**NSP**

**NORTHERN STATES POWER COMPANY**

MINNEAPOLIS, MINNESOTA 55401

December 29, 1978

Director of Nuclear Reactor Regulation
U S Nuclear Regulatory Commission
Washington, DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket No. 50-282 License No. DPR-42
50-306 DPR-60

License Amendment Request Dated December 29, 1978

Technical Specifications Associated with Use of
Exxon Nuclear Company Fuel in Prairie Island Units 1 and 2

Reference: (a) Letter, L O Mayer (NSP) to Director of Nuclear Reactor Regulation (NRC) dated September 8, 1978

Attached are three originals and 37 conformed copies of a request for a change of Technical Specifications, Appendix A, of Operating Licenses DPR-42 and 60. Also attached is one copy of the license amendment class determination and a check in the amount of $12700.00 for the amendment fee.

This submittal proposes changes in definition, reactor core design features, and limiting conditions for operation in the areas of minimum conditions for criticality and power distribution limits. These changes will allow operation of Prairie Island Unit 1 and 2 with Exxon Nuclear Company (ENC) Reload Fuel commencing with Cycle 5. The changes in design features and limiting conditions for operation had been identified previously in reference (a). Supporting documentation for this amendment is included as Exhibits C and D. In accordance with 10CFR 2.790(a)(4), ENC report, XN-NF-78-34, will be transmitted under separate cover due to the proprietary nature of the document. ENC affidavit will be included with that transmittal.

To assure timely startup of the Unit 1 reactor, issuance of this amendment is requested by no later than April 1, 1978.

This revision does not include Technical Specifications implementing ENC calculated exposure dependent $F_Q$ limits as described in Reference (a). Exxon Nuclear Company has determined that there will be no effect on Cycle 5 operation. A separate license amendment is expected to be forwarded in

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February, 1979 when Exxon will have completed documentation to support exposure dependent $F_Q$ limits for the Prairie Island units. This documentation will be similar to that previously submitted to the NRC to support exposure dependent $F_Q$ limits for Docket No. 50-315. Issue of the forthcoming Technical Specification change dealing with exposure dependent $F_Q$ limits will not be required until December, 1979.

Yours very truly,

L. O. Mayer

L O Mayer, PE
Manager of Nuclear Support Services

LOM/JAG/ak

cc: J G Keppler
G Charnoff
MPCA - Attn: J W Ferman

Attachments

UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT    Docket No. 50-282
50-306

REQUEST FOR AMENDMENT TO
OPERATING LICENSE NO. DPR-42 & DPR-60

(License Amendment Request Dated December 29, 1978)

Northern States Power Company, a Minnesota corporation, requests authorization for changes to the Technical Specifications as shown on the attachments labeled Exhibit A and Exhibit B. Exhibit A describes the proposed changes along with reasons for the change. Exhibit B is a set of Technical Specification pages incorporating the proposed changes. Exhibits C and D contain supporting documentation for the proposed changes.

This request contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By L J Wachter
L J Wachter
Vice President, Power Production
& System Operation

On this 29th day of December, 1978, before me a notary public in and for said County, personally appeared L J Wachter, Vice President, Power Production and System Operation, and being first duly sworn acknowledged that he is authorized to execute this document on behalf of Northern States Power Company, that he knows the contents thereof and that to the best of his knowledge, information and belief, the statements made in it are true and that it is not interposed for delay.

Denise E. Halvorson

DENISE E. HALVORSON
NOTARY PUBLIC - MINNESOTA
HENNEPIN COUNTY
My Commission Expires Oct. 10, 1981

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EXHIBIT A

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

License Amendment Request Dated December 29, 1978

PROPOSED CHANGES TO THE TECHNICAL SPECIFICATIONS
APPENDIX A OF OPERATING LICENSE
DPR-42 & DPR-60

Pursuant to 10CFR50.59, the holders of Operating Licenses DPR-42 and DPR-60 hereby propose the following changes to Appendix A, Technical Specifications:

1. List of Figures

PROPOSED CHANGE

Change the title of Figure TS.3.10-6 and add Figure TS.3.10-8 to the List of Figures on page TS-iv.

REASON FOR CHANGE

These figures are required based on the power distribution limit method of control PDC 2 specified by ENC, the fuel manufacturer.

SAFETY ANALYSIS

These changes have no safety significance and are administrative in nature.

2. Interim Fuel Limits Definition

PROPOSED CHANGE

Delete definition 1.S. Interim Fuel Limits. Reletter 1.T. Startup Operation and 1.U. Fire Suppression Water System to maintain alphabetical order of the definitions.

REASON FOR CHANGE

The interim fuel limits definition specifies (1) the power distributions to be used in the loss of coolant accident analyses and (2) the limit on Unit 1 Cycle 1 fuel residence time. Neither of these limits is applicable.

SAFETY ANALYSIS

Deletion of the definition of "Interim Fuel Limits" is warranted because the 1971 Policy Statement and 1972 Technical Report listed have been superseded by the 10CFR50 Appendix K criteria and the Section 3.10 "Power Distribution Limits" limiting conditions for operation currently in the Technical Specifications. This change will not affect the health and safety of the public since the references stated have been superseded.

3. Minimum Conditions for Criticality

PROPOSED CHANGE

Change specification 3.1.F.1 and the associated basis to read as shown in Exhibit B, Pages TS.3.1-17 and 18.

REASON FOR CHANGE

Previous safety analyses for the Prairie Island units (Cycles 1 through 4) were conducted assuming a non-positive moderator temperature coefficient. In order to obtain better fuel utilization, higher fuel loadings (within Technical Specification limits) may be required, resulting in higher RCS boron concentrations at the beginning of cycle. Under these conditions, a positive moderator temperature coefficient may exist. However, the core has been designed so that the isothermal temperature coefficient (the sum of the moderator and fuel temperature coefficient) is limited to being negative.

This change also deletes references that are no longer applicable.

SAFETY EVALUATION

Exhibit C is submitted to support this proposed change.

The isothermal coefficient is a more practical parameter for which to establish a Technical Specification limit because it is empirically determined during startup tests. The moderator temperature coefficient, on the other hand, is derived as the difference between the isothermal coefficient and the calculated fuel temperature coefficient at the test conditions. The isothermal temperature coefficient is negative to ensure that there is negative reactivity feedback on a power excursion.

XN-NF-78-35 contains the results of transient analyses that conservatively assume the most limiting isothermal temperature coefficient. These analyses assumed a more conservative $F_Q$ (2.32) than that being proposed (2.21) and moderator and fuel temperature coefficient values as listed in Table 4.1 (Page 69) of XN-NF-78-35. XN-NF-78-47 (Section 5.1.3) describes the moderator temperature coefficient considerations and predicts the hot zero power all rods out isothermal coefficient to be negative, which will be confirmed in the startup testing program described later. Other conditions e.g., higher power or partial insertion of rods would cause the isothermal coefficient to have a more negative value.

XN-NF-78-35, indicates MTC will be positive at a low power, approaches zero at about 70% power, and is calculated to be negative at full power. The transient analyses conducted envelope these predicted core conditions.

Values of the trip setpoints used in the analyses (XN-NF-78-35 Table 2.2 Page 10) are as or more conservative than the Technical Specifications or actual plant setpoints.

The change requested is consistent with previously established NRC acceptance of allowing plant operation with a positive moderator temperature coefficient, e.g., Amendment 40 to DPR-36, dated August 18, 1978.

As with any startup after a refueling, testing is appropriate to verify design values. The testing program for this reload will include the following as appropriate for refueling with fuel of a different supplier.

1. Verification of proper core loading.
2. Rod drop times (100% flow, RCS temperature >500$^{o}$F).
3. Source, and intermediate range calibrations (including functional tests, plateau curves, compensating voltage check as appropriate).
4. Rod position indication calibration.
5. Boron endpoint measurements (All rods out, Control Bank D in, Control Banks C and D in, all Control Banks in).
6. Boron worth.
7. Rod worths (Differential and Integral, Control Banks A, B, C, and D).
8. Isothermal Temperature Coefficient measurements (All rods out, Control Bank D in, Control Banks C and D in).
9. Flux maps (Hot Zero Power-all rods out, 48%, 100%.)
10. Axial offset power range calibration at 90% after 3 days of fuel preconditioning at 100% (3 flux maps at different power distributions).
11. Plant Heat Balance (35%, 48%, 90%).

These tests are conducted by the Prairie Island Nuclear Engineering staff using procedures that have been reviewed by the Plant Operations Committee and approved by the Plant Superintendent-Engineering and Radiation Protection.

Descriptions of many of these tests are included in the Unit 1 and 2 startup test reports previously forwarded to the USAEC/USNRC. Forty (40) copies of each of these reports were transmitted, as follows:

Unit 1 - L O Mayer to E Case, dated October 31, 1974
Unit 2 - L O Mayer to A Giambusso, dated May 15, 1975

These tests are consistent with those conducted during a startup after any refueling, regardless of fuel type. The procedures used and results for startup tests cycles 1 through 4 have been reviewed by Region III I&E personnel. Selection of the tests to be conducted is based on evaluation and consensus by corporate office licensing and core management and analysis personnel and plant nuclear engineering staff personnel.

In accordance with Technical Specification 6.7.A.1, a startup report will be filed with the NRC within 90 days after startup.

4. Specification 3.10.B.1 Power Distribution Limits

PROPOSED CHANGE

a. Change the existing section to read as shown on pages TS.3.10-1, 1A, -2, -7A, -8, and -9 and Figure TS.3.10-6.

b. Add Figure TS.3.10-8 of Exhibit B.

REASON FOR CHANGE

The $F_Q^N$ values and footnote 1 statement were imposed by NRC Order dated May 18, 1978. The $F_Q$ for both Prairie Island units was restricted to 2.24 if accumulator modifications were performed and 2.21 if those modifications were not performed.

The $F_Q^N$ and $F_{\Delta H}^N$ limits are no longer applicable based on subsequent reanalysis. In addition, ENC has adopted Power Distribution Control Phase 2 which imposes restrictions on $F_Q^N$ during target axial flux difference determination. These changes are required to reflect the different methodology in nuclear design.

SAFETY ANALYSIS

Westinghouse LOCA analyses for both units utilized the "February, 1978" model approved by the NRC. The analytical results will be provided under separate cover.

The ECCS analyses and results are contained in XN-NF-78-46. The analytical results presented in Section 2.4 and Table 2.2 show that the acceptance criteria of 10CFR50.46 would be met for an $F_Q$ of 2.21 for a core loaded with Exxon fuel. The analyses presented are applicable for both Prairie Island Units 1 and 2 using the Exxon fuel of the same design as that used in Unit 1, Cycle 5. This approach of applying the same ECCS analyses to both units is consistent with that heretofore employed with Westinghouse fuel and analyses. Containment and RCS parameters are identical for both units.

The ECCS analyses were conducted using the WREM IIA model, described in XN-NF-78-30, previously forwarded to the NRC by G F Owsley (ENC) to T A Ippolito (NRC) dated August 15, 1978. In addition, a calculation to evaluate the impact of upper plenum injection is presented. This UPI analysis is based on the interim model developed by the NRC staff and modified by Westinghouse. The interim UPI model changes and results are described in Section 3.1 of XN-NF-78-46. In summary, the calculations show that the peak clad temperature for a 0.4 DECLG will be 2198$^{\circ}$F (including a 1$^{\circ}$F penalty for the interim UPI model).

The rod bow penalty has heretofore been established for the Westinghouse fuel. The basis for this penalty not being applicable to the Exxon fuel is described in Section 6.3 of XN-NF-78-47.

The limit $F_Q^N \leq (2.145/P) \times (K(Z)/V(Z))$ is imposed based on the power distribution control phase 2 method applied to Prairie Island (Section 5.1.1 of XN-NF-78-47). The basis for PDC 2 procedures is described in XN-NF-77-57 "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors Phase 2".

The revised Figure TS.3.10-6 is consistent with the methodology explained in Specification 3.10.B.6.

The new figure (Figure TS.3.10-8) was added to support application of the new $F_Q^N$ limit imposed by Specification 3.10.B.1.b.

The bases were changed to provide the results of the Exxon analyses to the operator responsible for I control.

5. Specification 5.3.A Reactor Core Design Features

PROPOSED CHANGES

a. Change subsection 2 as shown in Exhibit B and delete subsection 3.

b. Change the highest enrichment to 3.5 rather than 3.4 weight percent and delete reference (2) of the end of Specification 5.3.A.2.

c. Delete the existing subsection 4 that refers to the burnable poison rods incorporated in the initial core and their construction.

d. Change the existing subsection 5 to become subsection 3.

e. Delete subsection 6 and carry forward the page TS.5.3-2 material.

REASON FOR CHANGE

a. Reference to the three fuel enrichments used in the initial core is no longer applicable. This change maintains consistency by referring only to reload cores which is appropriate for Prairie Island.

b. To obtain longer fuel cycles, higher uranium 235 enrichment loadings will be required. In addition, this change would make Section 5.3.A.2 consistent with Section 5.6.A.

c. Burnable poisons have only been used in the Prairie Island units as needed primarily to control the value of the moderator temperature coefficient at beginning of life.

d. The subsection 4 should be deleted for two reasons:

   1. That section is no longer applicable. It refers to the initial core burnable poison rods. Starting with Cycle 5, gadolinium oxide will be used as the burnable poison.

   2. Current technical specification regulatory guidance on design features for both PWR's and BWR's contains no reference to burnable poisons.

e. The requirement of 5.3.A.6 was superseded by Section 2.B.(3) of the operating licenses DPR-42 and -60.

SAFETY ANALYSES

a. A reload core typically consists of 4 or more regions of fuel depending on the individual licensee's plan for optimum fuel utilization. The typical burnup history for the 121 fuel assemblies at the beginning of a cycle is:

| | |
|---|---|
| 1 assembly | - 3 cycles exposure (~27 GWD/MTU) |
| 40 assemblies | - 2 cycles exposure (~23 GWD/MTU) |
| 40 assemblies | - 1 cycle exposure ( ~10 GWD/MTU) |
| 40 assemblies | - No exposure |

Thus the previous description is no longer applicable. This deletion should have no effect on the health and safety of the public.

b. For each reload, separate safety analyses are conducted to assure that operation with reload fuel of the enrichment selected will be in compliance with Technical Specifications and NRC regulations. Thus the change from 3.4 to 3.5 w/o should have no effect on public health and safety. In addition the 3.5 w/o selected is based on the criticality analyses conducted for the Prairie Island Spent Fuel Pit Expansion Hearings. The 3.5 w/o limit is also specified in Section 5.6.A.

c. Burnable poisons may be incorporated into the relo d core. These may be of the following designs -

   1. Burnable poison rod assemblies (BPRA's) consisting of 8, 12 or 16 rods of borosilicate glass clad with stainless steel.

   2. Gadolinium oxide dispersion in a uranium dioxide matrix.

Safety analyses for Cycle 1 considered use of the BPRA design as described in the plant FSAR. Use of the gadolinia design is described in XN-NF-78-34 and XN-NF-78-47.

The Exxon reload fuel assemblies will be loaded into the outer core locations. Only 64 burnable poison rods will be used, distributed evenly among 8 fuel assemblies as shown in Figures 3.1 and A.3 of XN-NF-78-47. Figure A2 of that document shows that gadolinium burnup should be completed by 5500 MWD/MTU pin cell exposure. This contrasts to the borosilicate glass rods initially loaded in the Prairie Island Units, which did not burn up completely within the first cycle ($\sim$17000 MWD/MTU) and were loaded into all three regions.

Supporting information for use of gadolinia is contained in Sections 5.3, 6.4, and Appendix A of XN-NF-78-47 and Exxon report XN-NF-78-34 on fuel design.

Use of the burnable poison uniformly dispersed in the fuel rod eliminates the disadvantages associated with the BPRA design. The main disadvantage of the BPRA design is that many difficulties have been experienced with the tool used to remove these devices from fuel assemblies. These problems can have a significant impact on refueling outage critical path time. Replacement of the discrete burnable poison rods with gadolinium oxide dispersed in the fuel should allow control of the moderator temperature coefficient while minimizing localized disturbances in power distribution.

This change also eliminates references that are no longer applicable and corrects a previous reference error.

EXHIBIT B

License Amendment Request dated December 29, 1978

Exhibit B consists of revised pages of Appendix A Technical Specifications as listed below:

Pages

TS-iv
TS.1-6
TS.3.1-17
TS.3.1-18
TS.3.10-1
TS.3.10-1A
TS.3.10-2
TS.3.10-7A
TS.3.10-8
TS.3.10-9
Figure TS.3.10-6
Figure TS.3.10-8
TS.5.3-1

The following Appendix A Technical Specification page would be deleted upon approval of this request:

TS.5.3-2

# APPENDIX A TECHNICAL SPECIFICATIONS

## LIST OF FIGURES

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 $140^{\circ}F$.

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 $200^{\circ}F$, 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

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

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

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

b. In addition, $F_Q^N$ (Z) shall be measured at the target flux difference once each effective full power quarter and must meet the following limit for a unit with Exxon fuel:

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

where:

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

(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

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 either of the following two actions shall be taken:

   1. Within 24 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 0 percent at the end of the cycle life.

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

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

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

   a. The indicated axial flux difference may deviate from its ± 5% target band for a maximum of one* hour (cumulative) in any 24-hour period provided 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. The rod bow penalties are not applicable for Exxon fuel based on independent measurements of rod-to-rod spacings for interior and peripheral rod bows of spent fuel similar in design to that used in the Prairie Island core. The plant technical specification includes a total nuclear peaking augmentation factor of 1.0815 (product of 1.03x1.05) in the calculation of ECCS safety limits. This factor is adequate to accommodate nuclear augmentation due to rod bow in a limiting assembly to 28,150 MWD/MTU. Fuel assembly exposures above this are expected to be operating well below the LOCA limits due to reduction of assembly reactivity. Thus no additional penalty due to rod bow needs to be applied to calculation of the LOCA limits.

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 instrumentation error.

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

3. The control bank insertion limits are not violated.

4. The part length control rods are not inserted.

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

The permitted relaxation in $F^N_{\Delta H}$ and $F^N_Q$ allows for radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 5 are observed, these hot channel factor limits are met. In specification 3.10 $F^N_Q$ is arbitrarily limited for $P \leq 0.5$ (except for low power physics tests).

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

The technical specifications on power distribution control assure that the $F^N_Q$ upper bound envelope of 2.145 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 $\pm 5$ percent $\Delta$ 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.

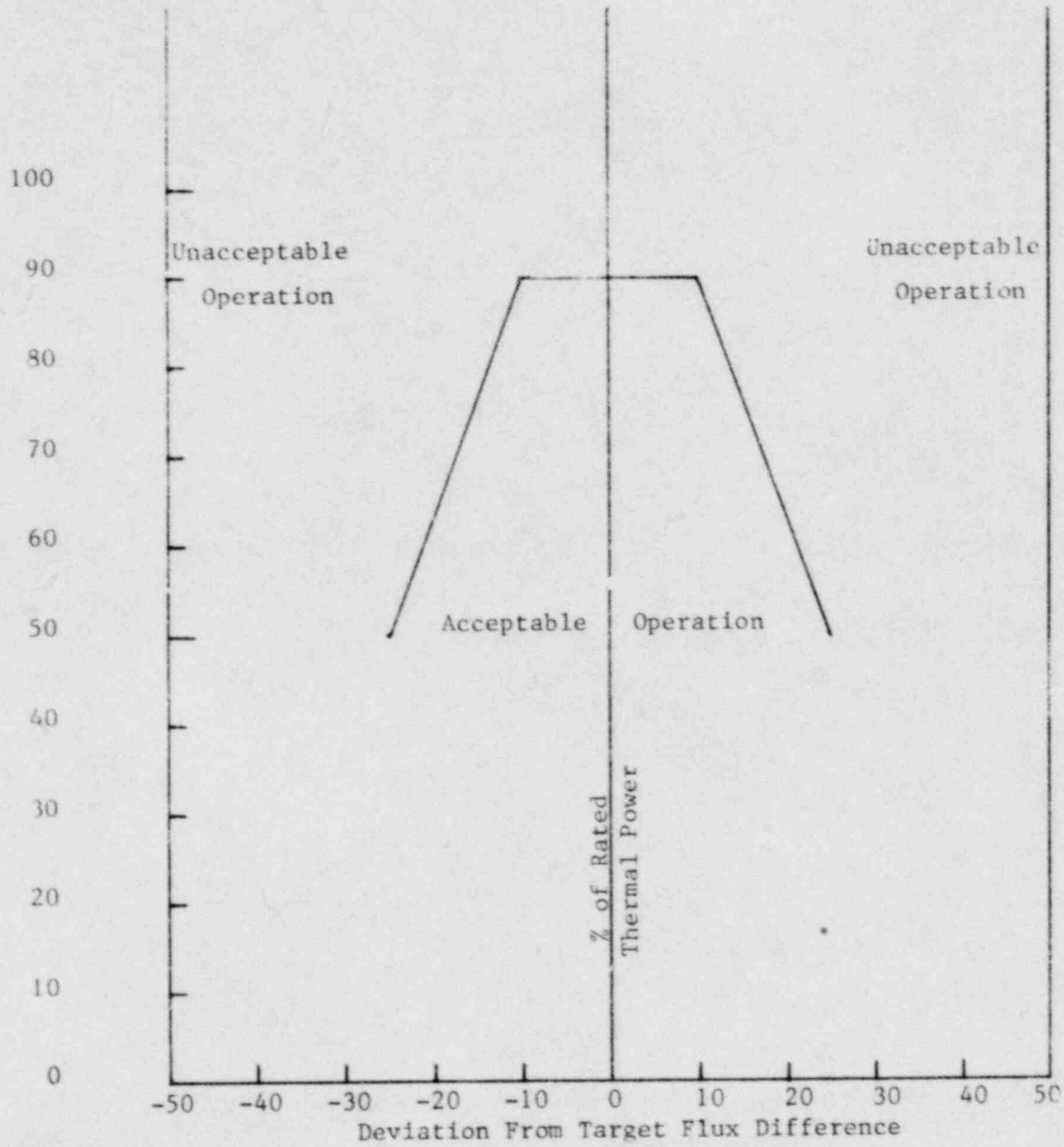


Figure TS.3.10-6 DEVIATION FROM TARGET FLUX DIFFERENCE AS A FUNCTION OF THERMAL POWER

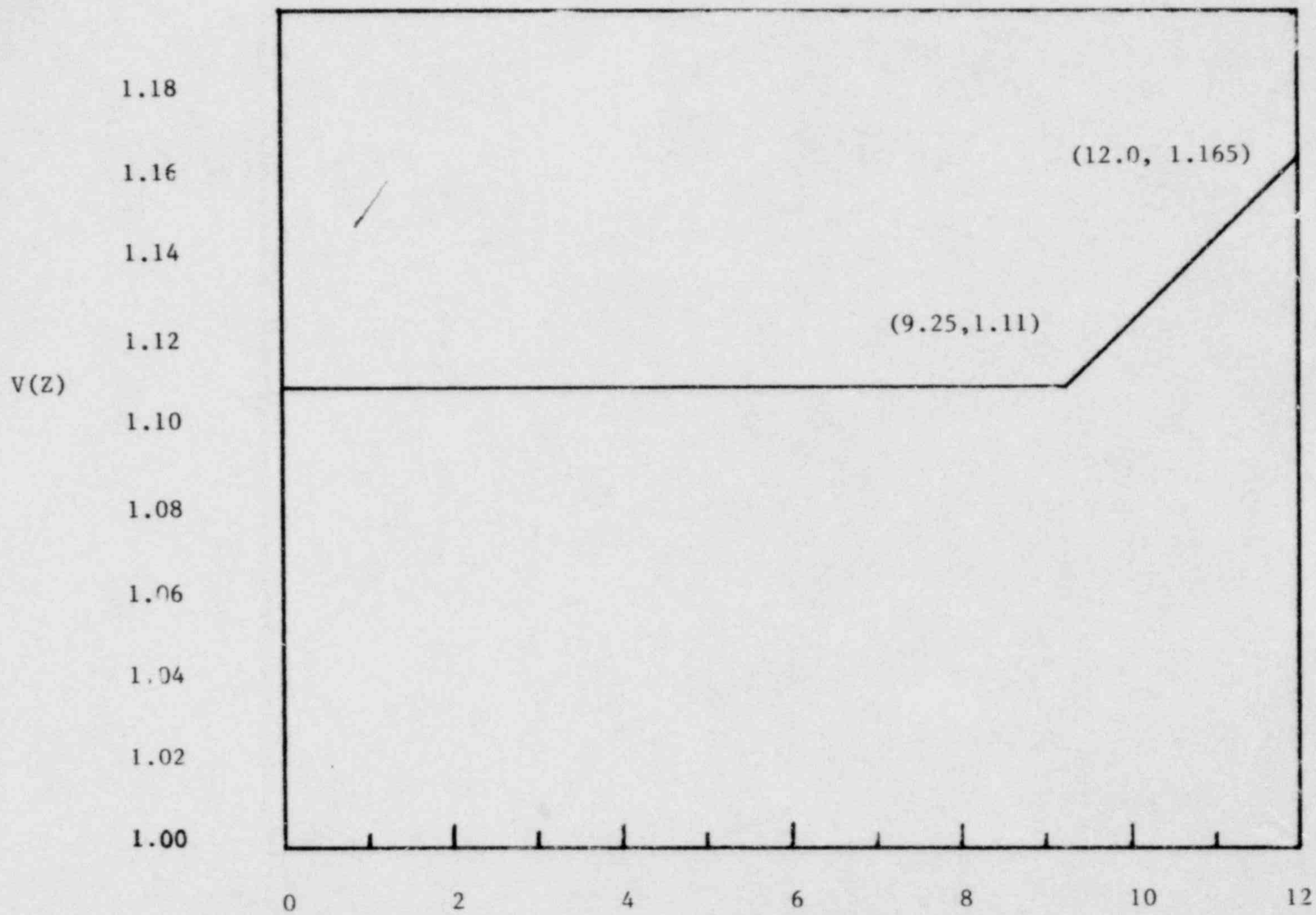


Figure TS.3.10-8 V(Z) as a function of Core Height

REV

## 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 for the 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

EXHIBIT C

License Amendment Request dated December 29, 1978

Exhibit C consists of the following Exxon Nuclear Company reports:

| | |
|---|---|
| XN-NF-78-35 | "Plant Transient Analysis for the Prairie Island Nuclear Power Plant Units 1 and 2". |
| XN-NF-78-46 | "ECCS Large Break Spectrum Analysis for Prairie Island Unit 1 using ENC WREM-IIA PWR Evaluation Model". |
| XN-NF-78-47 | "Prairie Island Unit 1 Nuclear Plant Cycle 5 Safety Analysis Report". |

EXHIBIT D

License Amendment Request dated December 29, 1978

Exhibit D contains the list of errata for the Exxon Documents included n Exhibit C.

| Document | Page | Comment |
|---|---|---|
| XN-NF-78-35 | 10 | Delete the line referring to "b) 3.32 ft steam generator level and 25 sec". |
| | 69 | The reference cycle values for moderator temperature coefficient should be corrected to read: |
| | 73 | Document number (Upper Right Hand Corner) should read XN-NF-78-35. |
| XN-NF-78-47 | 25 | 2nd paragraph, line 4 should read "... $28.15 \times 10^3$ MWD/MTU..." not "$28.5 \times 10^3$ KWD/MTU". |

| | BOC | EOC |
|---|---|---|
| Inactive Loop Startup | | -400 |
| Excessive Feedwater | 0 and | -400 |
| Excessive Load Increase | 0 and | -400 |

where the units are $(\Delta\rho/F) \times 10^6$