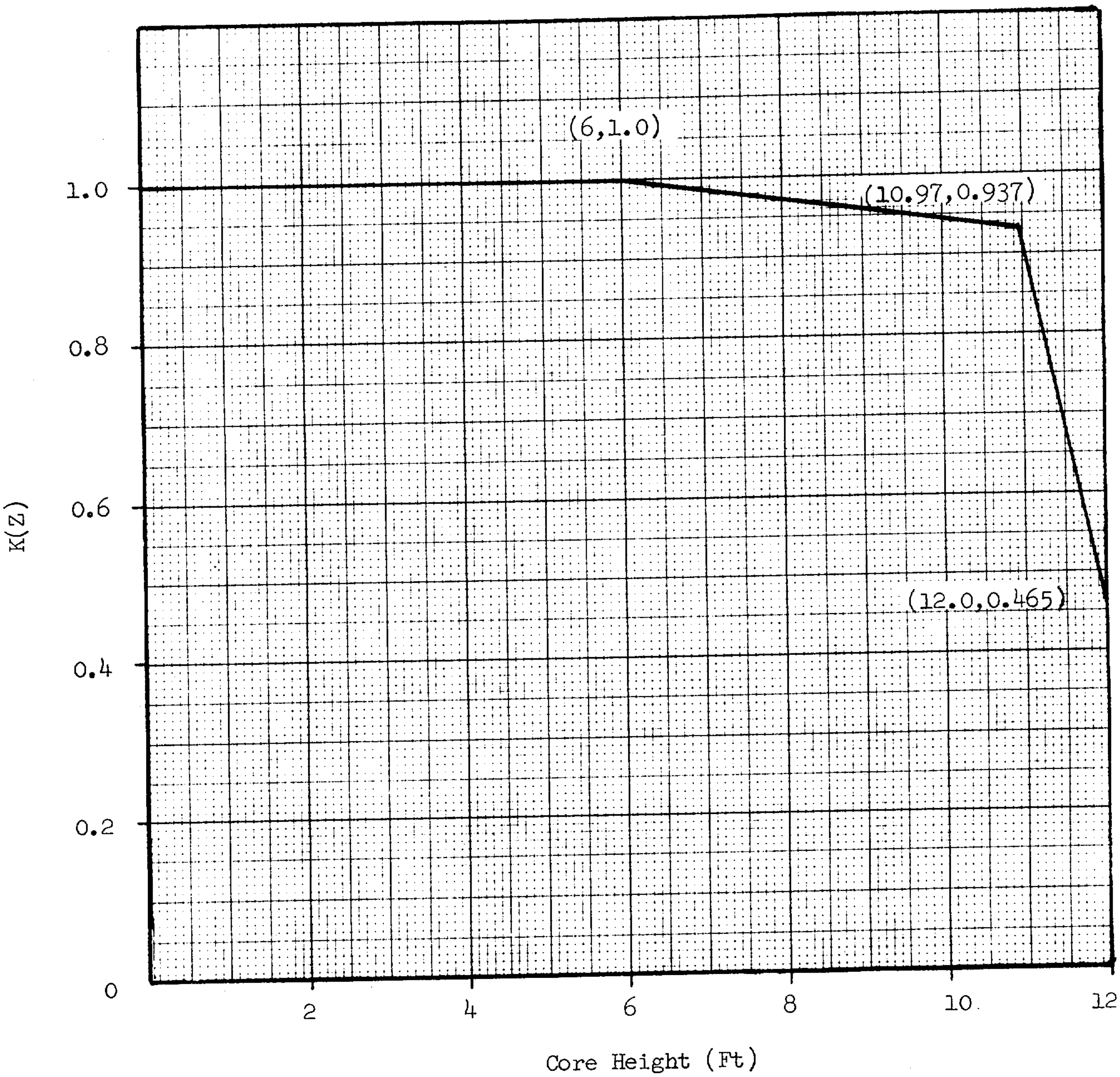
An increase in F_Q is not likely to occur with tilts up to 3% because misaligned control rods producing such tilts do not extend to the unrodded plane, where the maximum F_Q occurs.

