September 16, 1988

Dockets Nos. 50-282 and
50-306

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

Dear Mr. Musolf:

SUBJECT: AMENDMENTS NOS. 84 AND 77 TO FACILITY OPERATING LICENSES NOS. DPR-42 AND DPR-60: CHANGES TO POWER PEAKING FACTORS (TACS NOS. 68654 AND 68655)

The Commission has issued the enclosed Amendments Nos. 84 and 77 to Facility Operating Licenses Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant, Units Nos. 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated July 5, 1988, as supplemented by letter dated August 5, 1988.

The amendments change the Prairie Island TSs by increasing the hot channel enthalpy factor, $F_{\Delta H}$, and the total peaking factor, F_Q, to 1.70 and 2.50 respectively.

Copies of our related Safety Evaluation and the Notice of Issuance are also enclosed. This completes our work effort under TACs Nos. 68654 and 68655.

Sincerely,

Dominic C. DiIanni

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

Enclosures:
1. Amendment No. 84 to License No. DPR-42
2. Amendment No. 77 to License No. DPR-60
3. Safety Evaluation
4. Notice

cc w/enclosures:
See next page

SRXB W. Hodges 8/30/88

LA/PD31:DRSP	PM/PD31:DRSP	D/PD31:DRSP	OGC
RIngram	DDiIanni DCD	MVirgilio DCD/for M.V.	
8/26/88	8/26/88	9/1/88	/9/88

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

September 16, 1988

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Mr. D. M. Musolf, Manager
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See next page

Mr. D. M. Musolf
Northern States Power Company

Prairie Island Nuclear Generating Plant

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 84
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 July 5, 1988, as supplemented August 5, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.2 of Facility Operating License No. DPR-42 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 84, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Dominic C. DiIanni

Dominic C. DiIanni, Acting Director
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 16, 1988

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. 77
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 July 5, 1988, as supplemented August 5, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.2 of Facility Operating License No. DPR-60 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 77 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Dominic C. DiIanni

Dominic C. DiIanni, Acting Director
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 16, 1988

ATTACHMENT TO LICENSE AMENDMENT NOS. 84, 77

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

DOCKETS NOS. 50-282 AND 50-306

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
TS-x	TS-x
TS.3.10-1	TS.3.10-1
TS.3.10-2	TS.3.10-2
TS.3.10-9	TS.3.10-9

APPENDIX A TECHNICAL SPECIFICATIONS

LIST OF FIGURES

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.

Objectives

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

Specification

A. Shutdown Reactivity

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

B. Power Distribution Limits

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

$$F_Q^N \times 1.03 \times 1.05 \leq (2.50/P)K(Z)$$

$$F_{\Delta H}^N \times 1.04 \leq 1.70 \times [1 + 0.3(1-P)]$$

where the following definitions apply:

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

- Z is the core height location.

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

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

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

- 1.05 is applied to the measured F_Q^N to account for measurement uncertainty.

- 1.04 is applied to the measured $F_{\Delta H}^N$ to account for measurement uncertainty.

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

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

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

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

   $$F_Q^N \text{ (equil)} \times V(Z) \times 1.03 \times 1.05 \leq (2.50/P) \times K(Z)$$

   where V(Z) is defined Figure 3.10-7 and other terms are defined in 3.10.B.1 above.

3. (a) If either measured hot channel factor exceeds its limit specified in 3.10.B.1, reduce reactor power and the high neutron flux trip setpoint by 1% for each percent that the measured F_Q^N or by 3.33% for each percent that the measured $F_{\Delta H}^N$ exceeds the 3.10.B.1 limit. Then follow 3.10.B.3(c).

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

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

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

mechanical properties to within assumed design criteria. The ECCS analysis was performed in accordance with SECY 83-472. One calculation at the 95% probability level was performed as well as one calculation with all the required features of 10 CFR Part 50, Appendix K. The 95% probability level calculation used a peak linear heat generation rate of 14.2 kw/ft. The Appendix K calculation used a peak linear heat generation rate of 15.8 kw/ft for the F_Q limit of 2.5. Maintaining 1) peaking factors below the F_Q limit of 2.5 during all Condition I events and 2) the peak linear heat generation rate below 14.2 kw/ft at the 95% probability level assures compliance with the ECCS analysis.

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

F_Q^N is the measured Nuclear Hot Channel Factor, defined as the maximum local heat flux on the surface of a fuel rod divided by the average heat flux in the core. Heat fluxes are derived from measured neutron fluxes and fuel enrichment.

The K(Z) function shown in Figure TS.3.10-5 is a normalized function that limits F_Q axially. The K(Z) specified for the lowest six (6) feet of the core is arbitrarily flat since the lower part of the core is generally not limiting. Above that region, the K(Z) value is based on small break LOCA analyses.

