

EXHIBIT A

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

License Amendment Request Dated May 6, 1980

PROPOSED CHANGES TO THE TECHNICAL SPECIFICATIONS
APPENDIX A OF OPERATING LICENSES
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. SI Actuation Logic (Specification 3.5)

Proposed Changes

A. Page 3.5-2

- (1) Change second paragraph, 3rd line, to read
 ... "pressurizer pressure signal. These conditions would
 accompany the..."
- (2) At the bottom of the page, delete the footnote +.

B. Table 3.5-1

- (1) Delete Functional Unit 3+
- (2) Change Function Unit "3.a" to "3"
- (3) Delete the last footnote "+" at the bottom of the page

C. Table 3.5-3

- (1) Delete Function Unit 1d.
- (2) Change Functional Unit 1e. to 1d.
- (3) Delete the last footnote "+" at the bottom of the page.

Reason for Change

The existing page TS 3.5-2 and Tables 3.5-1 and 3.5-3 refer to the original pressurizer coincident low pressure and level SI actuation logic. This logic was modified in accordance with previously issued NRC License Amendments 36 (DPR-42) and 30 (DPR-60) dated May 1, 1979. These amendments authorized modification of the SI Actuation Logic to a 2-out-of-3 low pressurizer pressure coincidence only. In the proposed amendment dated April 26, 1979 we stated that we intended to submit a specification revision to eliminate references to the pressurizer coincident level/pressure logic after completion of the logic modification on the two Prairie Island units. These logic changes were completed on May 2 on Unit 1 and on May 8 on Unit 2. Thus this editorial change may now be made.

Safety Analysis

This change has no safety implication since the change is strictly editorial in nature.

2. Control Rod and Power Distribution Limits (Specification 3.10)

A. Introduction

Proposed Changes

Pages TS.3.10-1 through 13A and Figure TS.3.10-7 have been changed as shown in Exhibit B, pages TS.3.10-1 through 17 and Figure TS.3.10-7. The changes involve primarily:

- (a) power distribution control requirements, involving rod bow penalties, exposure dependence of F_O^N , and update of the power distribution control phase 2 methodology.
- (b) editorial changes incorporating consistent terminology, eliminating reference to part length rods, regrouping of the limiting conditions for operation and bases, and inclusion of more detailed information in the bases that is useful to the operator.

The following is a summary of power distribution control changes:

The rod bow related penalties for each fuel type, ($F_{\Delta H}^N$ for Westinghouse, F_O^N for Exxon Nuclear) would be removed based on documentation supplied by each vendor to the NRC staff. The Westinghouse data has been reviewed; the ENC data is under review. These changes are described in Sections 2B and 2C, respectively.

Exposure dependent F_O limits are proposed to conform to 10CFR 50 requirements. These changes are discussed in Section 2D.

The ENC distribution control phase 2 method technical specification requirements have been included based on further recommendations made by the fuel supplier (ENC). The changes are consistent with generic discussions held between the NRC Staff and ENC. These changes are described in Section 2E.

With regard to the editorial changes, throughout Section 3.10, wording has been simplified. The bases sequence has been organized to agree with the order of the limiting conditions for operation.

The following is a listing of changes in each section of specification 3.10 by section (existing number). For editorial changes, the reason for change is included with the proposed change. For the changes involving power distribution limits that are not editorial, sections 2B through 2E provide the proposed change, reason for change, and the safety evaluation. Editorial changes are indicated by an asterisk(*).

Proposed Changes

3.10A Shutdown Reactivity

Delete reference to the part length rods since these are no longer installed at Prairie Island (Reference 13).

3.10B Power Distribution Limits

Numbers specified are consistent with current Technical Specifications.

1. (a) Change the F_Q^N limits for $P > 0.5$ and $P < 0.5$ into a single limit, incorporating the F_Q^N requirement for $P < 0.5$ into the definition of P .
- (b) Simplify the F_Q^N and F_H^N limits to be a comparison of the measured value (including the engineering, measurement, and $V(Z)$ factors as appropriate) to the analytically determined limit (including burnup and axial dependence considerations).
- (c) Move measurement frequencies to section 2.
- (d) Eliminate the rod bow penalties (F_Q^N for ENC fuel and $F_{\Delta H}^N$ for Westinghouse fuel).
- (e) Revise the definitions to include all factors involved in the limit.
2. (a) The measurement errors previously in this section are in section 1. Measurement requirements are specified in this section.
- (b) Measurement frequencies are consistent with Reference 3.
- (c) Incorporate requirements previously in Section 3.10.B.2 (b) and (c) into 3.10.B.3(a) through (d). 3(b), (c), (d) have been modified to agree with Reference 3.
3. Delete to be consistent with Reference 3.
- *4. Change "Item" to "Specifications". Change "...the number of operable excore channels required by TS.3.5..." to "three operable excore channels" to make this specification more straightforward.
- *5. Change 5 as shown in Exhibit B to use similar phraseology as Reference 12.
- *6. (a) Eliminate unnecessary verbage describing the points of the envelope in Figure TS.3.10-6.
- * (b) In (c), change "...the number of operable excore channels required by B.3.5..." to "three operable excore channels" to simplify the specification.

- *7. In b change wording regarding excore channels to be consistent with 4 and 6(c) above.
- *8. Change the wording to be similar to that used in Reference 12.

3.10.C Quadrant Power Tilt Limits*

- 1. In 1.b. change "for every percent of quadrant power tilt above 1.0 " to "for every percent that the quadrant power tilt ratio exceeds 1.0". This change eliminates confusing wording. Wording is similar to that used in Reference 12.

3.10.D Rod Insertion Limits (*)

Delete 3.10.D.2 and footnotes 2 and 3 in this section. Renumber 3.10.D.3 through 5. Eliminate all reference to the part length control rods since these have been removed from the Prairie Island reactors. In the new 3.10.D.4 change, the statement "...for the low power margin..." to read "...for low power physics testing..." to clarify the intent of the specification.

3.10.E Rod Misalignment Limitations (*)

Delete reference to part length control rods in 3.10.E.1. Correct abbreviation for rod cluster control assembly to RCCA. Delete footnote 4.

In 3.10.E.2, refer to the rod as rod cluster control assembly.

3.10.F Inoperable Rod Position Indicator Channels (*)

In 3.10.F.2, change the term "sub-bank" to "group" which is consistent with plant terminology. In 3.10.F.3, delete reference to the part length rods and footnote 5. Also change "will be applied" to the imperative.

3.10.G Inoperable Rod Limitations (*)

In 3.10.G.3 and 4, correct Figure references to TS.3.10-3 and TS.3.10-4, respectively.

In 3.10.G.5, change last three lines to read "shall be reduced to a level consistent with safety analysis basis". The previous statement terminology was redundant, referring to "...the plant power level shall be reduced to an analytically-determined part-power level which is consistent with the safety analysis".

3.10.I and J (*)

Combine these two sections into one that deals with "Monitor Inoperability".

3.10.K DNB Parameters (*)

The reactor coolant flow specified is consistent with transient analyses assumptions. A higher flow value of 190,800 gpm had been used to offset DNB related penalties imposed in 1976. The DNB related $F_{\Delta H}^N$ penalty previously given in Figure TS.3.10-7 has been eliminated since for fuel regions with high burnups, the depletion of fissile nuclides and buildup of fission products greatly reduces power production capability. These combined burnup effects reduce $F_{\Delta H}^N$ sufficiently to cover rod bow penalties beyond a region average burnup of 40,000 MUD/MTU.

Bases (*)

Changes in this section have been organized in the same order as the limiting conditions for operation. The bases for the various LCO's are identified specifically.

In addition to the aforementioned changes, all pages in this section have been renumbered, eliminating 3.10-1A, -7A, -10a, and -13A. The main sidebars indicate revised wording in the section only since all pages except 3.10-1 would need a full sidebar if one considered changes of each page.

Reason for Editorial Changes

These have been provided above.

Safety Evaluation - Editorial Changes

The noted changes (*) have no safety implications and are strictly administrative in nature.

B. $F_{\Delta H}^N$ Penalty Associated with Rod Bow

Proposed Change

On page TS.3.10-1, change the Specification 3.10.B.1. $F_{\Delta H}^N$ limit specified to delete the [1-RBP(BU)] multiplier for the Westinghouse fuel. Delete the footnote (+) at the bottom of the page referring to this multiplier. This results from the combination of the rod bow penalty versus amount of rod bow recently approved by the NRC (Reference 1) with the rod bow versus burnup for Westinghouse 15 x 15 (14 x 14) fuel given by the NRC interim SER (Reference 10).

Reason for Change

To eliminate the rod bow penalty to accommodate expected $F_{\Delta H}^N$ values that will be encountered at rated power during reload cycles.

Safety Evaluation

This change only incorporates recent information on rod bow penalties and does not reduce the margin of safety as defined in the basis for this specification or as defined in the FSAR. Therefore, this change does not constitute an unreviewed safety question. The new RCS flow rate specified is that used in the transient analyses.

C. F_Q^N Limit Penalty Associated with Rod Bow

Proposed Change

On page TS.3.10-1, change the specification 3.10.B.1.limit to delete the multiplier

$$[1 - 2.35 \times 10^{-6} (\text{BU} - 2.8 \times 10^3)]$$

and delete footnote * at the bottom of the page.

Reason for Change

The aforementioned multiplier was imposed by the NRC staff because an empirically based topical report on rod bow had not been submitted by ENC for approval by the staff. This penalty factor results in a 2% F_Q restriction at EOC5. The penalty should not restrict plant operation during Cycle 5 because F_Q decreases over the plant cycle by approximately 10%. This decrease was shown in Reference 2 for a representative cycle for each Prairie Island unit. This decrease is also shown in Reference 3 for representative plants with poison using the power distribution phase 2 control methods. At beginning of a cycle a unit's measured F_Q^N is expected to be maximum and closest to the limit. For this reason, it is requested to have this change approved by August 1, 1980.

Safety Evaluation

Exxon Nuclear Company recently submitted "Computational Procedure for Evaluating Fuel Rod Bowing" XN-NF-75-32, [NP] (Reference 4) for NRC staff review. This document was prepared in response to suggestions by the NRC staff (Reference 5) and provides supporting documentation for eliminating the penalty.

The NRC staff, in the safety evaluation report for Prairie Island NGP Amendments 35 (DPR 42) and 29 (DPR-60), stated that the major reasons for unacceptance of the ENC rod bow methodology was that (1) the description of the statistical calculations in References 6 and 7 did not contain adequate detail to give a precise meaning to the 95/95 limit used, and (2) more information should be presented on the methods used for calculating rod bow magnitude.

XN-NF-75-32 [NP] "Computational Procedure for Evaluating Fuel Rod Bowing" describes -

- (1) the equipment and procedures for accumulating gap spacing data
- (2) the statistical model used for reducing the gap spacing data
- (3) the thermal hydraulic model used to evaluate the effect of rod bow magnitude on margin to DNB, and
- (4) the neutronic fuel rod dislocation model used to calculate the rod bow LHGR augmentation factor used in the LOCA analysis

This document describes the rod bow methodology in an empirical fashion, compared to the mechanistic AXIBOW model (Reference 8) previously reviewed by the NRC staff. XN-NF-75-32 addresses question heretofore raised by the NRC staff as appropriate for the ENC fuel.

D. Exposure Dependence of F_Q^N

Proposed Changes

Changes attributable to exposure dependence of F_Q^N are included in specification 3.10.B.1. and Figure TS.3.10-7. Simply stated, these changes are -

- (1) redefining the limit in a form in which the measured F_Q^N is to be compared to the analytically determined F_Q^N .
- (2) Including a normalized exposure dependent function.

Reason for Change

10CFR 50.46 requires that selected acceptance criteria are met. This change is necessary to assure that those criteria are met.

Safety Evaluation

10CFR 50.46 specifies the acceptance criteria for ECC systems for light water reactors. Limits are imposed on peak clad temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling. Appendix K describes the required and acceptable features of the ECCS evaluation models and the required documentation.

Reference 9 (Exhibit C) describes an exposure sensitivity study conducted for the Prairie Island units. This analysis includes the NRC high burnup fission gas release model, not included in the analysis described in Reference 7, submitted on December 29, 1978 with the license amendment related to use of ENC fuel. This analysis includes pin internal pressure uncertainties due to various parameters compared to rupture pressure at seven (7) exposure conditions. Also presented are 10CFR 50.46 comparative data at the same seven (7) peak pellet exposures.

The data presented clearly shows that 10CFR 50.46 criteria are met over the span of expected peak pellet exposures. Variation of the F_Q^N limit with burnup assures that the 50.46 criteria are met. This variation should not affect power operation of the Prairie Island units since F_Q^N limit decrease is not expected until late in the second cycle or early in the third cycle when F_Q^N values would be lower. The data is conservative relative to the higher burnup fuel (References 6, 11).

E. Power Distribution Control, Phase II Specification Update

Proposed Changes

Changes related to PDC2 are shown in sections 3.10.B.1 through 3.10.B.3. and in the bases. (Exhibit B). The F_Q^N and F_{Δ}^N limits are specified in the format presented to show that a measured value is being compared to an analytically determined limit.

Reason for Change

The changes proposed in Reference 3, Attachment A, are an outgrowth of generic discussions held between members of the NRC and ENC staffs. The changes are related to -

- (1) more clearly defining the measurement conditions
- (2) eliminating the unnecessarily restrictive definition of equilibrium conditions and the definition of P^1 .
- (3) including as a specification a commitment previously described in Reference 2, Section B
- (4) Redefining the requirement for measurement of the reference equilibrium indicated axial flux difference on a monthly rather than quarterly basis

Safety Evaluation

These changes are consistent with those proposed in Reference 3 based on the methodology described in that document.

References

1. Letter, J F Stolz (NRC) to T M Anderson (Westinghouse) dated April 5, 1979 (Staff review of WCAP-8691)
2. Letter, L O Mayer (NSP) to Director of Nuclear Reactor Regulation (NRC) dated March 30, 1979
3. Exxon Nuclear Company Topical Report, "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors Phase II", XN-NF-77-57, Supplement 1, June 1979
4. Exxon Nuclear Company Topical Report, "Fuel Rod Bowing for Exxon Nuclear Company Reload Fuel," XN-NF-79-43, May 1979
5. Letter, G Lear (NRC) to W Nechodom (ENC), dated June 7, 1978
6. Exxon Nuclear Company Report, "Generic Mechanical and Thermal Hydraulic Design for Exxon Nuclear 14 x 14 Reload Fuel Assemblies with Zircaloy Guide Tubes for Westinghouse 2-Loop Pressurized Water Reactors, XN-NF-78-34 [P], November 1978 (proprietary version), and XN-NF-78-34 [NP], January 1979 (non-proprietary version)
7. Exxon Nuclear Company Report, "ECCS Large Break Spectrum Analysis for Prairie Island Unit 1 Using ENC WREM-IIA PWR Evaluation Model," XN-NF-78-46, November 1978
8. Exxon Nuclear Company Topical Report, "Computational Procedures for Evaluating Fuel Rod Bowing (AXIBOW)", XN-NF-75-32, Supplement 1, July 1979
9. Exxon Nuclear Company Report, "Exposure Sensitivity Study for ENC XN-1 Reload Fuel at Prairie Island Unit 1 Using the ENC-WREM-IIA PWR Evaluation Model", XN-NF-79-18 (P), April 1980 (proprietary); XN-NF-79-18(NP), April 1980 (non-proprietary)
10. Interim Safety Evaluation Report on Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors - Revision 1, USNRC, February 16, 1977.
11. Letter, L O Mayer (NSP) to Director of Nuclear Reactor Regulation (NRC), dated October 30, 1979.
12. NUREG-0452, "Westinghouse Pressurized Water Reactor Standardized Technical Specifications".
13. Amendments No. 32 (DPR-42) and 26 (DPR-60), dated November 1, 1978.

EXHIBIT B

License Amendment Request dated May 6, 1980

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

Pages

TS iv
TS.3.5-2
Table TS.3.5-1
Table TS.3.5-3
TS.3.10-1 through -17
Figure TS.3.10-7

All pages in Section 3.10 were changed and renumbered.

APPENDIX A TECHNICAL SPECIFICATIONS

LIST OF FIGURES

<u>TS FIGURE</u>	<u>TITLE</u>
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
3.10-1	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	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

Safety Injection

The Safety Injection System is actuated automatically to provide emergency cooling and reduction of reactivity in the event of a loss-of-coolant accident or a steam line break accident.

Safety injection in response to a loss-of-coolant accident (LOCA) is provided by a high containment pressure signal backed up by the low pressurizer pressure signal. These conditions would accompany the depressurization and coolant loss during a LOCA.

Safety injection in response to a steam line break is provided directly by a low steam line pressure signal, backed up by the low pressurizer pressure signal and, in case of a break within the containment, by the high containment pressure signal.

The safety injection of highly borated water will offset the temperature-induced reactivity addition that could otherwise result from cooldown following a steam line break.

Containment Spray

Containment sprays are also actuated by a high containment pressure signal (Hi-Hi) to reduce containment pressure in the event of a loss of coolant or steam line break accident inside the containment.

The containment sprays are actuated at a higher containment pressure (approximately 50% of design containment pressure) than is safety injection (10% of design). Since spurious actuation of containment spray is to be avoided, it is initiated on coincidence of high containment pressure sensed by three sets of one-out-of-two containment pressure signals provided for its actuation.

Containment Isolation

A containment isolation signal is initiated by any signal causing automatic initiation of safety injection or may be initiated manually. The containment isolation system provides the means of isolating the various pipes passing through the containment walls as required to prevent the release of radioactivity to the environment in the event of a loss-of-coolant accident.

TABLE TS.3.5-1

REV

ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT LIMITING SET POINTS

FUNCTIONAL UNIT	CHANNEL	LIMITING SET POINTS*
1 High Containment Pressure (Hi)	Safety Injection*	≤ 4 psig
2 High Containment Pressure (Hi-Hi)	a. Containment Spray	≤ 23 psig
	b. Steam Line Isolation of Both Lines	≤ 17 psig
3 Pressurizer Low Pressure	Safety Injection*	≥ 1815 psig
4 Low Steam Line Pressure	Safety Injection*	> 500 psig
	Lead Time Constant	> 12 seconds
	Lag Time Constant	≤ 2 seconds
5 High Steam Flow in a Steam Line Coincident with Safety Injection and Low T_{avg}	Steam Line Isolation of Affected Line	d/p corresponding to $\leq 0.745 \times 10^6$ lb/hr at 1005 psig
		$> 540^\circ\text{F}$
6 High-high Steam Flow in a Steam Line Coincident with Safety Injection	Steam Line Isolation of Affected Line	\leq d/p corresponding to 4.5×10^6 lb/hr at 735 psig
7 High Pressure Difference Between Shield Building and Containment	Containment Vacuum Breakers	≤ 0.5 psi
8 High Temperature in Ventilation Ducts	Ventilation System Isolation Dampers	$\leq 120^\circ\text{F}$

- *Initiates also containment isolation, feedwater line isolation and starting of all containment fans.
- d/p means differential pressure

TABLE TS.3.5-3

REV

INSTRUMENT OPERATING CONDITIONS FOR EMERGENCY COOLING SYSTEM

<u>FUNCTIONAL UNIT</u>	<u>1</u> MINIMUM OPERABLE CHANNELS	<u>2</u> MINIMUM DEGREE OF REDUNDANCY	<u>3</u> PERMISSIBLE BYPASS CONDITIONS	<u>4</u> OPERATOR ACTION IF CONDITIONS OF COLUMN 1 or 2 CANNOT BE MET
1. SAFETY INJECTION				
a. Manual	2	1		Hot shutdown **
b. High Containment Pressure	2	1		Hot shutdown **
c. Steam Generator Low Steam Pressure/Loop	2	1	primary pressure less than 2000 psig	Hot shutdown **
d. Pressurizer Low Pressure	2	1	primary pressure less than 2000 psig	Hot shutdown **
2. CONTAINMENT SPRAY				
a. Manual	2	---*		Hot shutdown **
b. Hi-Hi Containment Pressure (Containment Spray)				Hot shutdown **
Channel a	2	1		
Channel b	2	1		
Channel c	2	1		
Logic	2	1		

● * - Must actuate 2 switches simultaneously.

● ** - If minimum conditions are not met within 24 hours, steps shall be taken to place the unit in cold shutdown condition.

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.

E. Power Distribution Limits

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

$$F_Q^N \times V(Z) \times 1.03 \times 1.05 \leq (2.21/P) \times K(Z) \times BU(E_j)$$
$$F_{\Delta H}^N \times 1.04 \leq 1.55 \times [1 + 0.2(1-P)]$$

where the following definitions apply:

- (a) $V(Z)$ is the Power Distribution Control Phase 2 penalty factor applied to the equilibrium measured F_Q^N to bound F_Q^N 's that could be measured at nonequilibrium conditions. $V(Z)$ as used in the limit is equal to the value shown in Figure 3.10-8 except for the top 10% and bottom 10% of the core where the $V(Z)$ factor does not apply.
- (b) $K(Z)$ is the axial dependence function shown in Figure TS.3.10-5.
- (c) Z is the core height location
- (d) E_j is the maximum pellet exposure in fuel assembly j for which the F_Q^N is being measured.

- (e) $BU(E_j)$ is the normalized exposure dependence function for Exxon Nuclear Company fuel shown in Figure TS.3.10-7. For Westinghouse fuel, $BU(E_j) = 1.0$
 - (f) P is the fraction of full power at which the core is operating. In the F_Q^N limit determination when $P \leq 0.50$, set $P = 0.50$.
 - (g) 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
 - (h) 1.03 is the engineering hot channel factor, F_Q^E , applied to the measured F_Q^N to account for manufacturing tolerance
 - (i) 1.05 is applied to the measured F_Q^N to account for measurement uncertainty
 - (j) 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 at equilibrium conditions according to the following schedule:
- (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 by 10% or more of rated thermal power, the thermal power at which target flux differences was last determined
3. (a) If the measured $F_{\Delta H}^N$ exceeds the 3.10.B.1 limit, reduce reactor power and the high neutron flux trip setpoint by 1% for each percent that the measured $F_{\Delta H}^N$ exceeds the 3.10.B.1 limit. Then follow 3.10.B.3.(c).
- (b) If the measured F_Q^N exceeds the 3.10.B.1 limit, take one of the following actions:
- 1. Place the reactor in an equilibrium configuration for which Specification 3.10.B.1 is satisfied, or
 - 2. Reduce reactor power and the high neutron flux trip setpoint by 1% for each percent that the measured $F_Q^N \times 1.03 \times 1.05$ exceeds the $(2.21/P) \times K(Z) \times BU(E_j)$ limit. Then follow 3.10.B.3(c).

- (c) If subsequent in-core mapping cannot, with 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^N or $F_{\Delta H}^N$ is demonstrated through in-core mapping to be within its limits.
- (d) If two successive measurements indicate an increase in the peak pin power $F_{\Delta H}^N$ with exposure, either of the following actions shall be taken:
1. F_Q^N shall be multiplied by $1.02 \times V(Z) \times 1.03 \times 1.05$ for comparison to the limit specified in 3.10.B.1, or
 2. F_Q^N shall be measured at least once per seven effective full power days until two successive maps indicate that the peak pin power, $F_{\Delta H}^N$, is not increasing.
4. Except during physics tests, and except as provided by Specifications 5 through 8 below, the indicated axial flux difference for at least three operable excore channels shall be maintained within a $\pm 5\%$ band about the target flux difference.
5. Above 90 percent of rated thermal power:
- If the indicated axial flux difference of two operable excore channels deviates from its target band, within 15 minutes either eliminate such deviation, or reduce thermal power to less than 90 percent of rated thermal power.
6. Between 50 and 90 percent of rated thermal 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 about the target flux difference does not exceed the envelope 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 less than 50% power and the high neutron flux setpoint reduced to less than 55% of rated power.

*May be extended to 16 hours during incore/excore calibration.

- c. A power increase to a level greater than 90 percent of rated power is contingent upon the indicated axial flux difference of at least three operable excore channels being within the target band.
7. Less than 50 percent of rated thermal power:
 - a. The indicated axial flux difference may deviate from its target band.
 - b. A power increase to a level greater than 50 percent of rated power is contingent upon the indicated axial flux difference of at least three operable excore channels not being outside the target band for more than one hour (cumulative) out of the preceding 24 hour period
 8. In applying 6a and 7b above, penalty deviations outside the +5% target band shall be accumulated on a time basis of:
 - a. One minute penalty deviation for each one minute of power operation outside of the target band at thermal power levels equal to or above 50% of rated thermal power, and
 - b. One-half minute penalty deviation for each one minute of power operation outside of the target band at thermal power levels between 15% and 50% of rated thermal power.
 9. If alarms associated with monitoring the indicated axial flux difference deviations from the +5% target band are not operable, the indicated axial flux difference value for each operable excore channel shall be logged at least once per hour for the first 24 hours and half-hourly thereafter until the alarms are returned to an operable status. For the purpose of applying this specification, logged values of indicated axial flux difference must be assumed to apply during the previous interval between loggings.

C. Quadrant Power Tilt Limits

1. Except for physics tests, if the percentage quadrant power tilt exceeds 2% but is less than 7%, the rod position indication shall be monitored and logged once each shift to verify rod position within each bank assignment and, within two hours, one of the following steps shall be taken:
 - a. Correct the tilt to less than 2%
 - b. Restrict core power level so as not to exceed rated power, less 2% for every percent that the quadrant power tilt ratio exceeds 1.0.

2. If the percentage quadrant power tilt exceeds 2% but is less than 7% for a sustained period of more than 24 hours, or if such a tilt recurs intermittently, the reactor shall be brought to the hot shutdown condition. Subsequent operation below 50% of rating, for testing, shall be permitted.
3. Except for physics tests if the quadrant power tilt ratio exceeds 1.07, the reactor shall be brought to the hot shutdown condition. Subsequent operation below 50% of rating, for testing, shall be permitted.
4. If the core is operating above 85% power with one excore nuclear channel out of service, then the core quadrant power balance shall be determined daily and after 10% power change using either 2 movable detectors or 4 core thermocouples per quadrant, per Specification 3.11.

D. Rod Insertion Limits

1. The shutdown rods shall be fully withdrawn when the reactor is critical or approaching criticality.
2. When the reactor is critical or approaching criticality, the control banks shall be limited in physical insertion; insertion limits are shown in Figure TS.3.10-2, -3 and -4 for normal and abnormal operating conditions.
3. Control bank insertion may be further restricted by specification 3.10.A if, (1) the measured control rod worth of all rods, less the worth of the worst stuck rod, is less than 5.52% reactivity at the beginning of the first cycle or the equivalent value if measured at any other time, or (2) if a rod is inoperable (Specification 3.10.G).
4. Insertion limits do not apply during physics tests or during periodic exercise of individual rods. The shutdown margin shown in Figure TS.3.10-1 must be maintained except for low power physics testing. For this test the reactor may be critical with all but one high worth full-length control rod inserted for a period not to exceed 2 hours per year provided a rod drop test is run on the high worth full-length rod prior to this particular low power physics test.

E. Rod Misalignment Limitations

1. If a full-length rod cluster control assembly (RCCA) is misaligned from its bank by more than 15 inches, the rod will be realigned or the core power peaking factors shall be determined within 2 hours, and Specification 3.10.B applied. If peaking factors are not determined within 2 hours, the high neutron flux trip setpoint shall be reduced to 85 percent of rating.
2. If the misaligned rod cluster control assembly is not realigned within a total of 8 hours, the rod shall be declared inoperable.

F. Inoperable Rod Position Indicator Channels

1. If a rod position indicator (RPI) channel is out of service then
 - a. For operation between 50% and 100% of rating, the position of the RCCA shall be checked directly by core instrumentation (excore detector and/or thermocouples and/or movable incore detectors) every shift or subsequent to rod motion exceeding a total of 24 steps, whichever occurs first.
 - b. During operation below 50% of rating, no special monitoring is required.
2. The plant shall be brought to the hot shutdown condition should more than one RPI channel per group or more than two RPI channels per bank be found to be inoperable during power operation.
3. If a full length rod having a rod position indicator channel out of service is found to be misaligned from 1.a. above, then apply Specification 3.10.E.

G. Inoperable Rod Limitations

1. An inoperable rod is a rod which (a) does not trip, (b) is declared inoperable under Specification 3.10.E. or 3.10.H. or (c) cannot be moved by its drive mechanism and cannot be corrected within 8 hours.

2. The plant shall be brought to the hot shutdown condition should more than one inoperable full length rod be discovered during power operation.
3. If the inoperable full-length rod is located below the 200 step level and is capable of being tripped, or if the full-length rod is located below the 30 step level whether or not it is capable of being tripped, then the insertion limits in Figure TS.3.10-3 apply.
4. If the inoperable full-length rod cannot be located, or if the inoperable full-length rod is located above the 30 step level and cannot be tripped, then the insertion limits in Figure TS.3.10-4 apply.
5. If reactor operation is continued with one inoperable full-length rod, the potential ejected rod worth and associated transient power distribution peaking factors shall be determined by analysis within 30 days unless the rod is earlier made operable. The analysis shall include due allowance for nonuniform fuel depletion in the neighborhood of the inoperable rod. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, the plant power level shall be reduced to a level consistent with the safety analysis.

H. Rod Drop Time

At operating temperature and full flow, the drop time of each full-length RCCA shall be no greater than 1.8 seconds from loss of stationary gripper coil voltage to dashpot entry. If the time is greater than 1.8 seconds, the rod shall be declared inoperable.

I. Monitor Inoperability Requirements

1. If the rod bank insertion limit monitor is inoperable, or if the rod position deviation monitor is inoperable, individual rod positions shall be logged once per shift, after a load change greater than 10 percent of rated power, and after 30 inches or more of rod motion.
2. If both the rod position deviation monitor and one or both of the quadrant power tilt monitors are inoperable for 2 hours or more, the nuclear overpower trip shall be reset to 93% of rated power in addition to the increased surveillance requirements.

3. If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs and the calculated power tilt shall be logged every two hours after a load change greater than 10% of rated power.

J. DNB Parameters

The following DNB related parameters limits shall be maintained during power operation:

- a. Reactor Coolant System Tavg $\leq 564^{\circ}\text{F}$
- b. Pressurizer Pressure ≥ 2220 psia*
- c. Reactor Coolant Flow $\geq 178,000$ gpm

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce thermal power to less than 5% of rated thermal power using normal shutdown procedures.

Compliance with a. and b. is demonstrated by verifying that each of the parameters is within its limits at least once each 12 hours.

Compliance with c. is demonstrated by verifying that the parameter is within its limit after each refueling cycle.

Bases

Throughout the 3.10 Technical Specifications, the terms "rod(s)" and "RCCA(s)" are synonymous.

Shutdown Reactivity

Trip shutdown reactivity is provided consistent with plant safety analyses assumptions. One percent shutdown is adequate except for the steam break analysis, which requires more shutdown reactivity due to the more negative moderator temperature coefficient at end of life (when boron concentration is low). Figure TS.3.10-1 is drawn accordingly.

Power Distribution Control

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operations) and II (Incidents of Moderate frequency) events by: (a) maintaining the minimum DNBR in the core > 1.30 during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding

*Limit not applicable during either a THERMAL POWER ramp increase in excess of (5%) RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of (10%) RATED THERMAL POWER.

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 2.21/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 neutron flux in the core divided by the average neutron flux in the core.

$V(Z)$ 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 on the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

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

$F_{\Delta H}^N$, 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. $F_{\Delta H}^N$ 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 $F_{\Delta H}^N$.

In the specified limit of $F_{\Delta H}^N$ there is an 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F_{\Delta H}^N \leq 1.55/1.08$. The logic behind the larger uncertainty in this case is that:

- (a) abnormal perturbations in the radial power shape (e.g. rod misalignment) affect $F_{\Delta H}^N$, in most cases without necessarily affecting F_Q^N ,
- (b) the operator has a direct influence on F_Q^N through movement of rods, and can limit it to the desired value, while he has no direct control over $F_{\Delta H}^N$ 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^N by tighter axial control, but compensation for $F_{\Delta H}^N$ is less readily available.

When a measurement of $F_{\Delta H}^N$ is taken, experimental error must be allowed for and 4 percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system.

Measurements of the hot channel factors are required as part of startup physics tests, at least once each effective 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. 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 $F_{\Delta H}^N$ and F_O^N 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_O^N is arbitrarily limited for $\rho \leq 0.5$ (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 $F_O(Z)$ upper bound envelope of 2.21/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.

The alarms provided are derived from the plant process computer which determines the one minute averages of the operable excore detector outputs to monitor indicated axial flux difference in the reactor core and alerts the operator when indicated axial flux difference alarm conditions exist. Two types of alarm messages are output. Above a preset (90%) power level, an alarm message is output immediately upon determining a delta flux (as determined from two operable excore channels) exceeding a preset band about a target delta flux value. Below this preset power level, an alarm message is output if the indicated axial flux difference (as determined from two operable excore channels) exceeded its allowable limits for a preset cumulative (usually 1 hour) amount of time in the past 24 hours. For periods during which the alarm on flux difference is inoperable, manual surveillance will be utilized to provide adequate warning of significant variations in expected flux differences. However, every attempt should be made to restore the alarm to an operable condition as soon as possible. Any deviations from the target band during manual logging would be treated as deviations during the entire preceding logging interval and appropriate actions would be taken. This action is necessary to satisfy NRC requirements; however, more frequent readings may be logged to minimize the penalty associated with a deviation from the target band to justify continued operation at the current power. The time that deviations from the target band occur are normally accumulated by the computer above 15% power. Below 15% the probability of exceeding the allowable limits becomes increasingly smaller as it becomes theoretically impossible to deviate from the target band. Between 15-50% power the deviations are more significant and are accumulated at 1/2 of their actual time. Above 50% the deviations are most significant and their time is accumulated on a one for one time basis.

Strict control of the flux difference (and rod position) is not as necessary during part power operation because xenon distribution control at part power is less significant than control at full power. Allowance has been made in predicting the heat flux peaking factors for less strict control at part power. Strict control of the flux difference is not possible during certain physics tests or during required, periodic, excore calibrations which require larger flux differences than permitted. Therefore, the specifications on power distribution control are not applied during physics tests or excore calibrations; this is acceptable due to the low probability of a significant accident occurring during these operations.

In some instances of rapid plant power reduction, automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band, however to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those

resulting from operation within the target band. The consequences of being outside the $\pm 5\%$ target band but within the Figure TS.3.10-6 limit for power levels between 50% and 90% has been evaluated and determined to result in acceptable $F_Q(Z)$ values. Therefore, while the deviation exists the power level is limited to 90 percent or lower depending on the indicated axial flux difference. In all cases the ± 5 percent target band is the Limiting Condition for Operation. Only when the target band is violated do the limits under Figure TS.3.10-6 apply.

If, for any reason, the indicated axial flux difference is not controlled within the ± 5 percent band for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50 percent is required to protect against potentially more severe consequences of some accidents.

As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished by using the boron system to position the full length control rods to produce the required indicated flux difference.

For Condition II events the core is protected from overpower and a minimum DNBR of 1.30 by an automatic protection system. Compliance with operating procedures is assumed as a precondition for Condition II transients, however, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

Quadrant Power Tilt Limits

Quadrant power tilt limits are based on the following considerations. Frequent power tilts are not anticipated during normal operation since this phenomenon is caused by some asymmetric perturbation, e.g. rod misalignment, x-y xenon transient, or inlet temperature mismatch. A dropped or misaligned rod will easily be detected by the Rod Position Indication System or core instrumentation per Specification 3.10.F, and core limits protected per Specification 3.10.E. A quadrant tilt by some other means (x-y xenon transient, etc.) would not appear instantaneously, but would build up over several hours and the quadrant tilt limits are set to protect against this situation. They also serve as a backup protection against the dropped or misaligned rod.

Operational experience shows that normal power tilts are less than 1.01. Thus, sufficient time is available to recognize the presence of a tilt and correct the cause before a severe tilt could build up. During start-up and power escalation, however, a large tilt could be initiated. Therefore, the Technical Specification has been written so as to prevent escalation above 50 percent power if a large tilt is present.

The numerical limits are set to be commensurate with design and safety limits for DNB protection and linear heat generation rate as described below.

The percentage quadrant power tilt of 2% at which remedial and corrective action is required has been set so as to provide DNB and linear heat generation rate protection with x-y power tilts. Analyses have shown that percentage increases in the x-y power peaking factor are less than or equal to twice the increase in the indicated quadrant power tilt.

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

Therefore, a limiting power tilt of 3 percent can be tolerated. However, a measurement uncertainty is associated with the indicated quadrant power tilt. Thus, allowing for a low measurement of power tilt, the action level of indicated tilt has been set at 2 percent. An alarm is set to alert the operator to an indicated tilt of 2 percent or greater for which action is required. To avoid unnecessary power changes, the operator is allowed two hours in which to verify the actual tilt with in-core mappings or to determine and correct the cause of the tilt.

Should this action not be taken, the margin for uncertainty in F_Q^N is reinstated by reducing the power by 2 percent for each percent of tilt above 1.0, in accordance with the relationship described above, or as required by the restriction on peaking factors.

The upper limit on the quadrant tilt at which hot shutdown is required has been set so as to provide protection against excessive linear heat generation rate. The ratio of overpower to normal operation is approximately 1.15. Since the x-y component of F_Q^N is bounded by the above described relation with indicated quadrant tilt, the overpower linear heat generation rate can be avoided if the indicated tilt is restricted below 7 percent.

Rod Insertion Limits

Rod insertion limits are used to assure adequate trip reactivity, to assure meeting power distribution limits, and to limit the consequences of a hypothetical rod ejection accident. The available control rod reactivity (or excess beyond needs) decreases with decreasing boron concentration. The negative reactivity required to reduce the core power level from full power to zero power is largest when the boron concentration is low since the power defect increases with core burnup.

The intent of the test to measure control rod worth and shutdown margin (Specification 3.10 D.) is to measure the worth of all rods less the worth of the the most reactive rod. The measurement would be anticipated as part of the initial startup program and infrequently over the life of the plant, to be associated primarily with determinations of special interest such as end of life cooldown, or startup of fuel cycles which deviate from normal equilibrium conditions in terms of fuel loading patterns and anticipated control bank worths. These measurements will augment the normal fuel cycle design calculations and place the knowledge of shutdown capability on a firm experimental as well as analytical basis.

An evaluation has been made of anticipated transients and postulated accidents, assuming that they occur during the portion of this test when the reactor is critical with all but one full-length control rod fully inserted. Further, the withdrawn full-length rod is assumed not to trip. As a result of this evaluation, it has been determined that for a steam line break upstream of the flow restrictor, the possibility of core DNB exists. However, even if core damage does result, any core fission product release would be low because of the low fission product inventory during initial startup physics testing; and further, would be contained within the reactor coolant system.

Thus, for the initial startup physics tests, this test will not endanger the health and safety of the public even in the event of highly improbable accidents coupled with the failure of the withdrawn control rod to trip. To perform this test later in life is equally valuable, as stated above. Therefore, this specification has been written to further minimize the likelihood of any hypothesized event during the performance of these tests later in life. This is accomplished by limiting to two hours per year the time the reactor can be in this type of configuration, and requiring that a rod drop test is performed on the rod to be measured prior to performance of test.

Operation with abnormal rod configuration during low power and zero power testing is permitted because of the brief period of the test and because special precautions are taken during the test.

Rod Misalignment Limitation

Rod misalignment requirements are specified to ensure that power distributions more severe than those assumed in the safety analyses do not occur.

Inoperable Rod Position Indicator Channels

The rod position indicator channel is sufficiently accurate to detect a rod +7 inches away from its demand position. A misalignment less than 15 inches does not lead to over-limit power peaking factors. If the rod position indicator channel is not operable, the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors, and/or core thermocouples, and/or movable incore detectors, will be used to verify power distribution symmetry. These indirect measurements do not have the same resolution if the bank is near either end of the core, because a 15-inch misalignment would have no effect on power distributions. Therefore, it is necessary to apply the indirect checks following significant rod motion.

Inoperable Rod Limitations

One inoperable control rod is acceptable provided that the power distribution limits are met, trip shutdown capability is available, and provided the potential hypothetical ejection of the inoperable rod is not worse than the cases analyzed in the safety analysis report. The rod ejection accident for an isolated fully-inserted rod will be worse if the residence time of the rod is long enough to cause significant non-uniform fuel depletion. The four-week period is short compared with the time interval required to achieve a significant non-uniform fuel depletion.

Rod Drop Time

The required drop time to dashpot entry is consistent with the safety analysis.

Monitor Inoperability Requirements

If either the rod bank insertion limit monitor or rod position deviation monitor are inoperable, additional surveillance is required to ensure adequate shutdown margin is maintained.

If the rod position deviation monitor and quadrant power tilt monitor(s) are inoperable, the overpower reactor trip setpoint is reduced (and also power) to ensure that adequate core protection is provided in the event that unsatisfactory conditions arise that could affect radial power distribution.

Increased surveillance is required, if the quadrant power tilt monitors are inoperable and a load change occurs, in order to confirm satisfactory power distribution behavior. The automatic alarm functions related to quadrant power tilt must be considered incapable of alerting the operator to unsatisfactory power distribution conditions.

DNB Parameters

The RCS flow rate, T_{avg} , and Pressurizer Pressure requirements are based on transient analyses assumptions. The flow rate shall be verified by calorimetric flow data and/or elbow taps. Elbow taps are used in the reactor coolant system as an instrument device that indicates the status of the reactor coolant flow. The basic function of this device is to provide information as to whether or not a reduction in flow rate has occurred. If a reduction in flow rate is indicated below the specification value indicated, shutdown is required to investigate adequacy of core cooling during operation.

For fuel regions with high burnups, the depletion of fissile nuclides and build-up of fission products greatly reduces power production capability. These combined burnup effects reduce $F_{\Delta H}^N$ sufficiently to cover residual rod bow penalties beyond a region average burnup of 40,000 MWD/MTU.

FIGURE TS.3.10-7

Normalized Exposure Dependent Function $BU(E_j)$ for Exxon Nuclear Company Fuel

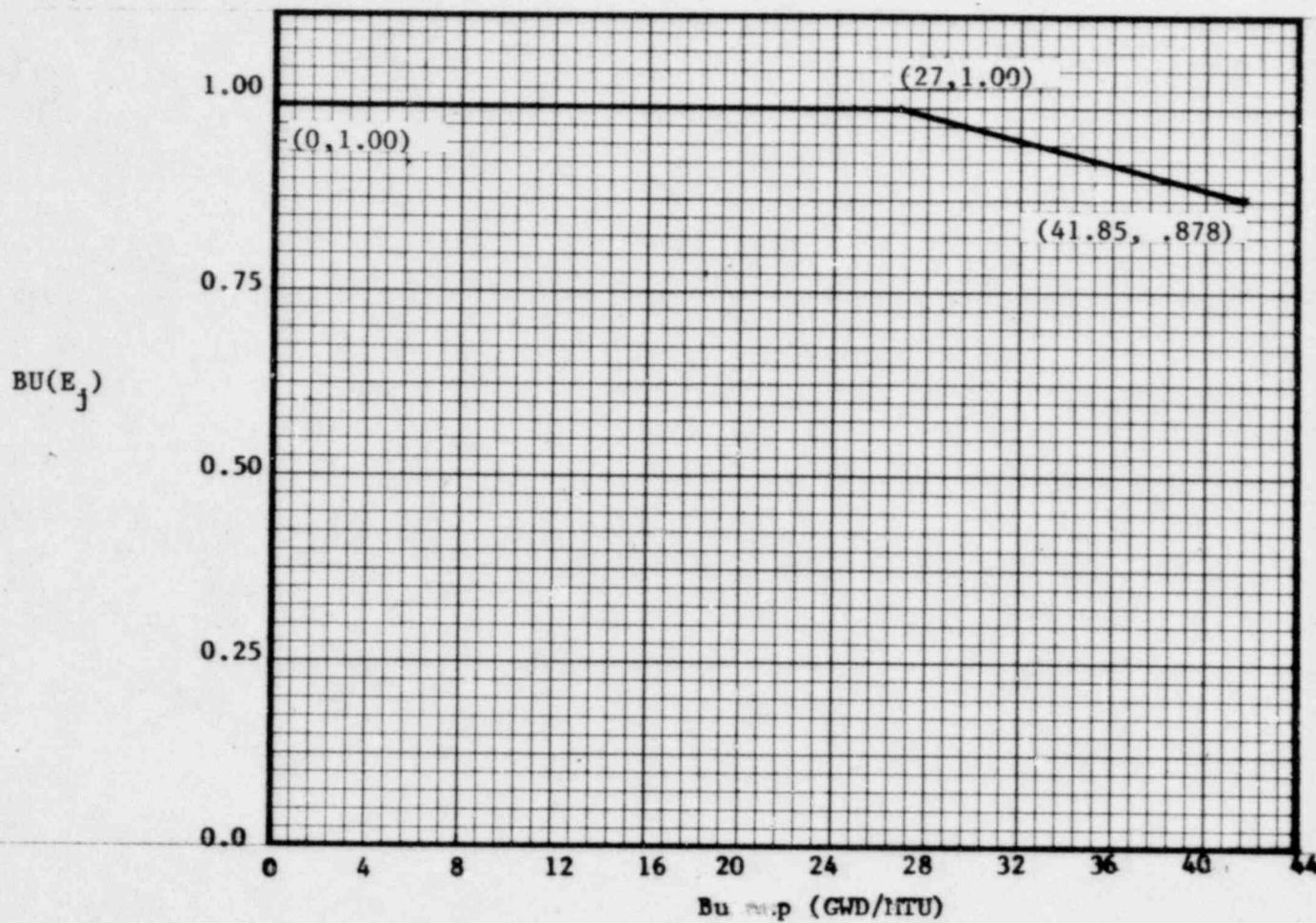


FIGURE TS.3.10-7
REV