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 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 accord with the relationship described above, or as required by the restriction on peaking factors.

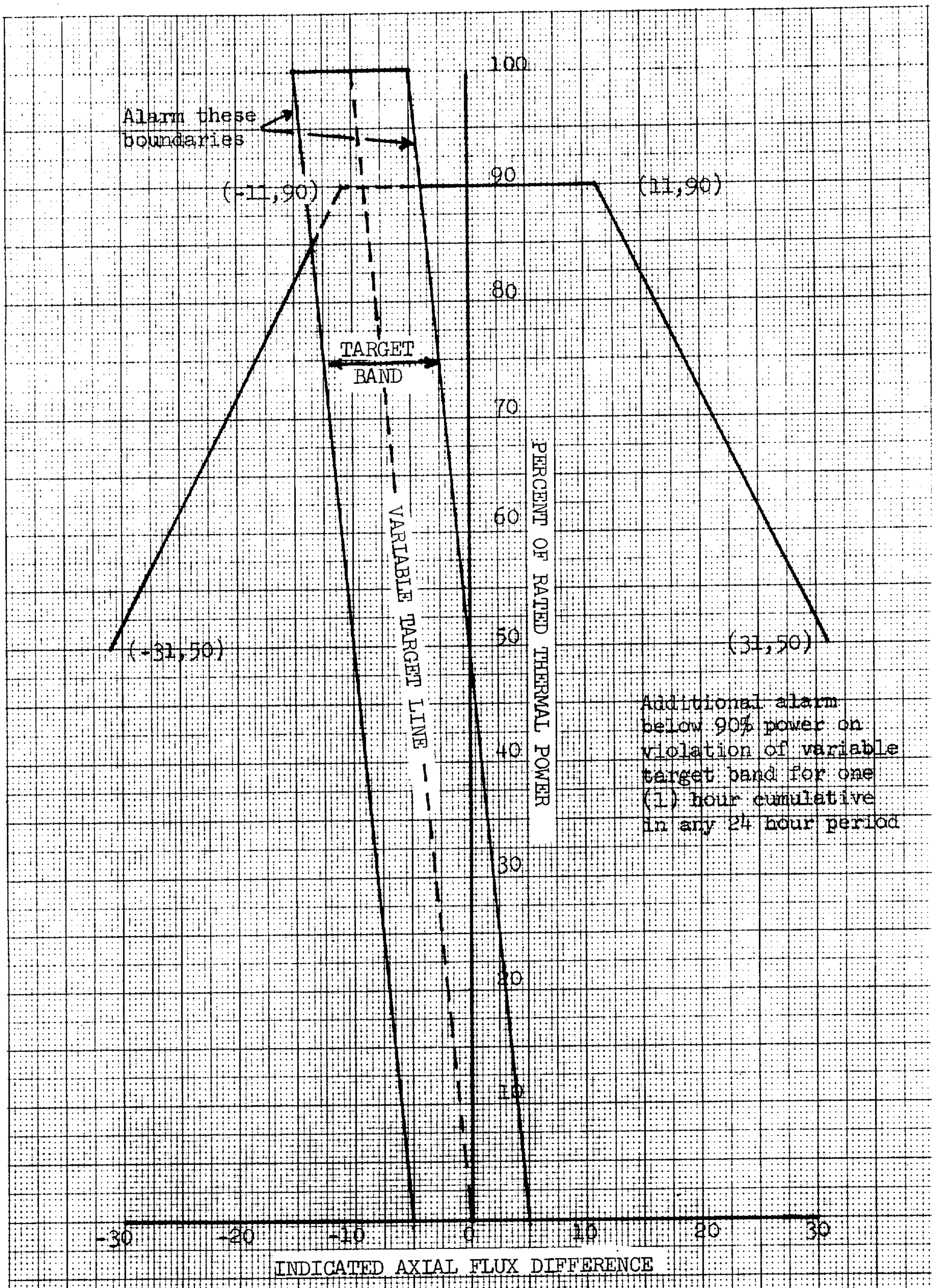
The upper limit on the quadrant tilt at which hot shutdown is required has been set so as to provide protection against excessive linear heat generation rate. The normal full power operating condition is 17.4 KW/ft and the maximum overpower condition is 20 KW/ft. The ratio of overpower to normal operation 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 generator rate can be avoided if the indicated tilt is restricted below 7 percent.