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

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

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

F_Q^N (equil) is the measured limiting F_Q^N obtained at equilibrium conditions during target flux determination.

$F_{\Delta H}^N$, <u>Nuclear Enthalpy Rise Hot Channel Factor</u>, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENTS NOS. 84 AND 77 TO

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

NORTHERN STATES POWER COMPANY

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

DOCKETS NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By letter dated July 5, 1988 (Ref. 1), Northern States Power Company (NSP or the licensee) requested amendments to the Technical Specifications (TSs) appended to Facility Operating Licenses Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant, Units Nos. 1 and 2. The proposed amendments would change TS 3.10, "Control Rod and Power Distribution Limits." Specifically, the licensee proposed to increase the hot channel enthalpy rise factor, $F_{\Delta H}$, and hot channel total peaking factor, F_Q, to 1.70 and 2.50, respectively, and to delete TS Figure 3.10-8, "Acceptable Values of $F_Q(F_{\Delta H})$ and $F_{\Delta H}(F_Q)$." This proposed TS change will start with the operation of Prairie Island Unit 1 Cycle 13 which is scheduled to begin on September 28, 1988. The Unit 1 Cycle 13 will contain a full core of Westinghouse 14x14 optimized fuel assemblies (OFA). Unit 2 Cycle 12 will be operating with two-thirds of OFA fuel and one-third of Exxon TOPROD fuel until March 1989.

The proposed TS change is an extension of the previous TS change requested in April 1987 for Unit 1 Cycle 12. For Unit 1 Cycle 12 operation, the safety analyses for the anticipated operational transients and accidents other than loss-of-coolant accident (LOCA) were analyzed with $F_{\Delta H}$ and F_Q of 1.70 and 2.50, respectively, and the safety limits in Figure 2.1-1 of TS 2.1 were determined to be conservative relative to $F_{\Delta H}$ and F_Q of 1.70 and 2.50 respectively. However Figure 3.10-8 was proposed to define F_Q as a function of $F_{\Delta H}$ with the maximum allowable values of F_Q and $F_{\Delta H}$ set at 2.40 and 1.66, respectively. This was based on the large break LOCA (LBLOCA) analysis performed with the Westinghouse 1981 Emergency Core Cooling System (ECCS) evaluation model (EM). Starting with Unit 1 Cycle 13 operation, the LBLOCA analysis is performed with a new Westinghouse ECCS evaluation model described in WCAP-10924-P, "Westinghouse Large-Break LOCA Best Estimate Methodolgy" (Ref. 2). This analysis indicated that the proposed $F_{\Delta H}$ of 1.70 and F_Q of 2.50 would not violate the acceptance criteria of 10 CFR 50.46.

2.0 DISCUSSION AND EVALUATION

2.1 LBLOCA Analysis

The Prairie Island units are Westinghouse designed two-loop plants equipped with low-pressure upper plenum injection (UPI) systems as part of ECCS. The previous ECCS evaluation model assumed the UPI water fell directly into the lower plenum without interaction with the core and could therefore be treated as if the plants were cold leg injection plants. In support of the proposed TS change to increase the power peaking factors, the licensee provided a new LBLOCA analysis in Exhibit E of the July 5, 1988 submittal (Ref. 1). This analysis uses a new Westinghouse ECCS evaluation model which is developed for application to the two-loop UPI plants and is described in Westinghouse topical report WCAP-10924-P (Ref. 2). This ECCS evaluation model used a best-estimate thermal-hydraulic code WCOBRA/TRAC and the approach described in SECY 83-472 (Ref. 3). Though the methodology described in WCAP-10924 is generic to all Westinghouse-designed two-loop UPI PWR plants, the analysis uses Prairie Island plant-specific data as a lead plant to demonstrate compliance to the Appendix K requirements and the SECY 83-472 guidelines. Therefore, the results of WCAP-10924 are directly applicable to the Prairie Island units.

The NRC staff review (Ref. 4) has concluded that WCAP-10924 is acceptable for licensing application to Westinghouse two-loop UPI plants with conditions that the UPI-licensees would apply for exemptions to Items I.D.3 and I.D.5 of Appendix K to 10 CFR 50. The exemptions are necessary because Item I.D.3, which requires the use of a carry-over fraction to calculate the reflood core exit fluid flow, and Item I.D.5 setting specific requirements for refill and reflood heat transfer calculation were intended for the conventional cold leg injection plants and are not applicable to UPI plants. The licensee's letter of July 28, 1988 (Ref. 5) requested exemption to these two requirements and the exemption request has been granted (Ref. 6).

The Appendix K calculation provided in WCAP-10924, Volume 2, Revision 1, was made prior to a proper implementation of Appendix K, Item I.C.4, which prohibits a return to nucleate boiling during the blowdown phase. Therefore the results provided in Exhibit E of the licensee's submittal, which was obtained from WCAP-10924, Volume 2, Revision 1, do not comply with the Appendix K requirement and are not acceptable. By letter dated August 5, 1988 (Ref. 7), the licensee submitted a reanalysis of the Appendix K calculation using the corrected version of WCOBRA/TRAC with proper implementation of a code logic to block the return to nucleate boiling during the blowdown phase as required by Appendix K. The analysis was performed with a full core of OFA fuel and appropriate power peaking factors of 1.70 and 2.50 for $F_{\Delta H}$ and F_Q, respectively. Since Unit 1 Cycle 12 still contains one-third of ENC TOPROD fuel, a mixed core penalty of 10°F, which was based on the previous EM model calculation, was added to the resulting peak cladding temperature (PCT). The results of analysis, shown in Table 2 of the August 5, 1988 submittal, show a PCT of 2041°F, the maximum local cladding oxidation of 11.65%, and the total cladding oxidation or hydrogen generation of less than 0.3%. These results are below the acceptance criteria set forth in 10 CFR 50.46 and are, therefore, acceptable.

2.2 Safety Analysis

In Exhibit D of the July 5, 1988 submittal, the licensee provided a revised safety analysis report to support the increased $F_{\Delta H}$ and F_Q. Even though the previous safety analysis for Unit 1 Cycle 12 was performed with the same $F_{\Delta H}$ and F_Q of 1.70 and 2.50, respectively, as proposed for this amendment, the limiting transients and accidents are reanalyzed. This is because several changes have been made in the analysis methodology. These changes include the following items:

(1) An error in the surface heat transfer coefficients in the film boiling region of the analysis of the rod ejection event is corrected.

(2) The reliability factor applied to the Doppler coefficient is increased from 10% to 25%. This change reflects the increase of Doppler coefficient uncertainty and 25% reliability factor bounds the uncertainty.

(3) The low setpoint for the high neutron flux trip is increased from 25% of the rated power to 40%. This change reflects a previously approved change (Ref. 8) in the Prairie Island TSs for the neutron flux low setpoint in the power range.

(4) The fuel rod bow penalty is reduced to 2.6%. This change is a result of a previously approved (Ref. 9) reduction from 33,000 MWD/MTU to 24,000 MWD/MTU as the cutoff burnup for the maximum rod bow penalty calculation.

(5) The scram reactivity insertion curve is revised to insure conservatism in all cases. The change reflects a revised model that takes into account the drag, buoyancy and gravity forces of the rod drop. The new calculation tends to delay the scram reactivity insertion compared to the previous analysis of assuming a constant control rod velocity and is therefore acceptable.

All limiting transients and potential accidents that are affected by these changes are reanalyzed. These include fast and slow control rod withdrawal, loss of power to both reactor coolant pumps (i.e., loss of flow), locked rotor of one reactor coolant pump, loss of electric load, rod cluster control assembly (RCCA) ejection and fuel handling accidents that assumed a radial peaking factor (F_Q) 1.70 as required by Regulatory Guide 1.25. The main steamline break is not significantly affected by these changes, and the licensee has determined that it is bounded by the analysis of Unit 2 Cycle 10. The results of analyses for the control rod withdrawal transients, loss of flow and loss of load transients indicate that the minimum departure from nucleate boiling ratios (DNBR) calculated for these transients do not fall below the minimum DNBR limit of 1.17 for the WRB-1 critical heat flux correlation used and that the system pressures do not exceed the acceptance criterion of 110% of the design pressure of 2500 psia.

For the Class IV locked rotor event, the DNBR falls below the 1.17 limit. Fuel rod failure is assumed to occur when a rod experiences DNB, i.e., the DNBR is below the 1.17 limit. The number of failed fuel rods is calculated to be 1.54%, well below the acceptance criterion of less than 20% fuel failure. The peak system pressure also does not exceed the acceptance criterion of 2750 psia.

For the RCCA rod ejection events initiated from hot full power and hot zero power conditions, the results indicate that the average hot spot fuel enthalpy remains below the acceptance criterion of 280 cal/gram and that the system pressure does not exceed the acceptance criterion of 2750 psia.

The fuel handling accidents have been reanalyzed considering the proposed change in the peaking factor, F_Q from 1.65 to 1.70 pursuant to 10 CFR Part 100. The results of this reanalysis indicate that the existing fuel handling accident analysis is still bounding with the high peaking factor. In addition, our Safety Evaluation (Ref 10) at the time of plant licensing showed that with a peaking factor of 1.72, the analysis of a fuel handling accident conservatively estimated the following doses:

	Two Hour Dose at Exclusion Boundary (714 Meters)	Course of Accident Dose at Low Population Zone (2410 Meters)
Thyroid	33 rems	55 rems
Whole Body	4 rems	6 rems

These doses are within the guidance values in the NRC Standard Review Plan (NUREG-0800) for fuel handling accidents, and are well within the values of 10 CFR Part 100 guidelines which consist of 300 rem (thyroid) and 25 rem (whole body).

