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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 19 TO PROVISIONAL OPERATING LICENSE NO. DPR-18

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

Introduction

By application dated January 6, 1978, as supplemented by letters dated January 10, March 27, April 6, April 17, and April 25, 1978, Rochester Gas and Electric Corporation (the licensee) requested authorization to operate the R. E. Ginna Nuclear Power Station in Cycle 8 with reload fuel supplied by Exxon Nuclear Company, Inc., and requested a change to the Technical Specifications involving power distribution control limits.

Discussion

The R. E. Ginna Nuclear Power Station has operated seven fuel cycles with fuel supplied by Westinghouse Corporation. Cycle 8 will involve the first use of fuel from a different vendor, Exxon Nuclear Company, Inc. (ENC). The loading for Cycle 8 will consist of 32 new ENC fuel assemblies loaded at the periphery of the core and 89 exposed Westinghouse assemblies scatter loaded in the center of the core. All assemblies are of similar design with the ENC assemblies designed to be compatible with the other fuel assemblies. Reactor power level, core average linear heat rate and primary coolant system temperature and pressure for Cycle 8 will remain the same as for the previous cycle.

The licensee has stated that all technical specification limits for the previous cycle are applicable to Cycle 8, with the exception of one limit involving power distribution control. The licensee also proposed a change to the bases of the specifications involving power distribution control to reflect a revised methodology used in the reactor physics analyses for Cycle 8.

The licensee's analyses for Cycle 8 also include the first use of ENC analytical methods to verify the acceptability of Ginna operating limitations and safety margins.

The staff evaluation which follows, addresses the acceptability of the use of the ENC assemblies in Cycle 8 and the acceptability of the proposed changes in Technical Specification. The evaluation includes the staff's review of nuclear, thermal-hydraulic and accident analyses for Cycle 8 operation.

Evaluation

1. Design of the New Fuel

The new fuel assemblies for the core periphery were designed by Exxon Nuclear Corporation to be compatible with the Westinghouse depleted fuel assemblies that are to remain in the Ginna core.

The Exxon fuel design is similar to the Westinghouse fuel bundle design (References 1 and 2).

The Exxon fuel design criteria and fuel design calculations are discussed in Exxon reports submitted with the application for Fuel Cycle 8 operation. Those aspects of the fuel design important to safety have been reviewed by the staff and found acceptable. Those aspects are: (1) the fuel performance during LOCA: (2) fuel clad collapse and fuel densification: (3) fretting wear; and (4) the effect of fuel rod bowing on the departure from nucleate boiling ratio (DNBR).

The GAPEX code (Reference 3) was used to calculate stored energy for LOCA calculations. GAPEX has been reviewed and approved by the staff for fuel temperature and internal pressure calculations in PWR fuel (Reference 4).

Reference 1 presents calculations which show that the cladding will not collapse during Cycle 8. These calculations utilize the RODEX and COLAPX codes. The RODEX code (Reference 5) calculates the cladding temperature and fuel rod internal pressure while COLAPX (Reference 7) calculates the collapse time using the RODEX input. COLAPX has been reviewed by the staff and found acceptable for cladding collapse calculations. RODEX has not been approved by the staff but the models in RODEX affecting clad temperature and internal pressure are similar to those in the GAPEX code, which has been approved. Moreover, since the clad collapse analyses for the Westinghouse fuel does not predict collapse during Cycle 8, and since the cladding for the Exxon fuel is thicker than that of the Westinghouse fuel (Reference 2) which makes it more resistent to clad collapse, we have concluded, with reasonable assurance, that the results of the RODEX analysis are acceptable.

Exxon tests to determine the magnitude of fretting at the fuel rod axial spacer contact points due to flow induced vibration revealed no active fretting corrosion and negligible difference in wear observed between 500, 1000, and 1500 hours. Based on these test results and the larger diameter - thicker clad of the Exxon fuel rods in the 14 x 14 fuel assemblies for Ginna and therefore greater stiffness, we have concluded that fuel rod integrity with respect to flow induced vibration and fretting wear is acceptable.

The effect of fuel rod bowing on Departure from Nucleate Boiling Ratio (DNBR) has been a subject of continuing discussion between the staff and Exxon. An Exxon analysis considered the fuel rod bowing penalties for the most limiting transients and attempted to show that there is sufficient margin to offset the calculated penalties. These results are presented in Reference 2. The staff has concluded that these analyses are not completely acceptable because the heat flux and pressure used to calculate the bowing penalties were for minimum DNBR conditions and do not represent the worst conditions for calculating the rod bowing penalties. However, Reference 2 shows that there is an 8.5 percent margin to the safety limit which offsets this nonconservatism. On this basis, we have concluded that there is adequate thermal margin to assure safe plant operation without violating the minimum DNBR safety limit.

Based on successful irradiation experience of Exxon fuel assemblies in other PWR cores and the analyses which have been done for Ginna Fuel Cycle 8, we have concluded that the Exxon fuel assemblies for Cycle 8 will perform in a safe and acceptable manner. The licensee has agreed (RG&E telecon 4/14/78) to submit plans for inspection of the Exxon fuel assemblies to NRC for concurrence at least 90 days prior to the end of Fuel Cycle 8 to enable additional NRC review of the fuel prior to its use in Cycle 9.

2. Thermal Hydraulic Design

The new Exxon fuel assemblies are designed to have thermal hydraulic characteristics equivalent to those of the existing fuel. Therefore, there will not be any major differences in the thermal hydraulic behavior of the core.

The licensee has shown that at 118 percent of rated power, the calculated minimum DNBR is 1.47. The corresponding value for the Westinghouse fuel assemblies is 1.43. The fuel and cladding temperature analysis uses Exxon calculational methods (Reference 7), assuming maximum power peaking and engineering tolerances. The calculated maximum fuel and cladding temperatures are well below the design limits. We, therefore, conclude that the Exxon fuel assemblies are compatible with the Westinghouse fuel assemblies in the Ginna core and that the thermal hydraulic criteria will not be exceeded during plant [illegible]

3. Nuclear Design

The Fuel Cycle 8 loading will consist of 89 fuel assemblies with burnups ranging from 7,178 MWD/MTU to 23,813 MWD/MTU and 32 fresh ENC fuel assemblies.

The licensee has specified new values for the target flux difference. They are between +5.0 and -7.5% for the beginning of cycle life and between +2.0 and -7.5% for the end of cycle life. For the intermediate times the values are obtained by linear interpolation. The licensee has compared the neutronic characteristics of the Cycle 8 and Cycle 7 cores and concluded that they are approximately the same. The reactivity coefficients of the Cycle 8 core are bounded by the coefficients used in the safety analyses and we have concluded that the coefficients are acceptable.

Justification of the assumed total rod worth uncertainty of 10% used in the determination of shutdown margin has not been presented. Confirmatory tests are therefore included in the startup physics tests for fuel Cycle 8.

The physics startup test program for Ginna Cycle 8 presented in the March 27, 1978 submittal (Reference 2), was reviewed with the licensee. Several changes to the rod worth and power coefficient measurements were made. These changes are documented in the Reference 17 submittal. As part of this test program, control rod reactivity worth will be measured for banks D, C, B and A in order to verify that adequate shutdown margin is available. If any one bank worth differs from the predicted value by more than 15% or the sum of the worths of these banks differs from the predicted value by more than 10%, the first shutdown bank should be measured. If the sum of the five measured banks differs from the predicted value by more than 10%, additional shutdown bank measurements will be performed to verify the technical specification shutdown margin.

We have concluded that the total physics startup test program as modified is acceptable. However, there are areas in the licensee's safety analysis that warrant verification in the physics startup test program. Therefore, a summary report as described in the March 27th submittal (Reference 2) will be submitted to the NRC. The licensee has agreed to submit the report within 45 days of completion of the program.

4. Steady State and Load Follow Operation

Compliance with F_Q and $F_{\Delta H}$ limiting conditions for operation is ensured by adherence to previously approved constant axial offset control strategy and core monitoring with incore and excore flux monitors. Incore monitoring is achieved using travelling fission chambers. Data from the fission chambers and calculated coefficients

(Reference 9) are processed by the computer code INCORE to obtain power distribution maps. Extensive comparisons of predicted and measured core power distributions have been performed by Exxon for 3 and 4 loop cores. In general, the results of these comparisons are favorable. However, R. E. Ginna is a two loop plant and there is only a single set of measured and calculated power distributions for R. E. Ginna, Cycle 7, at hot full power, 1000 MWD/MTU. The results of this comparison show good agreement between measurement and calculation and add credibility to the licensee's assertion that an F_Q uncertainty factor of 5% is appropriate for Cycle 8. However, additional data will be obtained during the fuel cycle 8 startup physics tests.

5. Safety Analyses

The licensee has analyzed the anticipated operating occurrences and postulated accidents using the plant transient simulator code PTSPWR (Reference 15). The results of these analyses are presented in Reference 14. Our review of this code has progressed sufficently to allow us to conclude that analyses using PTSPWR provide acceptable margins to peak linear heat generation rate and departure from nucleate boiling design limits. The reactivity coefficients assumed in the safety analyses are to be confirmed during the physics startup tests.

a. Steam Line Break Analyses

The Steam Line Break (SLB) accident analysis (Reference 14) is of particular concern. SLB analysis methods have not been generically approved. The licensee asserts that should a large SLB occur the plant would return to criticality, reaching a peak average core power of 22% of rated power at approximately 90 sec after accident initiation. The minimum DNBR at this condition, using the Macbeth critical heat flux correlation, would be 1.58. Even if DNB were to occur during a steam line break accident, DNB would be restricted to a small region of the core in the vicinity of the assumed stuck rod. It is noted that DNB anywhere in the core is unlikely if all control rods scram as expected. Of the fuel rods which might experience DNB in the vicinity of the stuck rod, some fraction would release their fission gas inventory. The fission gas would have to be transported to the secondary side of the coolant system (primary to secondary steam generator leakage) in order to represent a potential hazard. The potential release to the atmosphere would be significantly less than 10 CFR Part 100 limits. Accordingly, we have concluded that the consequences of a steam line break are acceptible.

b. ECCS Analysis

The licensee has submitted ECCS performance analyses for the Westinghouse (Reference 19) and new EXC fuels (Reference 1). The Westinghouse analysis was [illegible] for Cycle 7 fuel which the staff believes is a conservative [illegible] for the Westinghouse fuel during Cycle 8. The ECCS analysis was performed for Cycle [illegible] using the [illegible] ECCS [illegible] model (Reference 7) which is described in [illegible]. The applicability of the [illegible] to two-loop Westinghouse PWR plants was evaluated by ENC in Reference 10. The ENC evaluation model has been reviewed and approved conditionally by the NRC (Reference 16). The staff has recently considered whether the Westinghouse generic evaluation adequately represented the flow characteristics of the Westinghouse two loop units. The generic evaluation model assumes that all safety injection water is introduced directly into the lower plenum. For the two loop units, the safety injection water is injected into the upper plenum. Thus, the staff was concerned that the Westinghouse model did not consider interaction between UPI water and steam flow. (References 11 and 12). After plant specific submittals by the licensees operating two loop plants were reviewed, the staff concluded that the calculations provided by the licensees (with certain modifications to the staff's model) are acceptable as an interim basis for continued safe operation of the Westinghouse two loop plants, while long term efforts continue for developing a model specifically treating UPI. For the Ginna plant the calculations which specifically considered UPI using the modified version of the staff model, resulted in a change of only 15°F from those using the generic model in which the UPI-core interaction was not specifically considered (Reference 20). In the interim, before these models are developed, the licensee has provided a modification to the current Westinghouse model which accounts for UPI-core interaction (Reference 13). It was demonstrated that the modification resulted in the increase of peak clad temperature by 15°F. Since for the Ginna plant both ENC WREM-II and Westinghouse models predict similar PCT's (1922°F for ENC WREM-II and 1957°F for Westinghouse) it can be expected that the UPI modification, when applied to the ENC WREM-II model, would allow about the same increase in PCT. The licensee has drawn a similar conclusion and agreed to submit within 30 days, calculational results to confirm the validity of this conclusion. (Reference 21).

The ECCS analyses have been performed with the upper head fluid temperature equal to the fluid outlet (hot leg) temperature and assuming 10 percent of steam generator tubes plugged. The analyses included a spectrum of breaks which consisted of guillotine double ended cold leg (DEGCL) breaks with discharge coefficients of 1.0, 0.6 and 0.4 and split breaks with break areas of 8.25, 4.9 and 3.30 ft^2. No small break analysis was performed. The licensee has demonstrated, by showing analogy between the present analysis and the analyses performed previously for other plants, that the small break LOCA is not limiting (Reference 2). The critical break size was determined to be DEGCL with C_D=0.4.

The staff has concluded that although the Westinghouse and Exxon two-loop generic-evaluation models should be changed to consider upper plenum injection [illegible] the plant is modified [illegible] analyses at the [illegible] conditions applicable to the Ginna plant [illegible] the inaccuracy of disregarding upper plenum injection interaction on PCT [illegible] conditions will not be significant (less than 20°F PCT). Therefore, the staff believes that, for the limited range to which the models are applied for conditions at the Ginna plant, the models do not deviate from the requirements of 10 CFR 50 Appendix K item I.D.3. and the calculations are acceptable.

On March 23, 1978 Westinghouse informed the NRC that an error in the West-ECCS evaluation model had been found which had resulted in incorrectly calculated peak clad temperatures in all LOCA analyses previously submitted by their customers. For several plants preliminary estimates indicated that they would not meet the 2200°F limit of 10 CFR 50.46 at their present maximum overall peaking factor limits. Westinghouse and several of their customers met with the NRC staff on March 29, 1978 in Bethesda to discuss the error and its impact on specific plant analyses. Subsequent to that meeting, Westinghouse provided information through the licensees of operating reactors to justify continued operation at the interim peaking factor Technical Specification limits proposed by the NRC staff on April 3, 1978.

On April 17, 1978 (Reference 19) RG&E submitted a letter indicating that continued operation at their present Technical Specification limit of 2.32 (total peaking factor) was justified on the basis of additional generic Westinghouse analyses. Westinghouse had determined that the impact of correcting the error on the peak cladding temperature for the RE Ginna plant was significant but within the presently existing margin (228°F) to the 2200 F° acceptance criteria limit. The NRC Staff confirmed the conservatism of that and all other plant evaluations and on April 18, 1978 published a Safety Evaluation Report (Reference, attachment to Exemption). Since the Westinghouse and ENC fuels were analyzed using the respective Westinghouse and ENC evaluation models, and since there is no zirconium-water error in the ENC calculational model, the error in zirconium-water reaction in the Westinghouse calculational model has no effect on the Exxon calculations. The Zirconium-water reaction error in the Westinghouse model is the subject of an exemption request by the licensee dated April 26, 1978, (Reference 21) and a separate exemption action by NRC.

6. <u>Technical Specification Changes</u>

The proposed addition to the Technical Specifications restricts the permissable range of the target flux difference i.e. the ratio of the flux in the top half of the core minus the flux in the lower half of of the core to the total flux measured at 100% power, equilibrium conditions. The addition, Technical Specification 3.10.2.7, assures that axial power distributions realized in the reactor will be no more limiting with respect to linear heat generation rate than the axial power distributions used by Exxon to analytically confirm (Reference 13) that, limiting values of linear heat generation vs core height, Technical Specification 3.10.2.2, will not be violated. The restriction has been reviewed and approved on a generic basis and has [illegible] in the Technical Specifications of PWR's [illegible]

The change to Technical Specification 3.10.1.4 and the addition of specification 3.10.1.6 are required to permit the physics testing program as discussed in part 3 of our evaluation. The change and the addition are in accordance with the Standard Technical Specifications for Westinghouse PWR's which we have already reviewed and approved.

The changes to the basis of the Technical Specification related to core power distribution are in accordance with the Standard Technical Specification which we have approved and are therefore acceptable also.

Environmental Consideration

We have determined that the amendment does 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 amendment involves 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 this amendment.

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 Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: May 1, 1978

REFERENCES

(1) Letter from LeBoef, Lamb, Leiby and MacRae (Counsel for Rochester Gas and Electric Corporation) to E. G. Case (NRC), dated January 6, 1978.

(2) Letter from L. D. White, Jr. (Rochester Gas and Electric Corporation) to D. L. Ziemann (NRC), dated March 27, 1978.

(3) XN-73-25, "GAPEX: A Computer Program for Predicting Pellet-to-Cladding Heat Transfer Coefficients", June 1975.

(4) USNRC Report, "Technical Report on Densification of Exxon Nuclear PWR Fuel", February 27, 1975.

(5) XN-76-8(P), "RODEX: Fuel Rod Design Evaluation Code", February 1977.

(6) XN-72-23, "Clad Collapse Calculational Procedure", November 1, 1972.

(7) XN-NF-77-58, "ECCS analysis for the R. E. Ginna Reactor with ENC WREM-II PWR Evaluation Model", December 1977.

(8) XN-75-41, "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model", Vol I through III, July-August 1975 and Supplements 1 through 7, August-November 1975.

(9) XN-76-27, "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model Update ENC WREM-II", July 1976 and Supplements 1 and 2, September-November 1976.

(10) XN-NF-77-25, "Exxon Nuclear Company ECCS Evaluation of a 2-loop Westinghouse PWR with Dry Containment using the ENC WREM-II ECCS Model - Large Break Example Problem," August 1977.

(11) Letter from E. G. Case (NRC) to L. D. White, Jr. (Rochester Gas and Electric Corporation), dated December 16, 1977.

(12) Letter RG&E to NRC, Development of a New Model to Account for Upper Plenum Injection, dated March 5, 1978.

(13) Letter from L. D. Amish (Rochester Gas and Electric Corporation) to A. Schwencer (NRC), dated February 1978.

(14) XN-NF-77-40, "Plant Transient Analysis for the R. E. Ginna Unit 1 Nuclear Power Plant", November 1977.

(15) XN-74-5, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTSPWR)," Revision 1, May 1975.

(16) USNRC Technical Report Evaluation [illegible] Exxon Nuclear Company Report XN-[illegible] April 1978.

(17) Letter from L. D. White, Jr. (Rochester Gas and Electric Corporation) to D. L. Ziemann (NRC), dated April 6, 1978.

(18) Exxon Nuclear Power Distribution Control for Pressurized Water Reactors XN-76-40, September 1976.

(19) Letter from L. D. White, Jr., (RG&E) to A. Schwencer (NRC) dated April 7, 1977.

(20) Letter to RG&E dated April 28, 1978 transmitting staff SER of UPI model evaluation.

(21) Letter from RG&E to NRC dated April 25, 1978, related to ENC UPI calculations.

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

DO NOT REMOVE

Posted
Am 35 to
DPR-42

April 20, 1979

Docket Nos. 50-282
and 50-306

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

Dear Mr. Mayer:

In response to your submittal dated September 8, 1978 and application dated December 29, 1978, supplemented on January 23 and March 30, 1979, the Commission has issued the enclosed Amendment Nos. 35 and 29 to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2 respectively.

The amendments change the common station Technical Specifications for the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2 in connection with the refueling of Units 1 and 2 and incorporate changes to the Appendix A Technical Specifications to support operation in Cycle 5 with reload fuel by Exxon Nuclear Company.

During our review of your proposed amendments we found that certain modifications were necessary to meet our requirements. Your staff has agreed to these modifications and they have been incorporated in these amendments.

You are required to submit the Unit 2 SAR prior to the next Unit 2 reload to confirm that the Technical Specifications will not change for Unit 2. The Exxon Nuclear Company does not have an approved Rod Bow Topical Report. Therefore we have included a penalty factor in the Technical Specifications until such time as the Topical Report is approved.

The requirements of the NRC Order for Modification of License of Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2 dated May 18, 1978 have been satisfied by your submittal dated February 21, 1979.

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

Sincerely,

A. Schwencer

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

Enclosures:
1. Amendment No. 35 to DPR-42
2. Amendment No. 29 to DPR-60
3. Safety Evaluation
4. Notice of Issuance

cc: w/enclosures
See next page

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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. 35
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 December 29, 1978 and supplemented on January 23 and March 30, 1979, 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.

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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. 35, 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

A. Schwencer

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 20, 1979

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. 29
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 December 29, 1978 and supplemented on January 23 and March 30, 1979, 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. 29, 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

A. Schwencer

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: April 20, 1979

ATTACHMENT TO LICENSE AMENDMENT NOS.35 AND 29

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

DOCKET NOS. 50-282 AND 50-306

Replace the following pages of the Technical Specifications contained in Appendix A of the above-indicated licenses with the attached pages bearing the same numbers, except as otherwise indicated. The changed areas on the revised pages are reflected by a marginal line.

Remove	Insert
TS-iv	TS-iv
TS 1-6	TS 1-6
TS 3.1-17	TS 3.1-17
TS 3.1-18	TS 3.1-18
TS 3.10-1	TS 3.10-1
	TS 3.10-1A
TS 3.10-2	TS 3.10-2
TS 3.10-7A	TS 3.10-7A
TS 3.10-8	TS 3.10-8
TS 3.10-9	TS 3.10-9
Figure TS 3.10-5	Figure TS 3.10-5
Figure TS 3.10-6	Figure TS 3.10-6
	Figure TS 3.10-8
TS 5.3-1	TS 5.3-1
TS 5.3-2 - DELETED	

APPENDIX A TECHNICAL SPECIFICATIONS

LIST OF FIGURES

TS FIGURE	TITLE
2.1-1	Safety Limits, Reactor Core, Thermal and Hydraulic Two Loop Operation
3.1-1	Unit 1 and Unit 2 Reactor Coolant System Heatup Limitations
3.1-2	Unit 1 and Unit 2 Reactor Coolant System Cooldown Limitations
3.1-3	Effect of Fluence and Copper Content on Shift of RT_{NDT} for Reactor Vessel Steels Exposed to 550^{o} Temperature
3.1-4	Fast Neutron Fluence (E > 1 MeV) as a Function of Full Power Service Life
3.10-1	Required Shutdown Reactivity Vs Reactor Boron Concentration
3.10-2	Control Bank Insertion Limits
3.10-3	Insertion Limits 100 Step Overlap with One Bottomed Rod
3.10-4	Insertion Limits 100 Step Overlap with One Inoperable Rod
3.10-5	Hot Channel Factor Normalized Operating Envelope For F_Q = 2.21
3.10-6	Deviation from Target Flux Difference as a Function of Thermal Power
3.10-7	Rod Bow Penalty (RBP) Fraction Versus Region Average Burnup
3.10-8	V(Z) as a function of core height
4.4-1	Shield Building Design In-Leakage Rate
4.10-1	Prairie Island Nuclear Generating Plant Radiation Environmental Monitoring Program (Sample Location Map)
4.10-2	Prairie Island Nuclear Generating Plant Radiation Environmental Monitoring Program (Sample Location Map)
6.1-1	NSP Corporate Organizational Relationship to On-site Operating Organization
6.1-2	Prairie Island Nuclear Generating Plant Functional Organization for On-site Operating Group

3. Refueling Shutdown

A reactor is in the refueling shutdown condition when a refueling operation is scheduled, the reactor is subcritical by at least 10% $\Delta k/k$ and the reactor coolant average temperature is less than 140°F.

Q. Thermal Power

Thermal power of a unit is the total heat transferred from the reactor core to the coolant.

R. Physics Tests

Physics tests are those conducted to measure fundamental characteristics of the core and related instrumentation. Physics tests are conducted such that the core power is sufficiently reduced to allow for the perturbation due to the test and therefore avoid exceeding power distribution limits in Specification 3.10.B.

Low power physics tests are run at reactor powers less than 5% of rated power.

S. Startup Operation

The process of heating up a reactor above 200°F, making it critical, and bringing it up to power operation.

T. Fire Suppression Water System

The fire suppression water system consists of: Water sources; pumps; and distribution piping with associated sectionalizing isolation valves. Such valves include yard hydrant valves, and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe, or spray system riser.

F. MINIMUM CONDITIONS FOR CRITICALITY

Specification

1. The reactor shall be made critical only at or above the coolant temperature at which the following reactivity coefficient is negative and remains negative for any coolant temperature increase (except during low power physics tests):

   (a) Moderator temperature coefficient for a reactor loaded with Westinghouse fuel only.

   (b) Isothermal temperature coefficient for a reactor either full or partially loaded with Exxon fuel.

2. The reactor shall not be brought to a critical condition until the reactor coolant temperature is higher than that defined by the criticality limit line shown in Figure TS.3.1-1.

3. When the reactor coolant temperature is below the minimum temperature as specified in 1. above, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to reactor coolant depressurization.

Basis

At the beginning of a fuel cycle the moderator temperature coefficient has its most positive or least negative value. As the boron concentration is reduced throughout the fuel cycle, the moderator temperature coefficient becomes more negative. The safety analyses conducted for Prairie Island units with Westinghouse fuel assumed a non positive moderator temperature coefficient. The isothermal temperature coefficient is defined as the reactivity change associated with a unit change in the moderator and fuel temperatures. Essentially, the isothermal temperature coefficient is the sum of the moderator and fuel temperature coefficients. This coefficient is measured directly during startup physics testing, whereas the moderator temperature coefficient is an inferred parameter determined by subtracting the predicted fuel temperature coefficient from the experimentally determined isothermal temperature coefficient.

For extended optimum fuel burnup it is necessary to either load the reactor with burnable poisons or increase the boron concentration in the reactor coolant system. If the latter approach is emphasized, it is possible that a positive moderator temperature coefficient could exist at beginning of cycle (BOC). For cycles with Exxon fuel, safety analyses are conducted assuming a positive moderator temperature coefficient. These analyses predict the isothermal coefficient to be negative at an all rods out, hot zero power condition. Other conditions, e.g., higher power or partial rod insertion would cause the isothermal coefficient to have a more negative value. These analyses demonstrate that applicable criteria in the NRC Standard Review Plan (NUREG 75/087) are met.

Physics measurements and analyses are conducted during the reload startup test program to (1) verify that the plant will operate within safety analyses assumptions and (2) establish operational procedures to ensure safety analyses assumptions are met. The 3.1.F.1 requirements are waived during low power physics tests to permit measurement of reactor temperature coefficient and other physics design parameters of interest. Special operating precautions will be taken during these physics tests. In addition, the strong negative Doppler coefficient[(1)] and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The requirement that the reactor is not to be made critical except as specified in Figure TS.3.1-1 provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization whenever the reactor vessel is in the nil ductility temperature range. Heatup to this temperature will be accomplished by operating the reactor coolant pumps and by the pressurizer heaters. The pressurizer heater and associated power cables have been sized for continuous operation at full heater power. The shutdown margin in Specification 3.10 precludes the possibility of accidental criticality as a result of an increase of moderator temperature or a decrease of coolant pressure.[(2)]

References:

(1) FSAR Figure 3.2-10
(2) FSAR Table 3.2-1

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability

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

Objective

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

Specification

A. Shutdown Reactivity

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

B. Power Distribution Limits

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

$$F_Q^N (Z) \leq (2.145/P) \times K(Z) \quad \text{for } P > 0.5$$

$$F_Q^N (Z) \leq (4.29/P) \times K(Z) \quad \text{for } P \leq 0.5$$

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

b. $F_Q^N(Z)$ shall be measured at equilibrium conditions according to one of the following conditions, whichever occurs first;

(1) At the time of target flux difference determination, or

(2) At least once per 31 effective full-power days, or

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

and must meet the following limit:

$$F_Q^N(Z) \leq (2.145/P^1) \times \left[K(Z)/V(Z)\right] \left[1-2.35 \times 10^{-6}(BU' - 2.8 \times 10^3)\right]^{*}$$

for $P^1 > 0.50$

† The (1-RBP(BU)) multiplier is only applicable for Westinghouse Fuel.

* The $[1-2.35 \times 10^{-6}(BU' - 2.8 \times 10^3)]$ multiplier is only applicable with Exxon fuel in the core. BU' in this expression is the average Exxon fuel exposure.

1. c. In Specification 3.10.B.1, the following definitions apply:

   (1) P is the fraction of full power at which the core is operating
   (2) K(Z) is the function given in Figure TS.3.10-5
   (3) Z is the core height location of F_Q^N
   (4) RBP(BU) is the Rod Bow Penalty as a function of region average burnup as shown in Figure TS.3.10-7
   (5) Region is defined as those assemblies with the same loading date
   (6) V(Z) is the function given in Figure TS.3.10-8
   (7) P^1 is the largest fraction of full power at which the plant will operate prior to the next target flux measurement.
   (8) The F_Q^N of b, above, is not applicable in the following core regions as measured in core height from the bottom of the fuel; the lower region from 0 to 10% inclusive, and the upper region from 90 to 100% inclusive.
   (9) Equilibrium conditions are defined as -
      (a) The delta flux difference shall be constant within ± 1% ΔI over the previous 24 hour period.
      (b) The power level shall be constant within ± 2% over the previous 24 hour period.

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

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

   b. If either measured hot channel factor exceeds its limit specified under 3.10.B.1.a, 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.

c. If the measured hot channel factor F_Q^N exceeds its limit as specified under 3.10.B.1.b, then one of the following actions shall be taken:

1. Within 48 hours, place the reactor in a configuration for which Specification 3.10.B.1.b is satisfied; or

2. Reduce thermal power by 1% for each percent that the measured F_Q^N exceeds the limit specified in 3.10.B.1.b. Thermal power may be increased to a power such that the associated F_Q^N would comply with 3.10.B.1.b.

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 -3 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 $\pm 5\%$ 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 $\pm$ 5% target band for a maximum of one* hour (cumulative) in any 24-hour period provided that the difference between the indicated axial flux difference and the target flux difference does not exceed an envelope bounded by -10 percent and +10 percent at 90% power and increasing linearly to -25 percent and +25 percent at 50 percent power as shown in Figure TS.3.10-6.

   b. If 6.a is violated for two operable excore channels then the reactor power shall be reduced to 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 16 hours during incore/excore calibration.

$F_Q(Z)$, 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, Nuclear Hot Channel Factor, is 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 for F_Q^N of 2.145 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 an adequate peak clad temperature 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 of Westinghouse fuel 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 once 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 instrume ation error.

2. Control rod banks are seq ed with overlapping banks as described in Technical Spec 3.10.

3. The control bank insertion limits are not violated.

4. The part length control rods are not inserted.

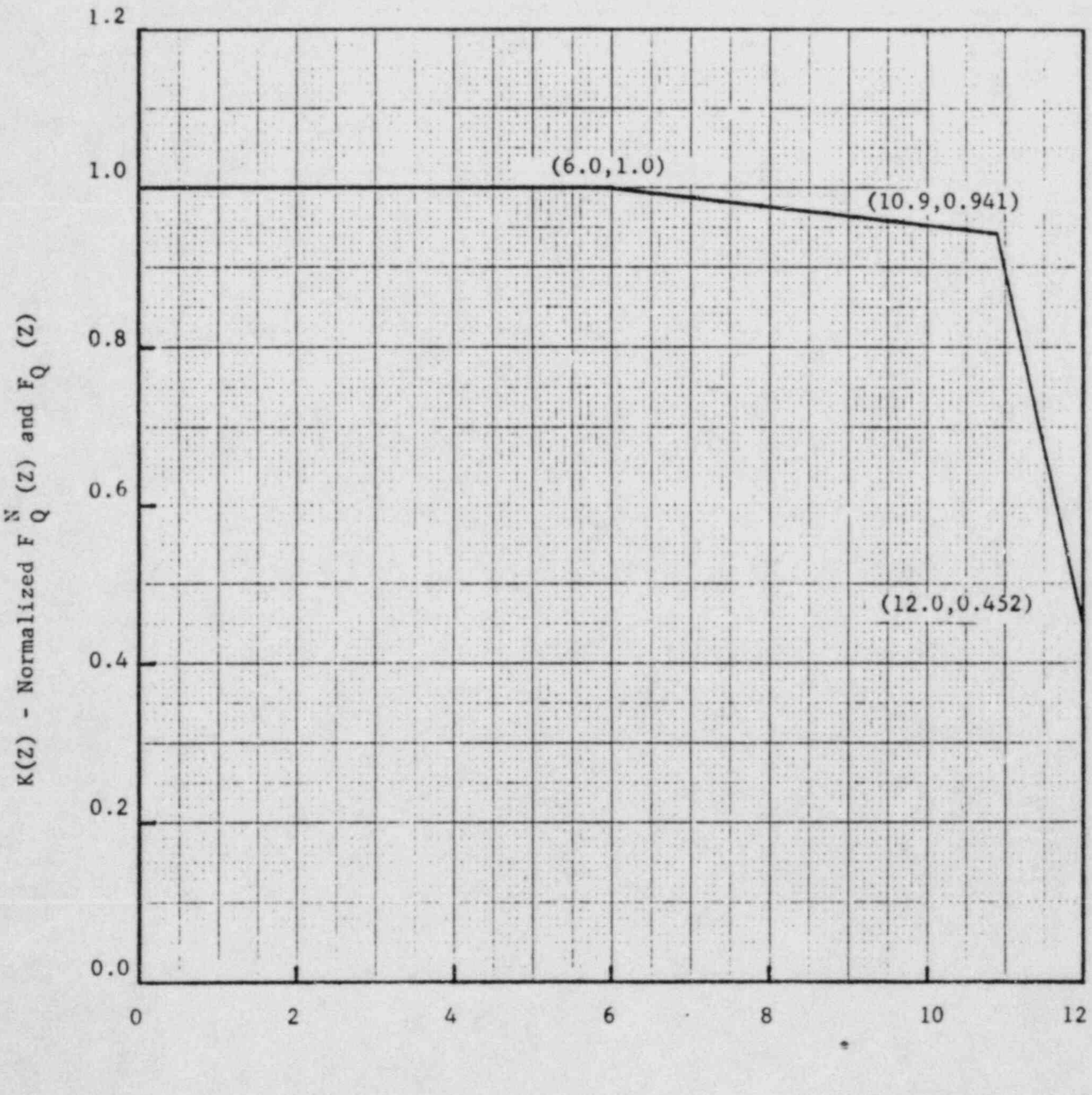
5. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure 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 (Δ I) and a reference value which corresponds to the full power equilibrium value of Axial Offset (Axial Offset = Δ 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 2.145 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 Δ I 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 the allowed deviation from target flux difference as a function of thermal power.



HOT CHANNEL FACTOR NORMALIZED

OPERATING ENVELOPE FOR F_Q = 2.21

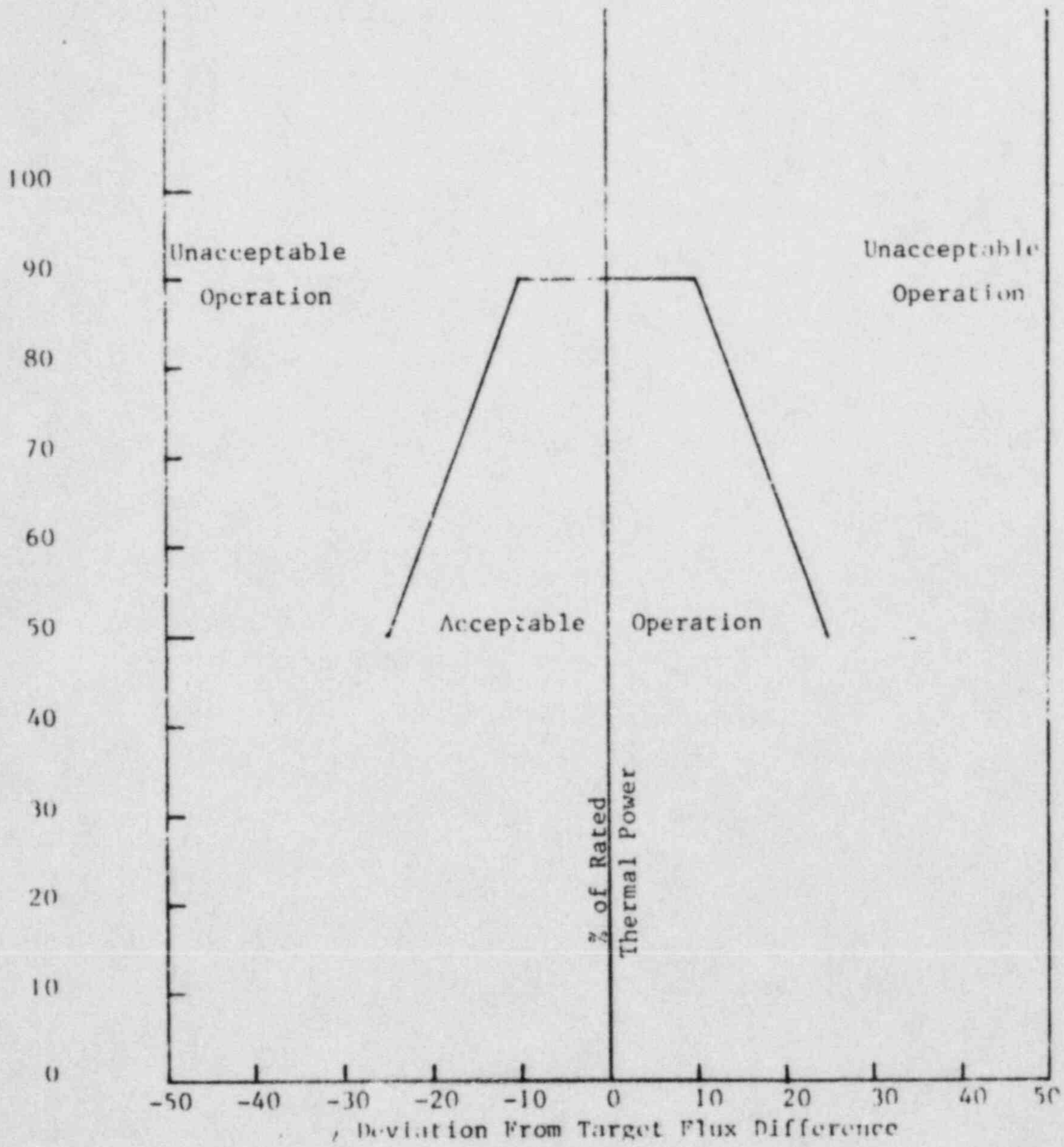
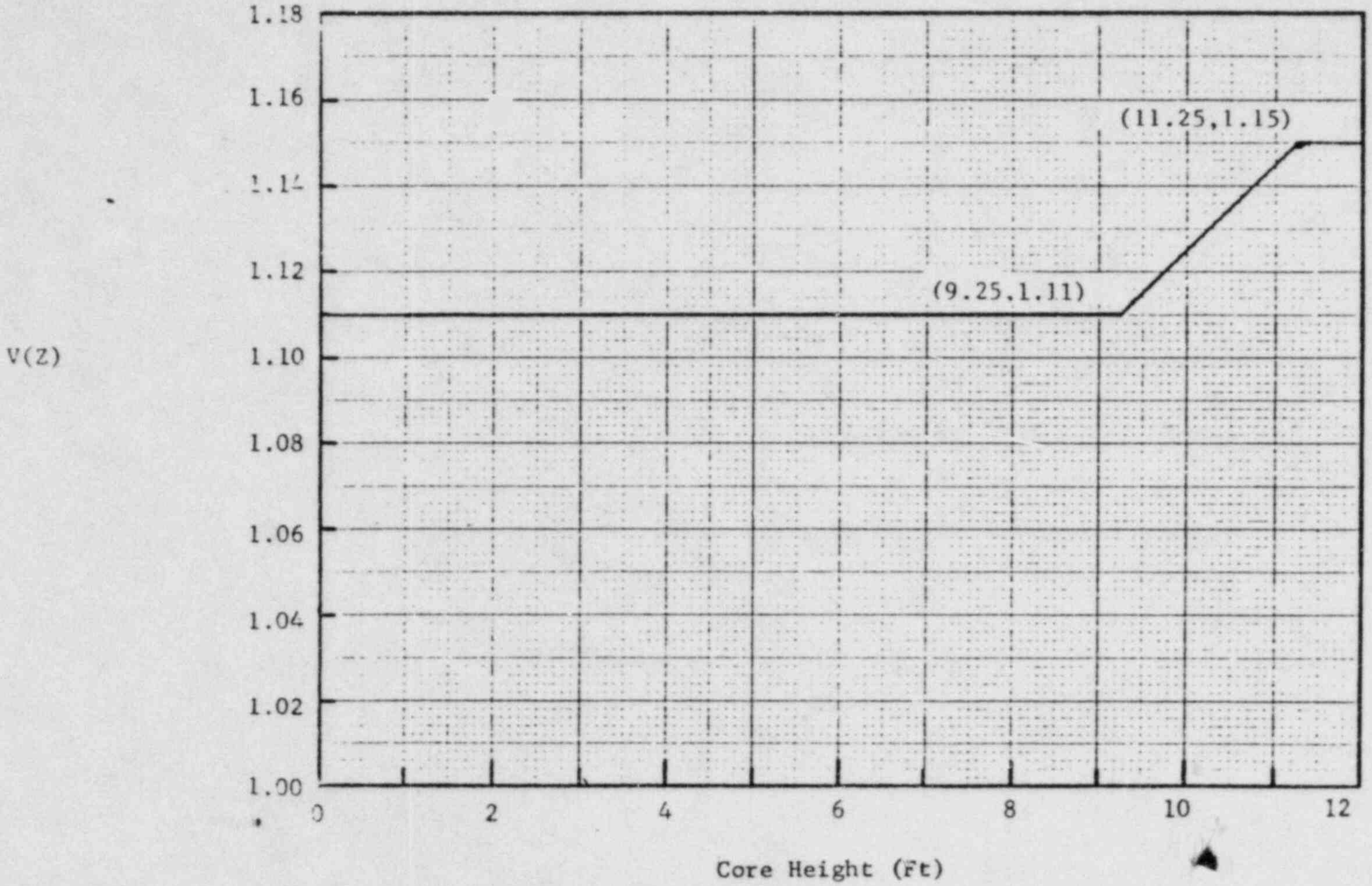


Figure TS.3.10-6 DEVIATION FROM TARGET FLUX DIFFERENCE AS A FUNCTION OF THERMAL POWER



V(Z) as a Function of Core Height

FIGURE TS.3.10-8

Amendment No. 35, Unit 1
Amendment No. 29, Unit 2

5.3 REACTOR

A. Reactor Core

1. The reactor core contains approximately 48 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 121 fuel assemblies. Each fuel assembly contains 179 fuel rods.[(1)]

2. The average enrichment of the reload core is a nominal 2.90 weight per cent of U-235. The highest enrichment is a nominal 3.50 weight per cent of U-235.

3. In the reactor core, there are 29 full-length RCC assemblies that contain a 142-inch length of silver-indium-cadmium alloy clad with stainless steel.[(2)]

B. Reactor Coolant System

1. The design of the reactor coolant system complies with all applicable code requirements.[(3)]

2. All high pressure piping, components of the reactor coolant system and their supporting structures are designed to Class I requirements, and have been designed to withstand:

   a. The design seismic ground acceleration, 0.06g, acting in the horizontal and 0.04g acting in the vertical planes simultaneously, with stresses maintained within code allowable working stresses.

   b. The maximum potential seismic ground acceleration, 0.12g, acting in the horizontal and 0.08g acting in the vertical planes simultaneously with no loss of function.

3. The nominal liquid volume of the reactor coolant system, at rated operating conditions, is 6100 cubic feet.

C. Protection Systems

The protection systems fc :he reactor and engineered safety features are designed to applicable codes, including IEEE-279, dated 1968. The design includes a reactor trip for a high negative rate of change of neutron flux as measured by the excore nuclear instruments.[(4)]
The system is intended to trip the reactor upon the abnormal dropping of more than one control rod[(4)] If only one control rod is dropped, the core can be operated at full power for a short time, as permitted by Specification 3.10.

References

(1) FSAR, Section 3.2.3
(2) FSAR, Sections 3.2.1 and 3.2.3
(3) FSAR, Table 4.1-9
(4) FSAR, Section 7

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 35 TO FACILITY OPERATING LICENSE NO. DPR-42
AMENDMENT NO. 29 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 December 29, 1978 (Reference 1), as supplemented February 21, 1979 (Reference 2) and March 30, 1979 (Reference 3), Northern States Power Company (the licensee) proposed to change the Technical Specifications for Prairie Island Nuclear Generating Plant Unit Numbers 1 and 2 to permit Cycle 5 operation. During our review of the proposed amendments we found that certain modifications were necessary to meet our requirements. These modifications were discussed with the licensee's staff, they have agreed to the modifications and the modifications are incorporated. We note that the Safety Analysis Report (Ref. 4) refers only to Unit 1 although the application is for Unit 1 and Unit 2. Our review applies to both units, however, the licensee is required to submit the SAR for Unit 2 prior to the next Unit 2 reload to verify that the Technical Specification for Unit 2 will remain unchanged.

The proposed reload consists of replacing 40 Westinghouse fuel assemblies with 40 fresh fuel assemblies manufactured by Exxon Nuclear Company (ENC). These assemblies will be loaded on the periphery of the core. The remaining 81 Westinghouse assemblies, which have a variety of burnups, will be scatter-loaded in the center portion of the core. The licensee supported his request by the analyses performed by Exxon Nuclear Company and reported in a series of technical documents (References 4, 5, 6, 7, 8, 9, 10 and 11). In addition, the Westinghouse ECCS analysis performed with the evaluation model corrected for the Zr-water reaction error is also provided (Reference 12).

The licensee has proposed the following changes to the Technical Specifications for the Prairie Island plant:

(1) Change of the limit curve for target flux difference (Fig. TS 3.10-6)

(2) Addition of a curve defining the transient allowance factor, V(Z), used in the Power Distribution Control, Phase 2 procedure (Fig. TS 3.10-8)

(3) Removal of the definition of the Interim Fuel Limits related to the power distributions to be used in the LOCA analyses and to the fuel residence time in Unit 1, Cycle 1. Deletion of this definition is warranted because the 1971 Policy Statement and 1972 Technical Report have been superseded by the 10 CFR 50 Appendix K criteria and the power distribution limits in Section 3.10 of the Technical Specifications.

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(4) Change of the requirement for negative reactivity coefficient during power operation. For the core containing Exxon fuel, it is required that only the isothermal temperature coefficient needs to be negative.

(5) Change of the limiting value for the nuclear hot channel factor F_Q^N from 2.25 to 2.145 and modification of the hot channel factor normalized operating envelope (Fig. TS 3.10-5)

(6) Change in the specifications for the highest fuel enrichment to 3.5 w/o of U-235 and deletion of the reference to the burnable poison rods

(7) Removal of the burnup dependent multiplier in the expression for the limiting enthalpy rise factor, $F^N_{\Delta H}$, for Exxon fuel

Evaluation

Fuel Design

The new Region 7 fuel has been specifically designed by ENC to be compatible to the fuel previously supplied by Westinghouse. The fuel is similar to the Westinghouse bundle design with the most significant differences listed below:

(1) The cladding thickness is 30 mils which is approximately 23% thicker than the Westinghouse cladding.

(2) There is a slight difference in fuel pellet design.

(3) The bimetallic spacers are made from Z. 4 with Inconel 718 spring.

(4) The fuel assembly tie plates are mechanically locked to the Zr-4 guide tubes.

(5) The mean pellet density is 94% of theoretical density.

(6) The enrichment is 3.40 w/o of U-235.

(7) There are 64 rods (in 16 assemblies) which contain 1 w/o of uniformly distributed gadolinia burnable poison (Gd_2O_3).

The details of the ENC fuel design are described in Reference 7. We reviewed those aspects of the design which are most relevant to the reactor safety and found them acceptable. They are outlined in the discussion which follows.

The GAPEX code (Reference 13) was used to calculate stored energy for input to the loss of coolant accident (LOCA) calculation. We have previously reviewed and approved the GAPEX code for fuel temperature and internal pressure calculations in PWR fuel (Reference 14).

The cladding mechanical stability was verified in order to demonstrate that it will not collapse into a gap caused by fuel densification. Reference 7 presents calculations which show that no cladding collapse is predicted for Cycle 5. The calculations are done with two computer codes. The RODEX code (Reference 15) calculates the cladding temperature and fuel rod internal pressure while COLAPX (Reference 16) calculates the collapse time using the RODEX input. We have reviewed COLAPX and found it acceptable for cladding collapse calculations. We have not approved RODEX. However, the models in RODEX which affect cladding temperature and internal pressure are similar to those in the GAPEX code, which has been approved. Moreover, since the clad collapse analyses for the Westinghouse fuel do not predict collapse during Cycle 5, and since the cladding for the Exxon fuel is thicker than that of the Westinghouse fuel, which makes it more resistant to clad collapse, we have reasonable assurance that the results of the RODEX analysis are acceptable. Exxon has demonstrated that because of the thicker cladding, the reload fuel is less susceptible to stress corrosion than the Westinghouse fuel. Based upon in-reactor experience and testing of nearly identical ENC fuel assemblies, it was also shown that the potential for fretting corrosion failure is very low in the reload fuel assemblies.

The licensee has considered the effect of fuel rod bowing on the DNBR limit by using the calculational procedure outlined in References 4 and 7. This procedure uses data on the magnitude of fuel rod bowing obtained by Exxon on fuel of similar design. We have reviewed these calculations and find that they are not acceptable because the description of the statistical calculations in the reference reports (References 4 and 7) were not described in sufficient detail to give a precise meaning to the 95/95 limit which was subsequently used. These calculations are being discussed generically with ENC (Reference 37). The licensee has demonstrated that Prairie Island Unit 1 has sufficient margin to overcome the maximum possible departure from nucleate boiling ratio (DNBR) reduction (that corresponding to full contact of the bowed fuel rod with adjacent rods in a sub-channel containing an unheated thimble tube). This margin is due to the difference between the minimum DNBR from the most limiting anticipated transient and the DNBR safety limit of 1.3.

Fuel rod bowing also affects F_Q^T by changing the local neutron moderation. We have not yet approved the ENC method for calculating the magnitude of fuel rod bowing. Therefore, the ENC method, used by the licensee, is also not acceptable at this time.

The usual method of accommodating the rod bow effect on F_Q^T is to make use of the fact the uncertainties in F_Q^T are independent of each other and may, therefore, be combined statistically as

$$1 + \sqrt{{F_Q^E}^2 + {F_Q^U}^2 + {F_Q^B}^2}$$

Where F_Q^E is the engineering uncertainty,

F_Q^U is the nuclear measurement uncertainty

and F_Q^B accounts for the effect of fuel rod bowing on F_Q^T

In the analysis, a value of the uncertainty assumed for F_Q^T is 1.0815. This value corresponds to an F_Q^B of .057. However, using the Westinghouse rod bowing curve as an upper limit to the amount of bowing expected in ENC fuel (a conservative assumption), the value of F_Q^B predicted for the end of Cycle 5 is .085. This, in turn, corresponds to an uncertainty of F_Q^U of 2% greater than the value used in the analysis at the end of the Cycle.

We require that this calculated 2% reduction in F_Q^T be included in the Prairie Island Technical Specifications until such time as it can be removed or modified by an NRC approved model. The licensee has chosen to treat this reduction as a function of burnup whose value at the end of the cycle will be 2%.

In the present reload, the licensee proposes to include 64 fuel pins (4 pins per assembly) containing 1 w/o of gadolinium oxide (Gd_2O_3) uniformly distributed in UO_2 matrix. ENC used similar fuel in the Palisades plant where 32 gadolinia bearing fuel rods were loaded in the core during the Cycle 3 reload. In addition, ENC had several years of experience with irradiating gadolinium bearing rods in BWR's. The examination of these fuel rods revealed no abnormalities and gamma scan measurements have demonstrated the accuracy of the ENC calculational methods used in predicting the depletion of gadolinium during fuel burnup.

After examining all the information available to us on gadolinium poisoned fuel and after evaluating the previous Exxon experience in this area, we conclude that the gadolinium bearing fuel rods would be expected to perform satisfactorily during the Cycle 5 operation. However, because of the relatively limited experience with gadolinium containing fuel rods and because this fuel is used for the first time in Prairie Island, we note that ENC perform a visual inspection of a sufficient number of irradiated fuel bundles to verify that the performance of the ENC fuel and especially the fuel containing gadolinium oxide is acceptable. The amount of surveillance should depend on the coolant activity during plant operation and will be decided by the licensee with our approval 90 days before scheduled plant shutdown for the next cycle refueling.

Based on successful experience with irradiating previous loadings of Exxon PWR fuel and the analyses which have been done for Cycle 5, it is we concluded that the fuel loading for Prairie Island Unit 1, Cycle 5 will perform in a safe and acceptable manner.

Thermal Hydraulic Design

The new ENC fuel was designed to have thermal hydraulic characteristics closely matching those of the existing fuel and it is not expected to introduce any major differences in the thermal hydraulic behavior of the core. Minor design changes included a slight difference in the flow areas of various assembly components and resulted in higher hydraulic loss coefficients. This change is very small and at nominal reactor operating conditions, the flow rate to each fuel type was within 5% of the core assembly average flow for a mixed core configuration. In addition, as it was pointed out by the licensee in Reference 4, the ENC fuel having a higher flow resistance would be located on the periphery of the core, and the fuel in the center of the core, with higher radial power peaking, will receive more flow. The licensee has shown that at 112% of rated power the minimum DNBR is 1.97 for ENC fuel which is only 4% lower than 2.05, the DNBR value for the Westinghouse fuel. Additional conservatism stems from the fact that the DNBR was calculated using $F_Q T$=2.32. The proposed limiting value for F_Q^T is 2.21. The analysis of fuel and cladding was performed for Cycle 5 with the NRC approved ENC methods (Reference 13). Even with the most conservative assumptions the calculated fuel and cladding temperatures were well below the design limits.

From the information and analyses presented by the licensee, we conclude that the ENC designed fuel is compatible with the present Westinghouse fuel in the Prairie Island plant and that the thermal hydraulic criteria will not be exceeded during the plant operation in Cycle 5.

Nuclear Design

The Cycle 5 loading will consist of one Region 4, 40 Region 5 and 40 Region 6 fuel assemblies with burnups ranging from 9,592 megawatt days per metric ton of uranium (MWD/MTU) to 27,208 MWD/MTU and 40 Regions 7A and 7B fresh fuel assemblies containing four fuel pins in each assembly with 1 w/o of gadolinium oxide (Gd_2O_3). The projected length of Cycle 5 is 11,300 MWD/MTU based on an assumed Cycle 4 length of 10,900 MWD/MTU. The Cycle 5 operation is designed with total peaking factor envelope limit of <2.21/P for two loop plant operation (where P is fraction of full power) and with the modified hot channel factor normalized operating envelope (Fig. TS 3.10-5) to account for the new value of F_Q. These new hot channel limits will assure that DNBR will be greater than 1.3 during steady state, load follow and transient conditions and that LOCA requirements are met at rated plant power.

The licensee has specified new values for the axial flux difference limits. These new limits relate to the allowable deviation of the axial flux difference from its target band when the reactor is operating below 90% of its rated power. These new limits are shown in Fig. TS 3.10-6 of Reference 1. They are more restrictive than the present Technical Specifications limits.

It was shown that neutronically there is a close similarity between Cycle 5 and the reference cycle (Reference 17). The gadolinia bearing assemblies are predicted to have only a relatively insignificant impact on the overall core neutronic behavior. Most of the kinetic parameters for Cycle 5 fall within the bounds of the values determined for the reference cycle and used in the previously reported safety analyses. A noted exception is the moderator temperature coefficient which is predicted to be positive at the beginning of Cycle 5 when reactor is above 70% of its rated power (moderator temperature coefficient (MTC= +1.00pcm/°F at beginning of cycle and hot zero power condition (BOC and HZP)). However, the licensee has indicated that although the moderator temperature coefficient could be positive, the isothermal coefficient remains always negative and at HZP, all rods out (ARO) condition it is equal to -0.7 pcm/°F which is sufficient to meet the revised Technical Specifications with no rod insertion. The licensee has predicted core power distribution for Cycle 5. The highest calculated values of F_Q^N and $F_{\Delta H}^N$ are 1.680 and 1.395, respectively, and hence they are well below the Technical Specification limits of F_Q^N=2.145 and $F_{\Delta H}^N$=1.55.

There are no changes proposed to the control rod insertion limits for Cycle 5. There are a number of criteria which the control rod insertion limits are checked against in each cycle. The most important are shutdown margin, ejected rod worth and $F^N_{\Delta H}$. The existing insertion limits are predicted to meet these criteria during Cycle 5. The hot full power shutdown margin is calculated by the licensee to be 2539 pcm at BOC and 2598 pcm at end of cycle (EOC) in Cycle 5 compared to the Technical Specification shutdown margin requirement of 1000 pcm and 2000 pcm for BOC and EOC, respectively, and a margin of 1800 pcm used in the steamline break analysis. The positive difference existing between the predicted and required margins and the fact that the predicted margins are reduced by 10% to account for calculational uncertainties makes the shutdown margins specified by the licensee for Cycle 5 acceptable. In addition, the validity of the prediction will be verified during the startup physics test program by measuring the worth of the regulating banks.

The licensee has performed extensive analyses in order to prove that the presence of 64 gadolinium bearing fuel rods located in 16 assemblies would not cause significant degradation of the power distribution in the core during Cycle 5. The calculations were performed for assembly-wide and core-wide power distributions using the standard ENC methodology (References 19, 20 and 21). We have reviewed these calculations and have ascertained that the presence of gadolinium oxide increased the power peaking in an assembly at BOC condition by about 3%. The power distribution among different assemblies in the core was calculated by the licensee for three different fuel exposures, corresponding to BOC, 2500 MWD/MTU and 5500 MWD/MTU when it was predicted that the gadolinium poison will be completely depleted. In the calculations, two gadolinium reactivity worths were assumed corresponding to the value used in the Cycle 5 design and to the value 40% higher. The resultant power distributions were compared to the predicted distribution assuming no gadolinium poison present. The licensee has shown that the gadolinium poison bearing fuel rods increased non-uniformity in power distribution between fuel assemblies at BOC by about 4-1/2% for the design reactivity worth and 6-1/2% for 140% of the design worth. For higher exposures, the effect of gadolinium on power distribution decreased and at about 5500 MWD/MTU it became insignificant. The licensee has also calculated the corresponding nuclear hot channel (F_Q^N) and enthalpy rise ($F^N_{\Delta H}$) peaking factors which are listed below:

	BOC		5500 MWD/MTU	
	F_Q^N	$F_{\Delta H}^N$	F_Q^N	$F_{\Delta H}^N$
No Gadolinium	1.70	1.40	1.53	1.35
Design Worth	1.63	1.40	1.54	1.36
1.4 x Design Worth	1.66	1.42	1.55	1.36

From these results, we conclude that the presence of gadolinium in Cycle 5 would not significantly affect power distribution in the core.

The accuracy of the predictive data is confirmed by the results obtained in the gadolinium demonstration program in the Palisades plant where ENC has compared the predicted and measured power distributions arrived at a 1% agreement.

Based on the above information, we conclude that the presence of 64 gadolinium bearing fuel rods would not produce the changes in core power distribution which would compromise safe operation of the plant in Cycle 5.

Power Distribution Control and Monitoring

The ENC LOCA analysis for the Prairie Island Units (Reference 6) assumes as an initial condition that the core peaking factor, F_Q^T, is 2.21. Provision is required to ensure that this F_Q^T is not exceeded in normal operation of the power plant in order for the conclusions of the LOCA analysis to remain valid. The licensee has proposed to accomplish this through use of ENC Power Distribution Control-Phase 2 (PDC-2) procedures for reload cores (Reference 10).

We have accepted an earlier ENC power distribution control strategy and analysis (Reference 18) which justifies that the peaking factor will not exceed 2.32 providing only that all of the conditions assumed in the analysis are observed in operation of the reactor. This scheme is the same as Westinghouse constant axial offset control and has been approved for use at Prairie Island for several years. PDC-2 uses all the rules of the present scheme, but differs in that the F_Q^T protected against is the product of the measured equilibrium peaking factor and a predetermined axial height dependent transient allowance factor, V(Z). Because the measured equilibrium peaking factor represents the actual state of the reactor, and not the spectrum of possible states necessarily assumed in the earlier analysis, PDC-2 can justify peaking factors considerably lower than 2.32, probably at least as low as 2.0, depending on reactor cycle and time during cycle.

Reactor experience will be needed to be more precise about how low a peaking factor can be justified with PDC-2. The reason is that the peaking factor values discussed are for the flat portion of the axial height dependence, at the core centerline. The axial dependence of F_Q^T has two components. First, is the familiar K(Z) curve contained in all Westinghouse reactor Technical Specifications. This dependence requires strongly reduced peaking toward the top of the core. Second, is the V(Z) function which increases toward the top of the core. Thus, even though the reactor naturally does not have strong peaking toward the top of the core, the decreasing requirement of K(Z) is in opposition to the increasing character of V(Z), so that the top of the core may be more limiting than the center regions we normally identify with a limiting value of F_Q^T.

Our review of PDC-2 is not complete, however, the review has progressed sufficiently, and special allowances and extra surveillance procedures have been agreed to by the licensee, such that we have an acceptable basis for use of PDC-2 at the Prairie Island reactors. The remainder of this section will discuss the status of our review of PDC-2, the provisions made for Prairie Island, and why they are acceptable.

Since PDC-2 uses the measured equilibrium power distribution to determine F_Q^T, we have been concerned with the sensitivity of F_Q^T to departures from equilibrium during the measurement. This concern has been covered for Prairie Island by putting into the Technical Specifications the following very stringent requirements for equilibrium on the power distribution measurement used to determine compliance with the peaking factor limit:

1. The delta flux difference shall be constant within $\pm$ 1% ΔI over the previous 24 hour period.

2. The power level shall be constant within $\pm$ 2% over the previous 24 hour period.

These allowable variations are sufficiently small that we are confident the measured power distribution will not be less than its equilibrium value. ENC is performing analyses to support a less stringent definition of equilibrium for future use or to allow for the removal of the restriction.

Another area of concern to us is that of potential increase in the measured equilibrium power distribution between measurement intervals (upburn). Known occurrences of this phenomenon involve an increase in the radial plane peaking factor, F_{xy}, as a result of depletion of burnable poison loaded into cores. Otherwise, in general, F_{xy} tends to decrease with increasing exposure. Further analysis by and discussions with ENC are anticipated to resolve the treatment of the potential for an increase in F_Q^T from upburn.

For Prairie Island, the licensee has agreed (Reference 1) either to apply to the measured equilibrium power distribution (in addition to the normal factors of 1.05 measurement uncertainty and 1.03 engineering uncertainty) a factor of 1.02 to account for upburn, or he will increase the frequency of the power distribution measurements from the normal once per month to once per week. We are convinced that the allowance of 1.02 will conservatively bound possible upburn effects between monthly maps. Alternatively, if the margin is needed to avoid a derate by application of the 1.02 factor, weekly core mapping is sufficiently frequent to incorporate upburn effects into the measured equilibrium power distribution.

Other areas of our review of PDC-2 which remain open are:

(1) Xenon modeling
(2) Uncertainty in the V(Z) function
(3) Allowed axial offset limits below 90% power
(4) Transient analyses of power shapes allowed by PDC-2.

We are concerned that item (1), the Xenon model, is tuned to one set of experiments, and therefore might lead to errors when applied to other situations. ENC has committed in discussions with us to demonstrate the applicability to the data to be obtained from Prairie Island.

The open questions involving items (2) and (3) involve a lack of familiarity with the detailed analyses ENC has performed to reach their conclusions. We are continuing our review in this area.

ENC is performing analyses which will show that the minimum DNBR in limiting transients is greater for initial conditions consisting of power shapes allowed by PDC-2 than for design power shapes. ENC will provide the results of these calculations.

We have concluded that use of PDC-2 in Prairie Island is acceptable even though the enumerated items (1) through (4) are not completely resolved because the analyses and review involved have progressed sufficiently that we are certain the outcome will permit the conservative use of PDC-2. In addition, the F_Q^T required for Prairie Island is 2.21, which allows more linear power density margin than if a lower peaking factor had to be protected by PDC-2. There is ample thermal margin in Prairie Island compared to power plants with a higher average power density. We consider that the linear power density and thermal margins compensate for any small uncertainty in PDC-2 until our review is completed.

As described above, the licensee has agreed (Reference 1) to provide extra surveillance and uncertainty allowances to permit application of PDC-2 to the Prairie Island reactors. He has also provided a suitable definition of equilibrium and other measures necessary to implement PDC-2 in proposed Technical Specification changes (Reference 1). We, therefore, find the proposed Technical Specification changes acceptable to ensure PDC-2 procedures will maintain the F_Q^T below 2.21 in normal operation of the Prairie Island reactors.

Transient and Accident Analysis

The licensee has reviewed and/or reanalyzed the anticipated operating occurrences and postulated accidents. The results of these analyses are presented in Reference 5. The calculations were performed using the transient simulator code PTS-PWR2 (Reference 22). This code is under review by the NRC and although it is not yet completed, the review has progressed sufficiently to justify, in conjunction with the conservative values of the kinetics parameters and of the initial state points, the conclusion that the analyses using PTS-PWR2 will provide sufficient margin to design criteria on peak linear heat generation rate and DNBR. The conservatism of the reactivity coefficients assumed in the safety analyses are to be confirmed as part of the startup measurement program which we require.

The reload fuel design has been shown to be both neutronically and hydraulically compatible with the Westinghouse fuel and hence we do not expect the system response during plant transients to be significantly.

different from the responses determined in the reference analyses (Reference 17). However, due to slightly different values of core parameters and in particular, to a positive moderator feedback coefficient at low power operation during the initial part of Cycle 5, some of the most limiting transients and accidents had to be reanalyzed. The licensee has presented reanalyses of the following events:

1. Fast and slow rod withdrawal
2. Loss of load
3. Loss of primary flow
4. Locked rotor
5. Large and small steam line break
6. Rod ejection

The analyses were performed assuming the most conservative starting conditions with the maximum hot channel factor, F_Q, at 2.32. Events 1 through 4 were initiated from hot full power (HFP) condition and event 5 from HZP condition. The analysis for event 6 was carried out for both HFP and HZP conditions and it was shown that the limiting results corresponded to HZP. In the transient analysis, the licensee has demonstrated that the criterion of system pressure not exceeding 110% of design pressure (2750 psia) was satisfactorily met by all the analyzed transients (References 3 and 5). In addition, except for the locked rotor and rod ejection accidents, the minimum DNBR determined in the analyses remained above the 1.3 limit based on W-3 correlation. Both accidents are category IV events with low probability of occurrence. For the locked rotor transient, the DNBR is calculated to reach 1.09. However, the licensee has shown that at this DNBR less than one percent of fuel rods will experience DNB and even if fuel failure is postulated to occur for all these rods, the potential release of activity is judged substantially less than the 10 CFR 50 Part 100 permitted site boundary dose rates.

The rod ejection analysis has been performed with the methods described in Reference 11. The results of this analysis have indicated that the maximum system pressure and the energy deposited in fuel pellets were less than the limits defined in Regulatory Guide 1.77 (Reference 23).

The licensee has provided a list of transients which were included in the original reference cycle analysis (Reference 17), but which were not reanalyzed for Cycle 5 because either they were not affected by the reload fuel or they were bounded by the corresponding reference analyses. We have reviewed all these transients in the past and concur with the licensee's conclusion that for the Cycle 5 operation these transients need not be reanalyzed.

ECCS Analysis

Two emergency core cooling system (ECCS) analyses were provided for Prairie Island, Units 1 and 2. One analysis was for ENC and one for a Westinghouse fueled plant.

The ENC large break analysis (Reference 6) was performed using the WREM-II PWR evaluation model described in References 24 and 25. The model has recently been modified by including the new REFLEX code to replace the existing RELAP4-EM/FLOOD portion of the Exxon's approved model (Reference 8) and by introducing several minor code updates (Reference 9). Both these changes have been reviewed and approved by the staff (References 26 and 27). The applicability of both ENC and Westinghouse ECCS evaluation model to the two-loop PWRs with upper head ECCS injection (UPI) have been challenged by the NRC on generic ground and the licensee was requested to provide an ECCS analysis performed with a model including the UPI effect correction. In the meantime, while this model is being developed, the licensee was requested to evaluate the impact of injecting ECCS fluid above the core using the model which was developed by the NRC staff (Reference 28), modified by Westinghouse (Reference 29) and subsequently approved by the NRC for the interim use in two-loop Westinghouse plant analyses (Reference 30). This model was used to correct the results obtained by the WREM-II model for the UPI effect.

The ENC ECCS analysis has been performed for a spectrum of breaks which included the guillotine double ended cold leg (DECLG) breaks with discharge coefficients of 1.0, 0.6 and 0.4 and split breaks with break areas at 8.25, 4.95 and 3.30 ft^2. The limiting value of total hot channel peaking factor, F_Q, was 2.21 and one percent of steam generator tube plugging was assumed. The results of the analysis are listed below:

Limiting Break: DECLG with C_D=0.4
Peak Clad Temperature: 2198°F
Local Zr-Water Reaction: 12.34%
Total Zr-Water Reaction: <1.0%

These values meet the 10 CFR 50.46 criteria and the ECCS analysis is, therefore, acceptable.

No small break LOCA analysis was provided for Prairie Island since, by analogy with the similar analysis previously performed for another two-loop Westinghouse plant (Reference 31), the licensee has found that the small break LOCA would not be limiting.

Westinghouse large break LOCA analysis had to be performed because an error was discovered in the Westinghouse ECCS evaluation model which resulted in incorrectly calculating peak clad temperatures in all previously submitted Westinghouse analyses (Reference 32) due to an error in the metal-water reaction calculation. Following discovery of this error, the licensee administratively reduced the total peaking factor limits for Units 1 and 2 from F_Q=2.32 to F_Q=2.21. This new value of F_Q was intended to conservatively accommodate the error. The licensee also committed to provide a new LOCA analysis which was to be performed with an acceptable evaluation model. These requirements were confirmed in the Order for Modification of License, issued for the Prairie Island Units 1 and 2 (Reference 32), where the NRC conditionally approved the total peaking factor limit of F_Q=2.21. In this order, we requested the licensee to provide, 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 metal-water reaction error. The current Westinghouse ECCS analysis (Reference 33) was submitted in response to this request. It was performed with the NRC approved February 1978 version of the Westinghouse evaluation model (Reference 34) which, in addition to including the correction of the Zr-water reaction error and several code maintenance and analytical improvements, contained the changes described in References 33 and 34.

The submitted analysis was performed with the total peaking factor, F_Q, of 2.28 and assuming one percent of steam generator tubes plugged. It was limited to only one break size which was DECLG with C_D=0.4. However, the licensee has provided a generic two-loop LOCA analysis performed for a spectrum of DECLG breaks with discharge coefficients ranging from C_D=0.4 to C_D=1.0 (Reference 33). From this analysis, it could be concluded that the replacement of the October 1975 version by the February 1978 version of the Westinghouse ECCS evaluation model would not alter the critical break size.

The correction of upper plenum injection (UPI) effect was not included in the present analysis because it was previously demonstrated (Reference 30) that it is negative and causes reduction of 10°F in peak clad temperature (PCT). Therefore, ignoring the UPI effect makes the analysis more conservative. The results of the analysis are provided below:

Limiting Break: DECLG with C_D=0.4
Peaking Clad Temperature: 2179°F
Local Zr-Water Reaction: 7.8%
Total Zr-Water Reaction: <0.3%

All the values reported are below the limits of 10 CFR 50.46.

The total peaking factor, F_Q, from the ENC analysis is more limiting (lower). It is used, therefore, in defining the plant's Technical Specification limits. Because it is below 2.32, the licensee would be required to use power distribution control by operating the plant in accordance with the PDC-2 methodology which was discussed in the previous section.

Startup Physics Tests

The startup physics tests for Prairie Island Units 1 and 2, Cycle 5 will be similar to those for previous startups following reloading. The proposed startup physics test program was described in the reload submittal (Reference 1,3). This program includes low power critical boron concentration tests, temperature coefficient tests, rod worth measurements and power distribution measurements. At higher powers, core power distribution measurements will be made. The acceptance criteria and the actions to be taken if the acceptance criteria are not met were specified in Reference 3. We have reviewed the entire program, including the tests to be performed, the acceptance criteria and the actions to be taken if the acceptance criteria are not met, and have found it to be acceptable. The results of this startup physics test program will be submitted to the NRC within 90 days after startup.

Summary

Based on the above evaluation, we conclude that the Prairie Island Nuclear Generating Plant Units 1 and 2, may be operated during Cycle 5 with the core comprising 40 new Exxon fuel assemblies. In addition, we have reviewed the ECCS submittal based on the corrected model (February 1978) and we find it 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) 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 Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: April 20, 1979

References:

1. Northern States Power Company letter (L. O. Mayer) to NRC, dated December 29, 1978, transmitting the proposed changes to the Technical Specifications for Prairie Island Nuclear Generating Plant, Units 1 and 2.

2. Northern States Power Company letter (L. O. Mayer) to NRC, dated February 21, 1979.

3. Northern States Power Company letter (L. O. Mayer) to NRC, dated March 30, 1979.

4. XN-NF-78-47, Exxon Nuclear Company, "Prairie Island Unit 1 Nuclear Plant Cycle 5 Safety Analysis Report," November 1978.

5. XN-NF-78-35, Exxon Nuclear Company, "Plant Transient Analysis for the Prairie Island Nuclear Power Plant Units 1 and 2," November 1978.

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

7. XN-NF-78-34(P), Exxon Nuclear Company, "Generic Mechanical and Thermal Hydraulic Design for Exxon Nuclear 14x14 Reload Fuel Assemblies with Zircaloy Guide Tubes for Westinghouse Two-Loop Pressurized Water Reactors," November 1978.

8. XN-NF-78-30, Exxon Nuclear Company, "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model Update ENC WREM-IIA," August 1978.

9. XN-NF-78-25, Revision 1, Exxon Nuclear Company, "Big Rock Point LOCA Analysis Using the Exxon Nuclear Company Non-Jet Pump BWR Evaluation Model - Large Break Problem," September 1978.

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

11. XN-NF-78-44, Exxon Nuclear Company, "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," January 1979.

12. Northern States Power Company letter (L. O. Mayer) to NRC, dated February 21, 1979, transmitting Large Break Loss of Coolant Accident Analysis using the Westinghouse February 1978 Evaluation Model.

13. XN-73-25, Exxon Nuclear Company, "GAPEX: A Computer Program for Predicting Pellet-to-Cladding Heat Transfer Coefficients," June 1975.

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18. XN-76-40, Exxon Nuclear Power Distribution Control for Pressurized Water Reactors," September 1976.

19. XN-75-27, Exxon Nuclear Company, "Exxon Nuclear Neutronic Design Method for Pressurized Water Reactors," June 1975; Supplement 1, September 1976; Supplement 2, December 1977.

20. XN-CC-26, Rev. 1, Exxon Nuclear Company, "XPIN - The Exxon Nuclear Revised HAMBUR USERS MANUAL," December 1975.

21. WAPD-TM-687, Westinghouse Electric Corporation, "PDQ7 Reference Manual," January 1965.

22. XN-74-5, Revision 1, Exxon Nuclear Company, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTSPWR)," May 1975.

23. US NRC, Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," May 1974.

24. XN-75-41, Exxon Nuclear Company, "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model, Vol. I through III, July - August 1975 and Supplements 1 through 7, August - November 1975.

25. XN-76-27, Exxon Nuclear Company, "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model Update ENC WREM-II," July 1976 and Supplements 1 and 2, September - November 1976.

26. Letter from US NRC (T. A. Ippolito) to Exxon Nuclear Company (W. Nechodom), dated March 30, 1979, transmitting: "Review of ENC WREM-Based Generic ECCS Evaluation Model Update EN WREM-IIA (XN-NF-78-30)."

27. Letter from US NRC (T. A. Ippolito) to Exxon Nuclear Company (W. Nechodom), dated March 28, 1979 transmitting "ENC RELAP4 - EM Updates."

28. US Nuclear Regulatory Commission, "Safety Evaluation Report on ECCS Evaluation Model for Westinghouse Two-Loop Plants," Analysis Branch, Division of Systems Safety, Office of Nuclear Reactor Regulation, November 1977.

29. Letter from Northern States Power Company (L. O. Mayer) to NRC (Director of NRR), dated February 24, 1978.

30. US Nuclear Regulatory Commission, "Safety Evaluation Report on Interim ECCS Evaluation Model for Westinghouse Two-Loop Plants," March 1978.

31. Letter from Rochester Gas and Electric Corporation (L. D. White, Jr.) to NRC (D. L. Ziemann), dated August 7, 1978.

32. Letter from NRC (A. Schwencer) to Northern States Power Company (L. O. Mayer), dated May 18, 1978, transmitting Order for Modification of License for Prairie Island Generating Plant, Units 1 and 2.

33. Northern States Power Company letter (L. O. Mayer) to NRC, dated February 21, 1979, transmitting "Large Break Loss of Coolant Accident Analysis, - February 1978 Evaluation Model."

34. WCAP-9220-P-A, Westinghouse Electric Corporation,"Westinghouse ECCS Evaluation Model, February 1978 version", February 1978.

35. Northern States Power Company letter (L. O. Mayer) to NRC (D. L. Ziemann), dated January 20, 2977, transmitting: "Major Reactor Coolant System Pipe Rupture - October 1975 Evaluation Model."

36. Letter from NRC (G. E. Lear) to Exxon Nuclear Company, Inc. (W. S. Nechodom), dated June 7, 1978.

37. Letter from NRC (T. Ippolito) to Exxon Nuclear Company (W. S. Nechodom), dated March 30, 1978.

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. 35 and 29 to Facility Operating License Nos. DPR-42 and DPR-60, issued to the Northern States Power Company (the licensee), which revised Technical Specifications for operation of Unit Nos. 1 and 2 of the Prairie Island Nuclear Generating Plant (the facilities) located in Goodhue County, Minnesota. The amendments are effective as of their date of issuance.

These amendments change the common station Technical Specifications for the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2 and incorporate changes to the Appendix A Technical Specifications to support operation in Cycles 5 through 8 with reload fuel by the Exxon Nuclear Company.

The requirements of the NRC Order for Modification of License of Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2 dated May 18, 1978 have been satisfied by the submittal dated February 21, 1979 and supplemented on March 30, 1979.

The application for 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

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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 November 22, 1978 (43 F.R. 54706). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

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

For further details with respect to this action, see (1) the submittal dated September 8, 1978 and the application for amendments dated December 29, 1978 and supplemented on January 23 and March 30, 1979, (2) Amendment Nos. 35 and 29 to License Nos. DPR-42 and DPR-60, respectively, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Environmental Conservation Library 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 20th day of April, 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

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