Trip shutdown reactivity is provided consistent with plant safety analysis assumptions. One percent shutdown is adequate except for steam break analysis, which requires more shutdown if the boron concentration is low. Figure TS.3.10-1 is drawn accordingly.



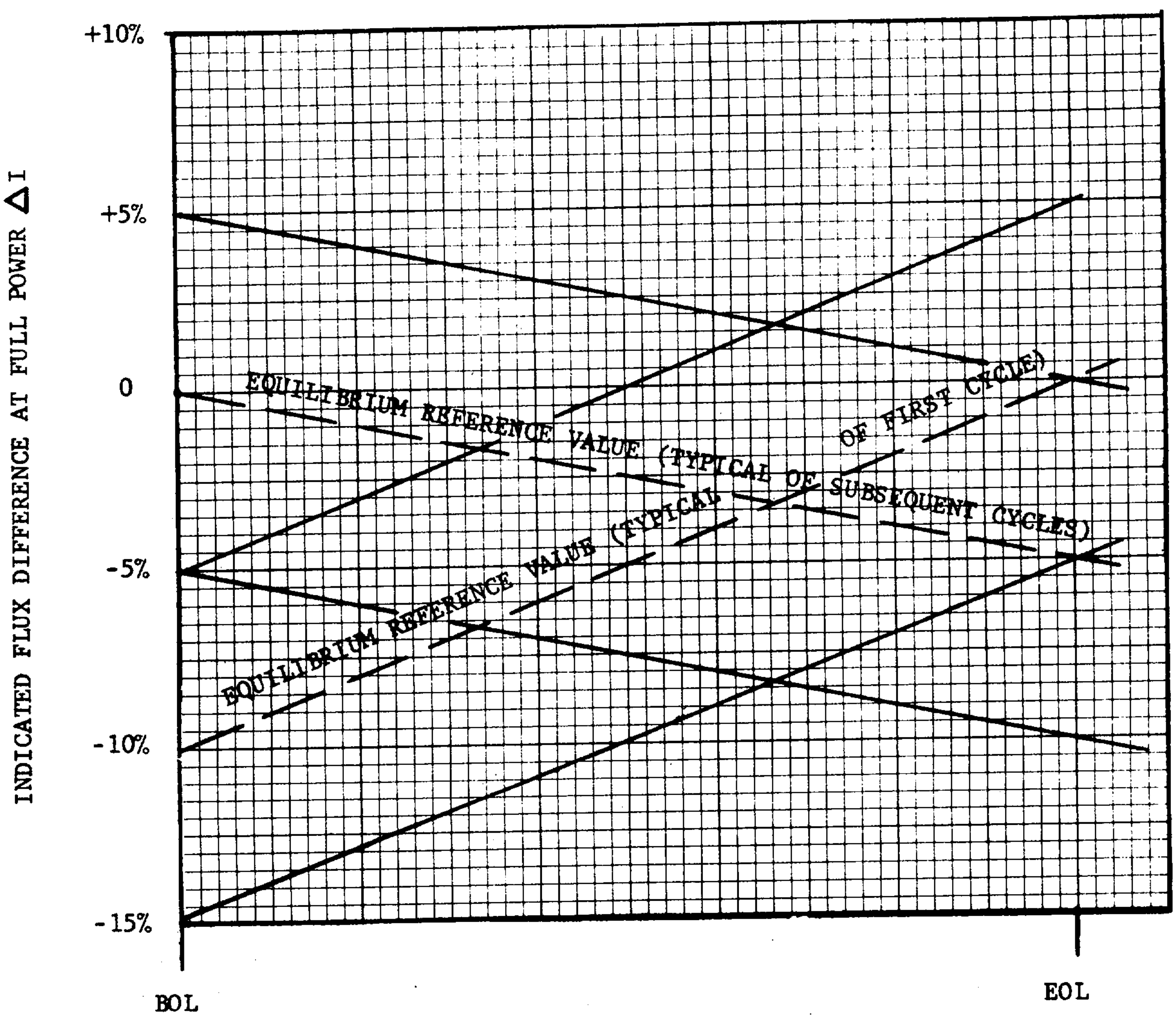
HOT CHANNEL FACTOR NORMALIZED

OPERATING ENVELOPE



FLUX DIFFERENCE CONTROL SCHEMATIC

FIGURE TS.3.10-7



PERMISSIBLE OPERATING BAND ON
INDICATED FLUX DIFFERENCE AS A
FUNCTION OF BURNUP (TYPICAL)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 16 AND 10 TO FACILITY LICENSE NOS. DPR-42 AND DPR-60

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NOS. 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By letter dated July 9, 1975 (this submittal superseded in part the licensee's letter of April 14, 1975), the Northern States Power Company (NSP) submitted a re-evaluation of the emergency core cooling system (ECCS) performance of the Prairie Island Nuclear Generating Plant (PINGP) Unit Nos. 1 and 2 in response to the Commission's Order of December 27, 1974. Supplemental information was also provided by letters from NSP to the Commission dated April 17, 1975, October 21, 1975 and January 7, 1976 and March 1, 1976. In addition, proposed Technical Specifications were submitted on February 2, 1976. The analyses submitted were performed with an acceptable evaluation model which is in conformity with 10 CFR 50, Appendix K with the exception that additional information, as indicated in a letter to NSP dated April 12, 1976, will be required regarding the analysis of the single failure criterion.

On August 27, 1976 an Order for Modification of License Nos. DPR-42 and DPR-60 was issued which added the following new requirement:

"As soon as possible, the Licensee shall submit a re-evaluation of ECCS cooling performance calculated in accordance with an approved Westinghouse Evaluation Model, with appropriate correction for upper head water temperature."

This proposed amendment was considered in the preparation of the Order. The effect of the Order on the proposed amendment is also considered in this SER.

2.0 EVALUATION