2.3 SUMMARY

The licensee proposed TS changes dealing with the values of $F_{\Delta H}$ and F_Q. These peaking factors would be changed from the existing requirements where the assigned values are based on a function of each other to a constant value for F_Q and $F_{\Delta H}$ of 2.50 and 1.70, respectively. The NRC staff concludes, based on the above evaluation, that the proposed changes to the peaking factors are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

An Environmental Assessment and Finding of No Significant Impact has issued for these amendments (53 FR 35134, September 9, 1988).

4.0 CONCLUSION

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

5.0 REFERENCES

1. Letter from David Musolf (NSP) to Director of Nuclear Reactor Regulation, USNRC, "Prairie Island Nuclear Generating Plant, Docket Nos. 50-282/50-306, License Nos. DPR-42/DPR-60, License Amendment Request Dated July 5, 1988, Best Estimate LOCA Analysis," July 5, 1988.

2. WCAP-10924-P, "Westinghouse Large-Break LOCA Best Estimate Methodology," Volume 1, June 1986, Volume 2, Revision 1, April 1988.

3. Information Report from William J. Dircks to the Commissioners, "Emergency Core Cooling System Analysis Methods, "SECY 83-472, November 17, 1983.

4. Letter from Ashok C. Thadani to W. J. Johnson (Westinghouse), "Acceptance for Referencing of Licensing Topical Report, WCAP-10924, Westinghouse Large Break LOCA Best Estimate Methodology," August 29, 1988.

5. Letter from David Musolf (NSP) to Director of Nuclear Reactor Regulation, USNRC, "Request for Exemption to Selected 10 CFR 50, Appendix K Requirements," July 28, 1988.

6. Letter from D. C. DiIanni to D. W. Musolf (NSP), "Acceptance of Request for Exemption to Selected 10 CFR 50, Appendix K Requirements," September 16, 1988

7. Letter from David Musolf (NSP) to Director of Nuclear Reactor Regulation, USNRC, "Supplemental Information for License Amendment Request Dated July 5, 1988, Best Estimate LOCA Analysis", August 5, 1988.

8. Letter from D. C. DiIanni to D. W. Musolf (NSP), "Amendment Nos. 83 and 76 to Facility Operating License Nos. DPR-42 and DRP-60: High Flux, Power Range (Low Setpoint) (TACS Nos. 66865 and 66866)," May 31, 1988.

9. Letter from Carl Berlinger (USNRC) to E. P. Rahe, Jr. (Westinghouse), "Request for Reduction in Fuel Assembly Burnup Limit for Calculation of Maximum Rod Bow Penalty," June 18, 1986.

10. Safety Evaluation of the Prairie Island Nuclear Generating Plant Units 1 and 2 dated September 28, 1972, Page 15-4 and Table 15.1-1.

Principal Contributors: Y. Hsii
James A. Martin

Date: September 16, 1988

UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY

DOCKETS NOS. 50-282 AND 50-306

NOTICE OF ISSUANCE OF AMENDMENTS TO

FACILITY OPERATING LICENSES

The United States Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 84 and 77 to Facility Operating Licenses Nos. DPR-42 and DPR-60, issued to the Northern States Power Company (the licensee), which revised the Technical Specifications (TSs) for operation of the Prairie Island Nuclear Generating Plant, Units Nos. 1 and 2, located in Goodhue County, Minnesota. The amendments are effective as of the date of issuance.

The amendments change the TSs by increasing the hot channel enthalpy factor, $F_{\Delta H}$, and the total peaking factor, F_Q, to 1.70 and 2.50, respectively.

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.

Notice of Consideration of Issuance of Amendments to Facility Operating Licenses and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on July 29, 1988 (53 FR 28737). No request for hearing or petition to intervene was filed following this notice.

Also in connection with this action, the Commission prepared an Environmental Assessment and Finding of No Significant Impact which was published in the FEDERAL REGISTER on September 9, 1988, at 53 FR 35134.

The amendments also involved the granting of an exemption to the Commission's regulations in 10 CFR Part 50, Section 50.46, and Appendix K.

For further details with respect to this action, see (1) the application for amendments dated July 5, 1988, as supplemented August 5, 1988, (2) Amendments Nos. 84 and 77 to Licenses Nos. DPR-42 and DPR-60, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, Gelman Building, 2120 L Street, NW., Washington, DC 20555, and at the Minneapolis Public Library, Technology and Science Department, 300 Nicollet Mall, Minneaplis, Minnesota 55401. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Director, Division of Reactor Projects - III, IV, V and Special Projects.

Dated at Rockville, Maryland, this 16th day of September 1988.

FOR THE NUCLEAR REGULATORY COMMISSION

Dominic C. DiIanni

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