OCT 3 1983

Docket Nos. 50-282
and 50-306

DISTRIBUTION:

Docket File	PMKreutzer-3
NRC PDR	DCDiIanni
L PDR	Gray File +4
NSIC	CBerlinger
ORB#3 Rdg	BSheron
DEisenhut	HRDenton
OELD SECY	RDiggs
CMiles	LJHarmon
EJordan	TBarnhart-8
JTaylor	WJones
DBrinkman	ACRS-10

Mr. D. M. Musolf
Nuclear Support Services Department
Northern States Power Company
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

Dear Mr. Musolf:

The Commission has issued the enclosed Amendment Nos. 66 and 60 to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2 in partial response to your application dated June 24, 1983 as revised July 29, 1983.

The amendments change the limit of the core local heat flux ratio F_Q^N from 2.21 to 2.32, allowing a localized linear heat generating rate increase from 14.31 to 15.02 kW/ft which includes a 1.02 factor for power uncertainty. In addition, the definition of F_Q^N is changed from a neutron flux comparison to a heat flux comparison derived from measured neutron flux and fuel enrichment. These amendments do not consider the increase in peak fuel pellet exposure from 51 to 55 GWD/MTU. This matter will be the subject of separate amendments.

A copy of the Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next regular monthly Federal Register Notice.

Sincerely,

Original signed by C. D. Dilanni

Dominic C. DiIanni, Project Manager
Operating Reactors Branch #3
Division of Licensing

(1) Square Carry
14.15.02 or 15.02
should be 15.02
that what 1st Notice
said that what
EXXON Supp (P.D)

Enclosures:
1. Amendment No. 66 to DPR-42
2. Amendment No. 60 to DPR-60
3. Safety Evaluation

cc w/enclosures:
See next page

ORB#3:DL	ORB#3:DL	ORB#3:DL	CPB	RSB	AD:OR:DL	OELD
PMKreutzer	DCDiIanni/pn	JRMiller	CBerlinger	BSheron	OCLainas	
9/29/83	9/29/83	9/30/83	9/29/83	9/29/83	9/30/83	9/ /83

(2) Clean Check for Pertinent or Comments immediately before Issuance. If any come back to ELD SOS 9/30/83

8310130072 831003
PDR ADOCK 05000282
P PDR

Northern States Power Company

cc:

Gerald Charnoff, Esq.
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N.W.
Washington, D. C. 20036

Ms. Sandra Gardebring
Executive Director
Minnesota Pollution Control Agency
1935 W. County Road B2
Roseville, Minnesota 55113

Mr. Wm. Miller
Goodhue County Auditor
Red Wing, Minnesota 55066

U. S. Environmental Protection Agency
Federal Activities Branch
Region V Office
ATTN: Regional Radiation
Representative
230 South Dearborn Street
Chicago, Illinois 60604

Mr. E. L. Watzl, Plant Manager
Prairie Island Nuclear Generating Plant
Northern States Power Company
Route 2
Welch, Minnesota 55089

Jocelyn F. Olson, Esquire
Special Assistant Attorney General
Minnesota Pollution Control Agency
1935 W. County Road B2
Roseville, Minneosta 55113

U.S. Nuclear Regulatory Commission
Resident Inspectors Office
Route #2, Box 500A
Welch, Minnesota 55089

Regional Administrator
Nuclear Regulatory Commission, Region III
Office of Executive Director for Operations
799 Roosevelt Road
Glen Ellyn, Illinois 60137

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. 66
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 June 24, 1983 as revised July 29, 1983, 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 (1) 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.

8310130076 831003
PDR ADOCK 05000282
P PDR

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 66 , 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

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 3, 1983

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. 60
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 June 24, 1983 as revised July 29, 1983, 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 (1) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 60 , 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

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 3, 1983

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NOS. 66 AND 60 TO FACILITY OPERATING LICENSE

NOS. DPR-42 AND DPR-60

DOCKET NOS. 50-282 AND 50-306

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the area of changes.