2.1 ECCS RE-ANALYSIS

The licensee submitted their loss-of-coolant accident (LOCA) analyses by letter dated July 9, 1975, that addressed large primary coolant system pipe breaks. These pipe breaks, represented by a three-break spectrum, (Moody coefficients of 1.0, 0.6 and 0.4) which was specific to PINGP, were submitted in the July 9, 1975 report. In addition, an applicable generic plant sensitivity study1/ was included in conformance with the break spectrum requirements of 10 CFR 50.46(a). The analyses submitted were performed using an acceptable model, discussed in the July 9 submittal, which is in conformance with 10 CFR 50, Appendix K. Small break analyses were previously submitted on April 14, 1975, also using an acceptable evaluation model2/.

The licensee's ECCS analyses for the worst break size, 0.4 double-ended cold leg break, which were performed assuming a total peaking factor F_Q of 2.15, resulting in a calculated peak cladding temperature of 2155°F; this is below the acceptable temperature limit of 2200°F specified in 10 CFR 50.46(b). The calculated maximum local metal/water reaction of 7.17% and total core metal/water reaction of less than 0.3% are well below the allowable limits of 17% and 1% respectively.

The licensee referenced Westinghouse topical report WCAP-8692 (non-proprietary version of WCAP 8691) "Fuel Rod Bowing," dated December 1975 as a basis for his submittal. We have completed our review of WCAP 8691 which proposed a statistical combination of the rod bow penalty with nuclear and engineering uncertainties. We have accepted this approach and have concluded that the present adjustment in the core peaking factor of 1.082 is sufficiently large to cover all deviations including the effects of fuel rod bowing. Since the analyses were presented only for two loop operation (PINGP is a two loop plant), single loop operation is prohibited as currently stated in Technical Specification 3.1.A.1.

1/ WCAP-8340 "Westinghouse Emergency Core Cooling System - Plant Sensitivity Studies" dated July 1974.

2/ WCAP-8220 WFLASH-A FORTRAN-IV, Computer Program for Simulation of Transients in a Multiloop PWR, dated June 1974.

During our review, the Commission Order of August 27, 1976 was issued, as indicated above. This Order pointed out that recent operating data gathered at the Connecticut Yankee facility has indicated that, contrary to expectation, the temperature of the water in the upper head is higher than the reactor inlet water temperature, by about some 60% of the difference between the reactor inlet and reactor outlet temperature. This increase in upper head water temperature over that used in the ECCS performance calculations would have the effect of increasing the calculated peak clad temperature.

Because this temperature difference was not accounted for in the ECCS evaluation by the licensee the Commission Order stated:

"In conformance with evaluations of the performance of the Emergency Core Cooling System (ECCS) of the facilities submitted by the Licensee on April 14, 1975, April 17, 1975, July 9, 1975, October 21, 1975, January 7, 1976 and March 1, 1976, the proposed Technical Specifications submitted by the Licensee on February 2, 1976 for the facilities limit the reactors total nuclear peaking factor (F_Q) to 2.15. The ECCS performance evaluation submitted by the Licensee was based upon previously approved ECCS evaluation model developed by the Westinghouse Electric Corporation (Westinghouse), the designer of the facilities, to conform with the requirements of the Commission's ECCS Acceptance Criteria, 10 CFR Part 50, §50.46 and Appendix K. The evaluation indicated that with a total nuclear peaking factor limited as set forth above, and with the other limits set forth in the facilities' Technical Specifications, the ECCS cooling performance for the facilities would conform with the criteria contained in 10 CFR §50.46(b) which govern calculated peak clad temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry and long term cooling."

The staff expects that, when revised calculations for the PINGP are submitted using an approved evaluation model with correct input for upper head water temperature, or assuming that the upper head water temperature equals reactor vessel outlet water temperature, such calculations will demonstrate that operation with the total nuclear peaking factor would conform to the criteria of 10 CFR §50.46(b). Such revised calculations fully conforming to the requirements of 10 CFR §50.46 are to be provided for the facilities as soon as possible.

2.2 ECCS CONTAINMENT PRESSURE EVALUATION

The ECCS containment pressure calculations for PINGP were done using the Westinghouse ECCS evaluation model. The NRC staff reviewed Westinghouse's model and issued a Status Report on October 15, 1974, which was amended November 13, 1974. We concluded that Westinghouse's containment pressure model is acceptable for ECCS evaluation. We required, however, that justification of the plant-dependent input parameters used in the analysis be submitted for our review of each plant. This information was submitted for PINGP by NSP's filing dated July 9, 1975. NSP has re-evaluated the containment net-free volume, the passive heat sinks, and operation of the containment heat removal systems with regard to the conservatism for ECCS analysis. This re-evaluation was based on measurements within the containment and on engineering drawings. Additional margin was added to the measurements by the licensee to increase the degree of conservatism of the re-evaluation. The containment heat removal systems were assumed to operate at their maximum capacities and minimum operational values for the spray water and service water temperature were assumed.

We have concluded that the plant-dependent information used by the licensee in the ECCS containment pressure analysis for PINGP is conservative and therefore the calculated containment pressure is in accordance with Appendix K to 10 CFR Part 50.

2.3 BORIC ACID CONCENTRATION DURING LONG TERM COOLING

The normal low pressure injection flow path which supplies borated water to the reactor vessel during long term core cooling is through two separate nozzles which penetrate the core barrel and discharge directly into the upper core plenum. This flow path avoids steam binding in the cold leg and its attendant deleterious effect on flow to the core which could occur should the borated water be introduced through the cold leg. In addition, it provides adequate protection against excessive boric acid buildup in the core during any cold leg break by providing adequate mixing within the core. However, this mode of injection would not provide the required mixing of the coolant being injected into the top of the core with the coolant in the lower portion of the core for a hot leg break. In such a case, the injected coolant would bypass the core, flow out through the hot leg and would not provide the needed dilution to prevent excessive buildup of boric acid in the core during long term cooling. To avoid this problem, we would require simultaneous injection directly into the core and into the cold leg by 19 hours after shutdown. This procedure has been discussed with and found to be acceptable to NSP. We conclude that implementation of this procedure will provide acceptable assurance that boron precipitation will not compromise core integrity during long term cooling following a LOCA.