Remove	Insert
TS-iv	TS-iv
TS.3.10-1	TS.3.10-1
TS.3.10-2	TS.3.10-2
TS.3.10-9	TS.3.10-9
TS.3.10-11	TS.3.10-11
Figure TS.3.10-5	Figure TS.3.10-5

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° Temperature
3.1-4	Fast Neutron Fluence (E >1 MeV) as a Function of Full Power Service Life
3.1-5	DOSE EQUIVALENT I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity >1.0 μCi/gram DOSE EQUIVALENT I-131
3.9-1	Prairie Island Nuclear Generating Plant Site Boundary for Liquid Effluents
3.9-2	Prairie Island Nuclear Generating Plant Site Boundary for Gaseous Effluents
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
3.10-6	Deviation from Target Flux Difference as a Function of Thermal Power
3.10-7	Normalized Exposure Dependent Function $BU(E_j)$ for Exxon Nuclear Company Fuel
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

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability

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

Objective

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

Specification

A. Shutdown Reactivity

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

B. Power Distribution Limits

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

$$F_Q^N \times 1.03 \times 1.05 \leq (2.32/P)^* \times K(Z) \times BU(E_j)$$

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

where the following definitions apply:

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

(b) Z is the core height location.

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

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

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

*(2.21/P) shall be used for Westinghouse assemblies.

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

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

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

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

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

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

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

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

   $$F_Q^N \text{ (equil)} \times V(Z) \times 1.03 \times 1.05 \leq (2.32/P) \times K(Z) \times BU(E_j)$$

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

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

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

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

      2. Reduce reactor power and the high neutron flux trip setpoint by 1% for each percent that the measured F_Q^N(equil) x 1.03 x 1.05 x V(Z) exceeds the (2.32/P) x K(Z) x $BU(E_j)$ limit.

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

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

$F_Q(Z)$, Height Dependent Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods. The maximum value of $F_Q(Z)$ is 2.32/P for the Prairie Island reactors. This value is restricted further by the K(Z) and $BU(E_j)$ functions described below. The product of these three factors is $F_Q(Z)$.

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

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

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

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

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

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

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

3. The control bank insertion limits are not violated.

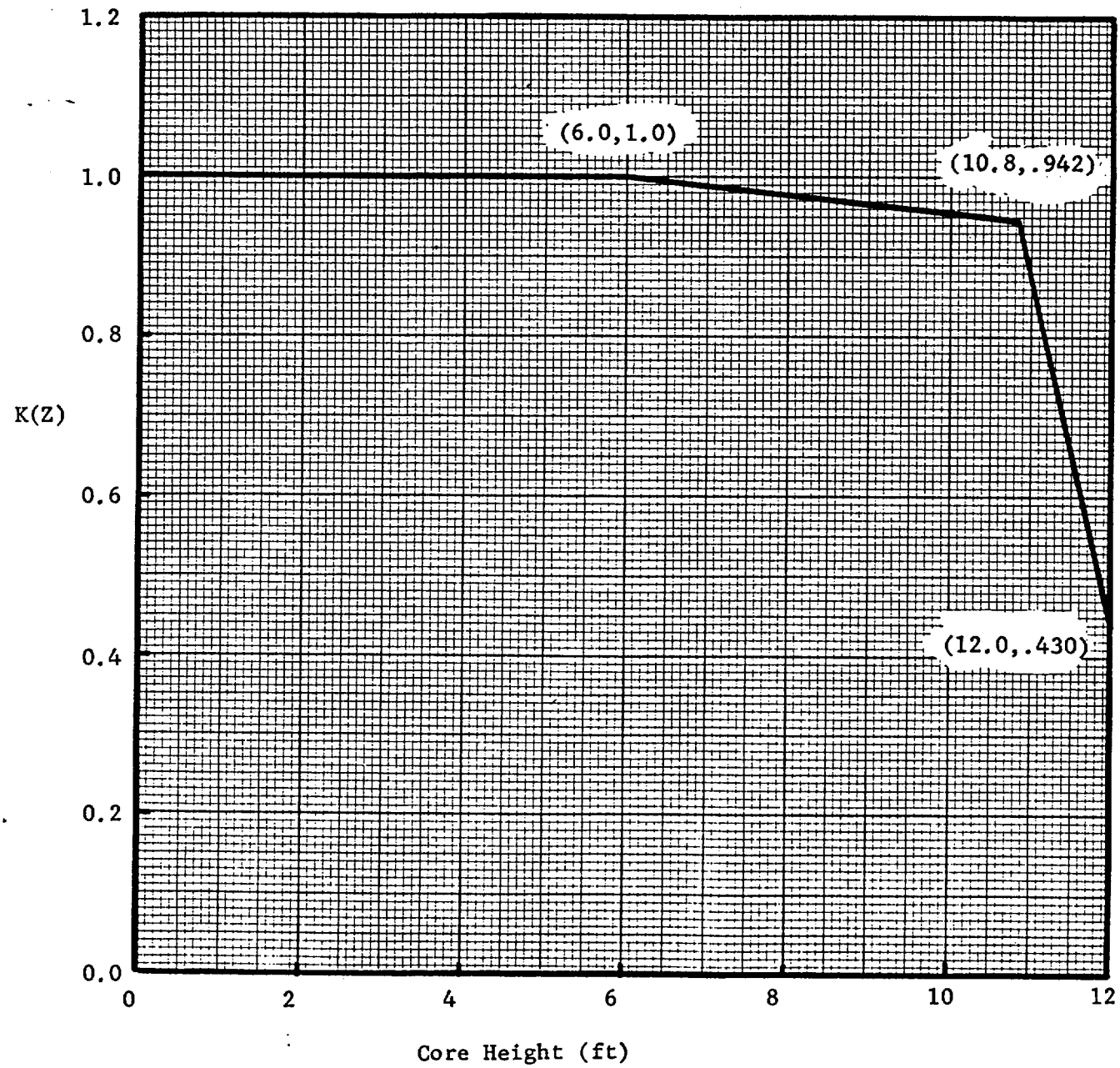
4. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure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

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

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

The technical specifications on power distribution control assure that the $F_Q(Z)$ upper bound envelope of 2.32/P times Figures TS.3.10-5 and TS.3.10-7 is not exceeded and xenon distributions are not developed which at a later time, would cause greater local power peaking even though the flux difference is then within the limits specified by the procedure.

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



HOT CHANNEL FACTOR NORMALIZED

OPERATING ENVELOPE FOR F_Q = 2.32

Prairie Island Unit 1 - Amendment No. ~~35~~, 66
Prairie Island Unit 2 - Amendment No. ~~29~~, 60

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 66 AND 60

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

NORTHERN STATES POWER COMPANY

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

DOCKET NOS. 50-282 AND 50-306

Introduction

By letter dated June 24, 1983 (Ref. 1), Northern States Power Company (NSP) requested changes to the Technical Specifications (TS) for the Prairie Island Nuclear Generating Plant Units Nos. 1 and 2. The proposed TS changes involve the local power hot channel factor F_Q^N limit in the TS 3.10-1, 3.10-2, 3.10-9 and 3.10-11 where (1) a numerical value in the F_Q^N limit expression will be changed from 2.21 to 2.32; and (2) the normalized exposure dependence function Bu(z) curve will be changed to 1.0 for all values of peak pellet exposure from 0 to 55 GWD/MTU. The change involving F_Q^N applies only for fuel manufactured by the Exxon Nuclear Company. The F_Q^N factor would remain the same for the Westinghouse manufactured fuel (i.e., 2.21). In addition, the definition of F_Q^N will be changed from a "neutron flux comparison" to "heat flux comparison". In its letter of July 29, 1983 (Ref. 2), NSP also proposed a change to the axial dependence function K(z) in Figure TS 3.10-5.

These TS change requests are based on a new LOCA analysis performed by the Exxon Nuclear Company for Prairie Island. The analysis results are described in the report XN-NF-83-38 (Ref. 3). The staff evaluation regarding the TS changes follows.

Staff Evaluation

The purpose of limiting condition of operation on F_Q^N is to limit the peak linear power density during normal operation to provide assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded. In order to support the proposed TS changes, the licensee in the XN-XF-83-38 report (Ref. 3) describes the analysis and results performed by Exxon for the Prairie Island Units 1 and 2 for a postulated large break LOCA. The analysis was made with the following conditions:

(1) The double-ended cold leg guillotine break with a discharge coefficient of 0.4: This scenario has been identified in the previous analyses as the most limiting break.

(2) An entire core with ENC TOPROD fuel: With respect to LOCA, the TOPROD fuel design is more limiting than the ENC XN-1 and XN-2 fuels due to the smaller pin diameter and the increased core flow area which reduces core reflood rates in the LOCA analysis.

8310130079 831003
PDR ADOCK 05000282
P PDR

(3) 5% of steam generator tubes being uniformly plugged.

(4) Including 2% power uncertainty resulting in the allowable linear heat generation rate of 15.02 kW/ft corresponding to total power peaking factor of 2.32 and nuclear enthalpy rise factor of 1.55 for the entire fuel exposure.

(5) Maximum peak pellet exposure of 55 GWD/MTU.

The reactor coolant system nodalization was modeled in accordance with an approved ENC ECCS modeling described in XN-NF-77-25(A) (Ref. 4) for a 2-loop Westinghouse PWR with dry containment. The LOCA analysis was performed using ENC's EXEM/PWR ECCS evaluation model (Ref. 5). This evaluation model uses the following computer codes:

(1) RELAP-EM (Ref. 6) for the system blowdown and hot channel blowdown calculations;

(2) CONTEMPT-LT/22 as modified in CSB G-1 (Ref. 7) for the containment back pressure calculation;

(3) REFLEX (Ref. 8) for system reflood calculation;

(4) RODEX2 (Ref. 9) for initial fuel rod stored energy, fission gas release and internal gas inventory calculations; and

(5) TOODEE2 (Ref. 10) for the calculation of final fuel rod heatup.

The RELAP-EM, CONTEMPT-LT/22, REFLEX and TOODEE2 codes have previously been approved by the NRC. The RODEX2 code has recently been reviewed by the staff and has been found acceptable for use in the LOCA initial stored energy and rod pressure calculations (formal SER for RODEX2 is being prepared). In addition, an approved cladding swelling and rupture model described in XN-NF-82-07, Revision 1 (Ref. 11) was used in the calculation of the cladding rupture, strain and flow blockage in the ENC's EXEM/PWR ECCS evaluation model and, therefore this part of the analysis is acceptable. The overall EXEM/PWR ECCS evaluation model is still under review by the staff. However, the review has progressed to the point to conclude that the evaluation model is acceptable for the Prairie Island LOCA analyses.

The LOCA analysis was performed with two analyses: from the beginning of life to 15 GWD/MTU and from 15 GWD/MTU to 55 GWD/MTU. The most limiting fuel conditions in the respective exposure ranges were used in the analysis. The combination of the highest stored energy, rod pressure and decay power was used to bound the LOCA-ECCS analysis over the exposure ranges. The results of these analyses have shown the maximum peak cladding temperatures to be 2091°F and 2142°F, respectively, for the fuel exposure ranges prior to and after 15 GWD/MTU. The local metal-water reactions are 4.68% and 5.6% for the two fuel exposure ranges, and total core metal-water

reaction is less than 1%. These peak clad temperatures (PCT) and metal-water reaction are within the limit imposed in 10 CFR 50.46, "Acceptance criteria for ECCS for light water nuclear power reactors". Therefore, the proposed TS change in F_Q^N limit is acceptable.

The staff also reviewed the change to Figure TS 3.10-5 proposed in the July 29, 1983 letter. This curve is a core axial dependent function, usually referred to as the K(z) curve. The K(z) curve should be revised whenever a large or small break LOCA analysis is performed with a changed peaking factor. We have independently determined that the curve submitted in the July 29, 1983 letter (not the curve submitted in the June 24, 1983 letter) is correct. This proposed TS change is therefore acceptable.

With regard to the proposed change in the definition of F_Q^N from a "neutron flux comparison" to a "heat flux comparison", we find that the heat flux comparison is more appropriate since the concern of PCT in the LOCA analysis is directly related to the thermal power. Therefore the proposed change is acceptable.

In conclusion, the staff has reviewed the NSP proposed TS changes regarding the F_Q^N limiting condition of operation and found them acceptable for only the Exxon manfactured fuel. This acceptance is based on the LOCA analysis results provided by the licensee using the approved computer codes, and the staff's review of the ENC's EXEM/PWR ECCS evaluation model used in this analysis has progressed to the point where the evaluation model is acceptable for the Prairie Island LOCA analysis. Therefore, although a reanalysis could be required if the final evaluation of the EXEM/PWR ECCS model by the staff identified any problems or condition, we have concluded that the proposed change in the Prairie Island Units 1 and 2 Technical Specification is acceptable.

Since the effects of the increase in the fuel exposure to 55 GWD/MTU on the off-site doses have not been addressed here, the staff's acceptance does not include the limitation of fuel burnup to 55 GWD/MTU.

Since the proposed TS change is based on the LOCA analysis results showing that the acceptance criteria imposed in 10 CFR 50.46 will not be violated, the staff has determined that these TS changes will not involve a significant reduction in safety margin.

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 the 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: October 3, 1983

Principal Contributors:
Yihsiung Hsii
Marvin Dunenfeld
Shih-Liang Wu

References

1. Letter from D. Musolf (NSP) to NRC, "Prairie Island Nuclear Generating Plant, Docket Nos. 50-282, 50-306, License Amendment Request dated June 24, 1983, LOCA Analysis", June 24, 1983.

2. Letter from D. Musolf (NSP) to NRC, "Revision No. 1 to License Amendment Request dated June 24, 1983 LOCA Analysis", Docket Nos. 50-282, 50-306, July 29, 1983.

3. XN-NF-83-38, "Prairie Island Units 1 and 2 Limiting Break LOCA-ECCS Analysis Using EXEM/PWR", Exxon Nuclear Company, May 1983. Exhibit C To NSP June 24, 1983 letter.

4. XN-NF-77-25(A), "ENC ECCS Evaluation of a 2-loop Westinghouse PWR with Dry Containment Using the ENC WREM-II ECCS Model - Large Break Example Problem", September 1978.

5. XN-NF-82-20, "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates", Revision 1, August 1982; Supplement 1, March 1982; Supplement 2, March 1982.

6. Letter from T. A. Ippolito (NRC) to W. S. Nechodam (ENC), "SER for ENC RELAP4-EM Update", March 1979.

7. NRC Branch Technical Position CSP G-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation".

8. Memorandum from R. L. Tedesco to B. K. Grimes, "Review of Exxon Nuclear Company Topical Report (ENC WREM-Based Generic PWR ECCS Evaluation Model Update ENC WREM-IIA), TAC-10691", March 23, 1979.

9. XN-NF-81-58(P), Revision 2: Fuel Rod Thermal-Mechanical Response Evaluation Model", February 1983.

10. NUREG-75/057, "TOODEE2: A Two-Dimensional Time Dependent Fuel Element Thermal Analysis Program", May 1975.

11. XN-NF-82-07(P), Revision 1, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model", August 1982.