3.0 PROPOSED TECHNICAL SPECIFICATIONS

Based on the above evaluation, the Technical Specifications proposed by NSP in its letter of February 2, 1976, are acceptable with the two exceptions discussed below.

The December 27, 1974 Order requires that in the event the alarms associated with the ±5% flux difference band around the largest value for operation are temporarily inoperable, the flux difference shall be logged hourly for the first 24 hours and every half-hour thereafter. Therefore the licensee's proposal to log the flux difference on an hourly basis after the associated alarms have been inoperable for periods greater than 24 hours is unacceptable. Section 3.10.B.9 of the Technical Specifications would be appropriately revised.

The licensee proposed that if the percentage quadrant tilt exceeds 2% but is less than 7% one of three optional corrective actions would be taken. One of these actions, Specification 3.10.C.1.b, is misleading. The action required would be to obtain a measurement of the core peaking factors and then apply Specifications 3.10.B.1 and 3.10.B.2 to verify that the power distribution limits have not been exceeded and/or to implement additional actions. Since the hot channel values associated with Specifications 3.10.B.1 and 3.10.B.2 are upper boundary values which would only be approached during severe transients with significant xenon oscillations, an operator might assume a false sense of security by merely verifying that hot channel values are less than those stated in Specifications 3.10.B.1 and 3.10.B.2 especially if measurements were taken at steady state conditions. That is, if core peaking factor measurements were taken at steady state conditions one would expect these values to be significantly below those stated in Specifications 3.10.B.1 and 3.10.B.2. If the measured values are approaching or are just slightly less than those of Specification 3.10.B.1 and 3.10.B.2, this could be a cause of concern during steady state operation and would require an immediate evaluation to assure that an unsafe condition does not exist. To preclude misinterpretation, Specification 3.10.C.1.b would be deleted.

During our review of the proposed changes, we found that the above modifications to the proposal were necessary to meet NRC requirements. These changes were discussed with the licensee's staff. The licensee has agreed with these changes and the changes would be incorporated into the amendments.

The above evaluation of the amendment request dated February 2, 1976 will serve to assess the proposed technical specifications which would be in effect until the revised calculations are submitted. If, as expected, the revised calculations indicate no change, these proposed technical specifications would remain in effect.

4.0 ENVIRONMENTAL CONSIDERATIONS

The Commission's staff has evaluated the potential for environmental impact associated with operation of PINGP in the proposed manner as revised by the staff. From this evaluation, the staff has determined that there would be no change in effluent types or total amounts, no change in authorized power level and no significant environmental impact attributable to the proposed action. Having made this determination, the Commission has further concluded pursuant to 10 CFR Section 51.5(c)(1) that no environmental impact statement need be prepared for this action. A Negative Declaration and supporting Environmental Impact Appraisal are being issued with these amendments to the licenses. As required by Part 51, the Negative Declaration is being filed with the Office of the Federal Register for publication.

5.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) the ECCS cooling performance conforms with the peak cladding temperature and maximum oxidation and hydrogen generation criteria of 10 CFR 50.46 subject to the re-evaluation required by the Order of August 27, 1976, and the review of additional information regarding the analysis of the single failure criterion, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (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: October 4, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

NEGATIVE DECLARATION

REGARDING PROPOSED CHANGES TO THE

TECHNICAL SPECIFICATIONS OF LICENSE NOS. DPR-42 AND DPR-60

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NOS. 1 AND 2

NORTHERN STATES POWER COMPANY

DOCKET NOS. 50-282 AND 50-306

The U. S. Nuclear Regulatory Commission (the Commission) has considered the issuance of amendments to Facility Operating License Nos. DPR-42 and DPR-60 issued to Northern States Power Company (the licensee) for operation of the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2 (PINGP) located in Goodhue County, Minnesota. These amendments would revise the Technical Specifications for PINGP to authorize operation of the facilities with modified operating limits based upon an evaluation of emergency core cooling system performance calculated in accordance with an acceptable evaluation model that conforms with the requirements of 10 CFR Part 50 of the Commission's regulations except for the analysis of the single failure criterion.

The Commission has prepared an environmental impact appraisal for the revised Technical Specifications and has concluded that an environmental impact statement for this particular action is not warranted because there will be no significant environmental impact attributable to the proposed action. The environmental impact appraisal is 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.

Dated at Bethesda, Maryland, this 4th day of October, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

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

ENVIRONMENTAL IMPACT APPRAISAL BY THE DIVISION OF OPERATING REACTORS

SUPPORTING AMENDMENT NOS. 16 AND 10 TO LICENSES DPR-42 AND DPR-60

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NOS. 1 AND 2

1. Description of the Proposed Actions

On February 2, 1976, the Northern States Power Company (NSP) proposed to change the Technical Specifications, Appendix A, appended to Facility License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2 (PINGP). The proposed changes were based on a re-evaluation of the emergency core cooling system (ECCS) in response to the Commission's Order of December 27, 1974, and are consistent with the requirements of Section 50.46 and Appendix K to 10 CFR Part 50. The proposed changes are in Section 3.10 Control Rod and Power Distribution Limits and involve adjustments which are necessary to bring reactor operation into conformity with the results of the ECCS evaluation.

The Northern States Power Company is presently licensed to operate each unit, located in Goodhue County, Minnesota, at power levels up to 1650 MWt.

2. Environmental Impacts of Proposed Actions

The proposed changes in the Technical Specifications include changes in power distribution limits, quadrant power limits, rod insertion limits and in the corresponding bases for these limits. In each case, the changes are made to bring reactor operation into conformity with the results of the ECCS evaluation. In each case there is no change in power level, no change in cooling water requirements and consequently no increase in environmental impact from radioactive effluents or thermal effluents for normal or post-accident conditions which, in turn, could lead to significant increase in radiation dose or thermal stress to the public or to biota in the environment.

The Commission's calculated releases of radioactive effluents, both gaseous and liquid, are based on expected releases to the environment and are quantified on the bases of the total quantity of nuclear fuel within the reactor. The estimates of radionuclides and release rates will not be affected by the proposed action, and since the total quantity of nuclear fuel is unchanged, no increase in the calculated release of radioactive effluents is predicted. Consequently, no increase in radiation doses to man or other biota is predicted.

3. <u>Conclusion and Basis for Negative Declaration</u>

On the basis of the foregoing analysis, it is concluded that there will be no significant environmental impact attributable to the proposed action. Having made this conclusion, the Commission has further concluded that no environmental impact statement for the proposed action need be prepared and that a negative declaration to this effect is appropriate.

Date: October 4, 1976

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. 16 and 10 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 Units 1 and 2 of the Prairie Island Nuclear Generating Plant (the facilities) located in Goodhue County, Minnesota. The amendments are effective as of their date of issuance.

The amendments revised the Technical Specifications for the facilities to authorize operation of the facilities with modified operating limits based upon an evaluation of emergency core cooling system performance calculated in accordance with an acceptable evaluation model that conforms with the requirements of 10 CFR Part 50 of the Commission's regulations with the following exception. The analysis of the single failure criterion and correction of the ECCS analysis according to the Commission Order of August 27, 1976 will be the subject of separate actions at a later date.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Notice of Proposed Issuance of Amendments to Facility Operating Licenses in connection with this action was published in the FEDERAL REGISTER on March 1, 1976 (41 F. R. 8837). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

In connection with issuance of these amendments, the Commission has issued a Negative Declaration and Environmental Impact Appraisal.

For further details with respect to this action, see (1) the application for amendments dated February 2, 1976, and earlier filings dated April 14, 1975, April 17, 1975, July 9, 1975, October 21, 1975, January 7, 1976 and March 1, 1976, (2) Amendment No. 16 to License No. DPR-42 and Amendment No. 10 to License No. DPR-60, (3) the Commission's related Safety Evaluation, (4) Commission's Negative Declaration dated October 4, 1976 (which is also being published in the FEDERAL REGISTER) and the associated Environmental Impact Appraisal. 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 The Environmental Conservation Library of the Minneapolis Public Library, 300 Nicollet Mall, Minneapolis, Minnesota 55401.

A copy of items (2), (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 4th day of October, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors