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November 16, 1984

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Ms. E. G. Adensam, Chief
Licensing Branch No. 4

Subject: McGuire Nuclear Station
Docket Nos. 50-369 and 50-370
McGuire 2/Cycle 2 OFA Reload

Dear Mr. Denton:

Mr. H. B. Tucker's (DPC) November 14, 1983 letter to Mr. H. R. Denton (NRC/ONRR) described planned changes in the fuel design for McGuire Nuclear Station, Units 1 and 2. Commencing with the first refueling of each of the units, the standard fuel assemblies in use are to be replaced over the next four refuelings with optimized fuel assemblies (OFA). The letter transmitted a reference safety evaluation describing the safety impact of operation with a transition core and an all OFA core, and indicated that since the transition to OFA involves changes in operating limits, license amendments will be required for operation of both Units 1 and 2 beyond the first cycle. McGuire Unit 1 has already begun this process with the NRC having approved the necessary license amendments via Ms. E. G. Adensam's April 20, 1984 letter to Mr. H. B. Tucker, and Unit 1/Cycle 2 is currently operating with an OFA reload region.

Attached are proposed license amendments to facility operating licenses NPF-9 and NPF-17 for McGuire Nuclear Station Units 1 and 2, respectively. The proposed amendments change plant operating limitations given in the Technical Specifications affected by use of the OFA design for McGuire Unit 2/Cycle 2 to ensure plant operation consistent with the design and safety evaluations. It should be noted that certain Unit 2 reload changes are applied to Unit 1 also (as opposed to affecting only Unit 2), but these involve only administrative type changes (corrections of minor errors/typos, clarifications, etc.) or are improvements incorporated for the Unit 2 specifications which are more conservative than the existing Unit 1 specifications. In addition, Technical Specification 3.5.1.2 is revised to delete the inadvertent application to Unit 1 of provisions which do not apply to the current core design.

Attachment 1 contains the proposed technical specification changes, and Attachment 2 discusses the Justification and Safety Analysis to support the proposed changes. Included in Attachment 2 is: A) the cycle-specific reload safety evaluation for McGuire Unit 1/Cycle 2 including Fq surveillance and RAOC/Base Load Technical Specifications. The peaking factor limit report for McGuire Unit 2/Cycle 2

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1/1 # 767506

which is required in accordance with the proposed McGuire Unit 2 Technical Specification paragraph 6.9.1.9 (as given in Attachment 1) will be submitted by December 14, 1984. Pursuant to 10 CFR 50.91, Attachment 3 provides an analysis performed in accordance with the standards contained in 10 CFR 50.92 which concludes that the proposed amendments do not involve a significant hazards consideration. The proposed amendments have been reviewed and determined to have no adverse safety or environmental impact.

For Unit 2/Cycle 2, the large break LOCA analysis applicable for transition and full OFA core cycles of McGuire 1 and 2 was performed utilizing the OFA design consistent with the methodology given in the above-mentioned reference safety evaluation for the OFA transition. This analysis utilized the currently approved UHI large break ECCS evaluation model modified to incorporate BART core reflood heat transfer models. BART has been approved for use on non-UHI plants (WCAP-9561-P-A, "BART-A1: A Computer Code for the Best Estimate Analysis of Reflood Transients," Young, M. Y. , et. al., March 1984). However, McGuire 1 and 2 are UHI plants. Addendum 1 to WCAP-9561 requesting approval for use of BART technology on UHI plants will be submitted by Westinghouse before the end of November 1984. Also, certain Technical Specification changes such as those involving limiting safety system settings changes (e.g. steam generator low-low level setpoint changes and updating of the lag time constants in the Delta-T and T_{avg} channels) require plant modifications which are scheduled to be performed during the refueling outage. Since these changes are contingent upon NRC approval, any concerns with these should be resolved as expeditiously as possible so as not to impact the modification work.

It is requested that the proposed amendments receive timely review and approval in view of the current McGuire Unit 2/Cycle 2 startup schedule. Unit 2 first refueling shutdown is currently planned for late January with return to service planned for late March 1985. Any changes to this schedule will be provided to the NRC staff.

Pursuant to 10 CFR 170.3(y), 170.12(c), and 170.21, Duke Power proposes that this application contains license amendments for McGuire Units 1 and 2 subject to fees based on the full cost of the review (to be calculated using the applicable professional staff rates shown in 10 CFR 170.20) and must be accompanied by an application fee of $150, with the NRC to bill Duke Power at six-month intervals for all accumulated costs for the application or when review is completed, whichever is earlier. Accordingly, please find enclosed a check in the amount of $150.

We will be pleased to meet with the NRC staff to discuss this matter at the staff's convenience.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

PBN:scs
Attachments

cc: (w/all attachments)
Mr. J. P. O'Reilly, Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

Mr. Dayne Brown, Chief
Radiation Protection Branch
Division of Facility Services
Department of Human Resources
P. O. Box 12200
Raleigh, North Carolina 27605

Mr. W. T. Orders
Senior Resident Inspector
McGuire Nuclear Station

HAL B. TUCKER, being duly sworn, states that he is Vice President of Duke Power Company; that he is authorized on the part of said Company to sign and file with the Nuclear Regulatory Commission this revision to the McGuire Nuclear Station License Nos. NPF-9 and NPF-17 and that all statements and matters set forth therein are true and correct to the best of his knowledge.

Hal B. Tucker

Hal B. Tucker, Vice President

Subscribed and sworn to before me this 16th day of November, 1984.

Sue C. Sherrill

Notary Public

My Commission Expires:

September 20, 1989

ATTACHMENT 1

Proposed McGuire Unit 1 and 2 Technical Specification Changes

REPLACE ~~FIG~~ WITH THE FOLLOWING FIGURE ⊗

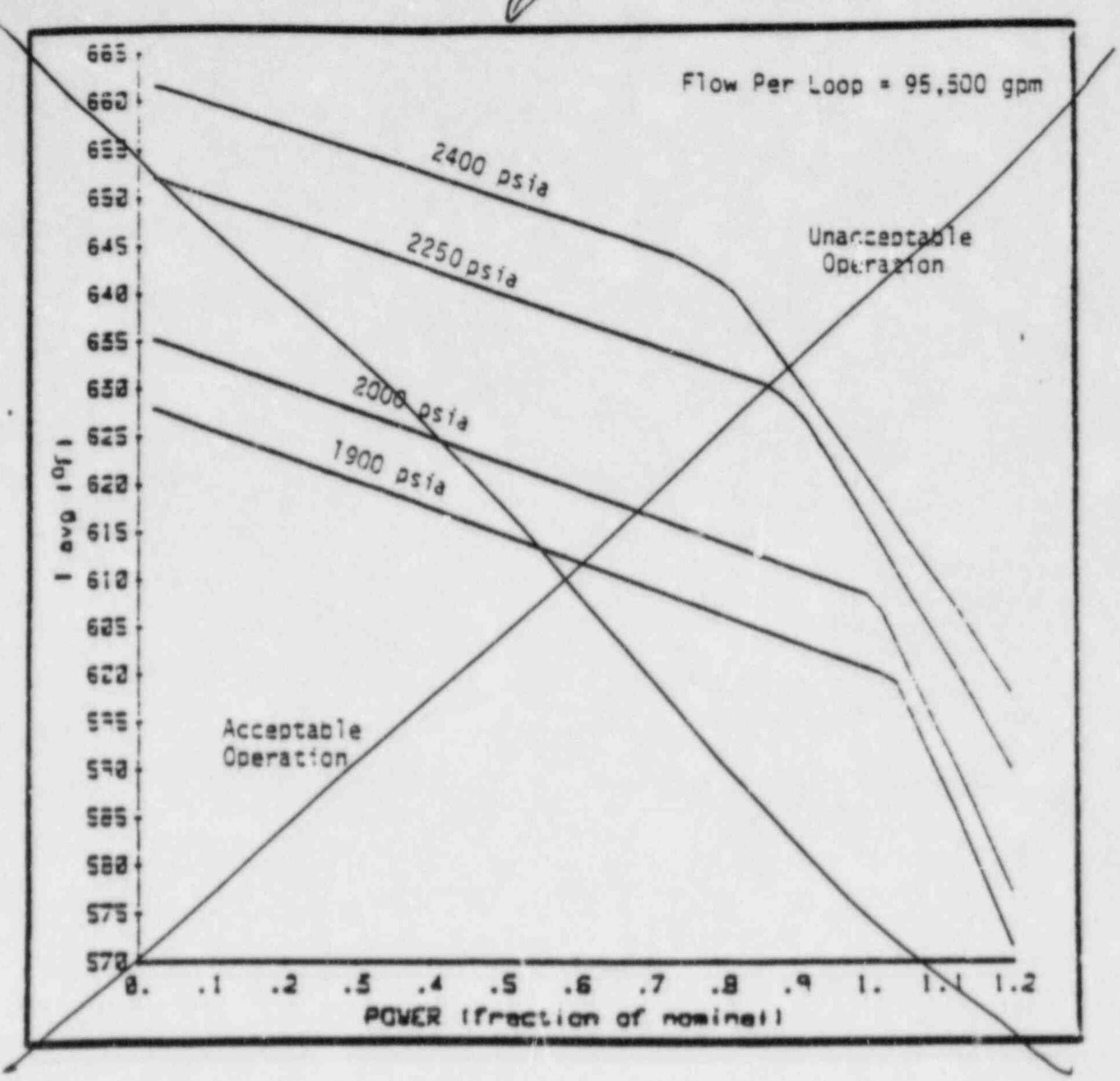
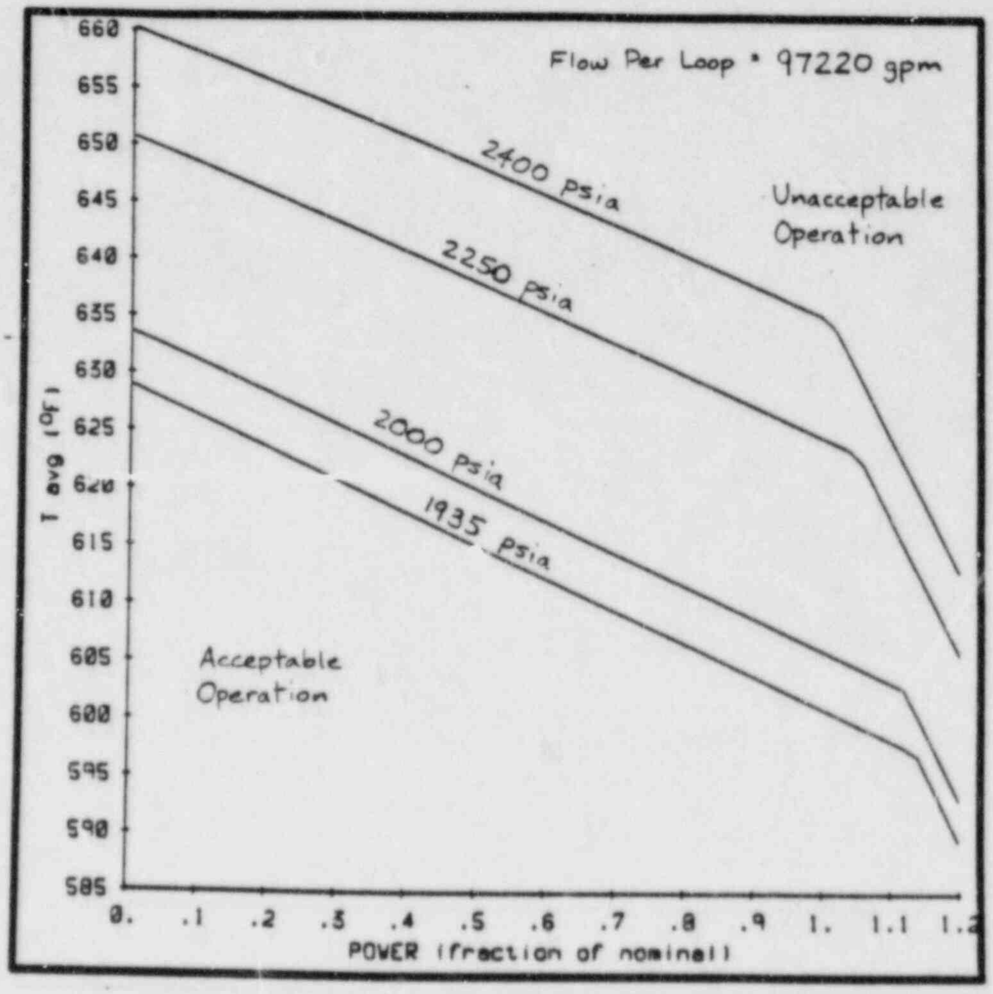


FIGURE 2.1-1b

UNIT 2

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

FIGURE X



~~REACTOR CORE SAFETY LIMITS~~

~~FOUR LOOPS IN OPERATION~~

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	N.A.	N.A.
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 25% of RATED THERMAL POWER High Setpoint - $\leq$ 109% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER High Setpoint - $\leq$ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
5. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 3
9. Pressurizer Pressure--Low	$\geq$ 1945 psig	$\geq$ 1935 psig
10. Pressurizer Pressure--High	$\leq$ 2385 psig	$\leq$ 2395 psig
11. Pressurizer Water Level--High	$\leq$ 92% of instrument span	$\leq$ 93% of instrument span
12. Low Reactor Coolant Flow	$\geq$ 90% of design flow per loop*	$\geq$ 89% of de[illegible] loop*

*Design flow is 98,400 gpm per loop for Unit 1 and ~~95,500~~ 97,220 gpm per loop for Unit 2.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level--Low-Low	$\geq$ 12% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to $\geq$ 54.9% (unit 1), 40.0% (unit 2) of span at 100% of RATED THERMAL POWER.	$\geq$ 11% of span from 0 to 30% of RATED THERMAL POWER, increasing to 53.9% (unit 1), 39.0% (unit 2) of span at 100% of RATED THERMAL POWER.
14. Undervoltage-Reactor Coolant Pumps	$\geq$ 5082 volts-each bus	$\geq$ 5016 volts-each bus
15. Underfrequency-Reactor Coolant Pumps	$\geq$ 56.4 Hz - each bus	$\geq$ 55.9 Hz - each bus
16. Turbine Trip		
a. Low Trip System Pressure	$\geq$ 45 psig	$\geq$ 42 psig
b. Turbine Stop Valve Closure	$\geq$ 1% open	$\geq$ 1% open
17. Safety Injection Input from ESF	N.A.	N.A.
18. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6, Enable Block Source Range Reactor Trip	$\geq 1 \times 10^{-10}$ amps	$\geq 6 \times 10^{-11}$ amps
b. Low Power Reactor Trips Block, P-7		
1) P-10 Input	10% of RATED THERMAL POWER	$\geq$ 9%, $\leq$ 11% of RATED THERMAL POWER
2) P-13 Input	$\leq$ 10% RTP Turbine Impulse Pressure Equivalent	$\leq$ 11% RTP Turbine Impulse Pressure Equivalent

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \left(\frac{1 + \tau_1 S}{1 + \tau_2 S}\right) \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_1 - K_2 \left(\frac{1 + \tau_4 S}{1 + \tau_5 S}\right)\left[T\left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P-P') - f_1(\Delta I)\right\}$$

Where: ΔT = Measured ΔT by RTD Manifold Instrumentation,

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT,

τ_1, τ_2 = Time constants utilized in the lead-lag controller for ΔT, τ_1 $\cancel{=}$ $\geq$ 8 sec., τ_2 $\cancel{=}$ $\leq$ 3 sec.,

$\frac{1}{1 + \tau_3}$ = Lag compensator on measured ΔT,

τ_3 = Time constants utilized in the lag compensator for ΔT, τ_3 $\cancel{=}$ $\leq$ 2 sec. (unit 1), 6 sec. (unit 2)

ΔT_o = Indicated ΔT at RATED THERMAL POWER,

K_1 $\leq$ ~~1.0952~~ 1.200 (Unit 2), 1.4060 (Unit 1),

K_2 = ~~0.0133 (Unit 2)~~, 0.0222 ~~(Unit 1)~~.

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation,

τ_4, τ_5 = Time constants utilized in the lead-lag controller for T_{avg}, τ_4 $\cancel{=}$ $\geq$ 28 sec, ~~(Unit 1), τ_4 = 33 sec. (Unit 2)~~, τ_5 $\cancel{=}$ $\leq$ 4 sec.,

T = Average temperature, °F,

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg},

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 1: (Continued)

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, τ_6 ≤ ~~≥~~ 2 sec (UNIT 1) ~~(Units 1 & 2).~~ 6 sec (UNIT 2)

T' ≤ 588.~~1~~2°F Reference T_{avg} at RATED THERMAL POWER,

K_3 = ~~0.000647 (Unit 2),~~ 0.001095 ~~(Unit 1)~~,

P = Pressurizer pressure, psig,

P' = 2235 psig (Nominal RCS operating pressure),

S = Laplace transform operator, sec^{-1},

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between ~~-36%~~ -29% and ~~+8.0%~~ +9.0% (Unit 2), - 41% and -4.0% (Unit 1); $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds ~~-36%~~ -29% (Unit 2), -41% (Unit 1), the ΔT Trip Setpoint shall be automatically reduced by ~~1.173% (Unit 2),~~ 3.151% ~~(Unit 1)~~ of its value at RATED THERMAL POWER; and

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds ~~+8.0%~~ +9.0% (Unit 2), -4.0% (Unit 1), the ΔT Trip Setpoint shall be automatically reduced by ~~0.901%~~ 1.50% (Unit 2), 1.447% (Unit 1) of its value at RATED THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 2: OVERPOWER ΔT

$$\Delta T \left(\frac{1 + \tau_1 S}{1 + \tau_2 S}\right)\left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right)\left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left[T\left(\frac{1}{1 + \tau_{\cancel{4}\,6} S}\right) - T''\right] - f_2(\Delta I)\right\}$$

Where: ΔT = As defined in Note 1,

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = As defined in Note 1

τ_1, τ_2 = As defined in Note 1

$\frac{1}{1 + \tau_3 S}$ = As defined in Note 1,

ΔT_o = As defined in Note 1,

K_4 $\leq$ 1.0900 (Unit 2), 1.0708 (Unit 1),

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature,

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag controller for T_{avg} dynamic compensation,

τ_7 = Time constant utilized in the rate-lag controller for T_{avg}, $\tau_7 \geq$ 5 sec ~~(Units 1 & 2)~~,

$\frac{1}{1 + \tau_6 S}$ = As defined in Note 1,

τ_6 = As defined in Note 1,

K_6 = ~~0.00126/°F (Unit 2),~~ 0.00169/°F ~~(Unit 1)~~ for T > T" and K_6 = 0 for T $\leq$ T",

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

T	=	As defined in Note 1,
T"	=	$\leq$ 588.~~1~~2°F Reference T_{avg} at RATED THERMAL POWER,
S	=	As defined in Note 1, and
$f_2(\Delta I)$	=	0 for all ΔI.

Note 3: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2%.

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and reactor coolant temperature and pressure have been related to DNB. ~~through the WRB-1 correlation.~~ ~~The WRB-1 DNB correlation~~ THIS RELATION has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30 (based upon W-3 correlation). This value corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figures 2.1-1 and 2.1-2 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

These curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.55\ [1 + 0.2\ (1-P)]$$

Where P is the fraction of RATED THERMAL POWER.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the Setpoints to provide protection consistent with core Safety Limits.

~~For Unit 1,~~ The DNB design basis is as follows: there must be at least a 95% probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 correlation in this application). The correlation DNBR set such that there is a 95% probability with 95% confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95% confidence that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure, and average temperature below which the calculated DNBR is no less than the design DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

The curves are based on a nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, of 1.49 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.49\ [1 + 0.3\ (1-P)]$$

Where P is the fraction of RATED THERMAL POWER.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

Power Range, Neutron Flux (Continued)

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power, a rod drop accident of a single or multiple rods could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBR's will be greater than ~~1.30~~. THE DESIGN LIMIT DNBR VALUE.

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} >200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to ~~1.6% delta k/k (Unit 2)~~, 1.3% delta k/k ~~(Unit 1)~~ for four loop operation.

APPLICABILITY: MODES 1, 2*, 3, and 4.

ACTION:

With the SHUTDOWN MARGIN less than ~~1.6% delta k/k (Unit 2)~~, 1.3% delta k/k ~~(Unit 1)~~, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to ~~1.6% delta k/k (Unit 2)~~, 1.3% delta k/k ~~(Unit 1)~~:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0 at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1e., below, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

*See Special Test Exception 3.10.1.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. ~~For Unit 1,~~ Less positive than the limits shown in Figure 3.1-0, AND

~~b. For Unit 2, less positive than 0 delta k/k/°F for the all rods withdrawn, beginning of cycle life (BOL), hot zero THERMAL POWER condition; and~~

b. ~~c.~~ ~~For Units 1 and 2,~~ Less negative than -4.1×10^{-4} delta k/k/°F for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specifications 3.1.1.3a. ~~and 3.1.1.3b.~~ - MODES 1 and 2* only.#
Specification 3.1.1.3~~c~~b. - MODES 1, 2, and 3 only.#

ACTION:

a. With the MTC more positive than the limit of Specification~~s~~ 3.1.1.3a. ~~or 3.1.1.3b.~~ above, operation in MODES 1 and 2 may proceed provided:

1. ~~For Unit 1,~~ Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the limits shown in Figure 3.1-0 within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

~~2. For Unit 2, control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than 0 delta k/k/°F within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;~~

2. ~~3.~~ The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

3. ~~4.~~ A Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of Specification 3.1.1.3~~c~~b. above, be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.0.

#See Special Test Exception 3.10.3.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit of Specification~~s~~ 3.1.1.3a. ~~and 3.1.1.3b.~~, above, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading; and

b. The MTC shall be measured at any THERMAL POWER and compared to -3.2×10^{-4} delta k/k/°F (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than -3.2×10^{-4} delta k/k/°F, the MTC shall be remeasured, and compared to the EOL MTC limit of Specification 3.1.1.3~~c~~., at least once per 14 EFPD during the remainder of the fuel cycle. b

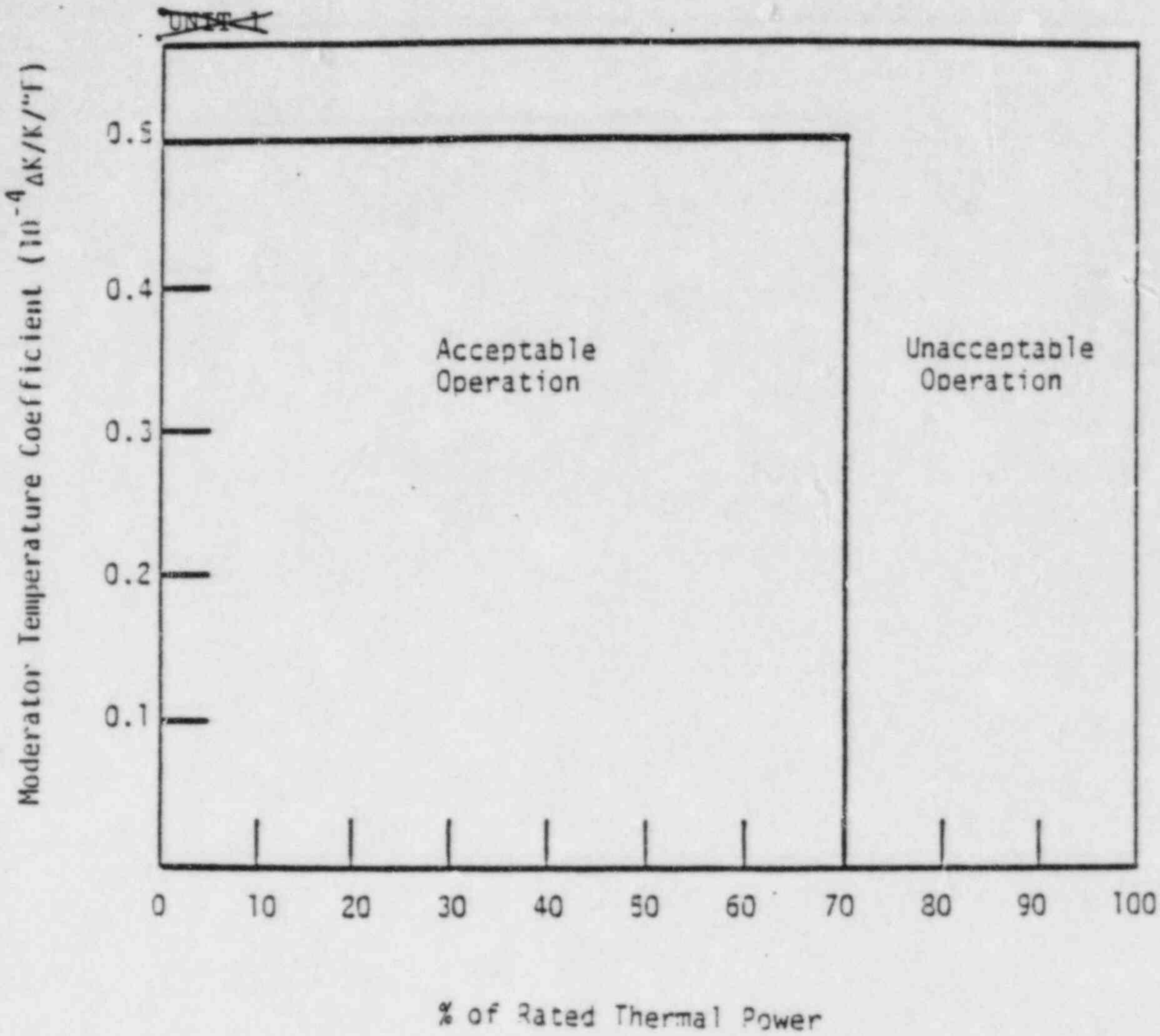


FIGURE 3.1-0

MODERATOR TEMPERATURE COEFFICIENT VS POWER LEVEL ~~(UNIT 1)~~

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (~~UNIT 1~~ AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within:

a. the allowed operational space defined by Figure 3.2-1 for RAOC operation, or

b. within a ± 3 (unit 1), 5 (unit 2) percent target band about the target flux difference during base load operation.

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER*.

ACTION:

a. For RAOC operation with the indicated AFD outside of the Figure 3.2-1 limits,

   1. Either restore the indicated AFD to within the Figure 3.2-1 limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux - High Trip setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. For Base Load operation above APL^{ND}** with the indicated AXIAL FLUX DIFFERENCE outside of the applicable target band about the target flux difference:

   1. Either restore the indicated AFD to within the target band limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than APL^{ND} of RATED THERMAL POWER and discontinue Base Load operation within 30 minutes.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the Figure 3.2-1 limits.

---

*See Special Test Exception 3.10.2.

**APL^{ND} is the minimum allowable power level for base load operation and will be provided in the Peaking Factor Limit Report per Specification 6.9.1.9.

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2. At least once per hour for the first 24 hours after restoring the AFD Monitoring Alarm to OPERABLE status.

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its limits when at least two OPERABLE excore channels are indicating the AFD to be outside the limits.

4.2.1.3 When in Base Load operation, the target axial flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 When in Base Load operation, the target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference ~~pursuant to 4.2.1.3 above~~ or by linear interpolation between the most recently measured value and ~~0 percent~~ at the end of cycle life. The provisions of Specification 4.0.4 are not applicable.

THE CALCULATED VALUE

IN CONJUNCTION WITH THE SURVEILLANCE REQUIREMENTS OF SPECIFICATION 3/4.2.2

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3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (UNIT 2)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the following target band (flux difference units) about the target flux difference:

a. ± 5% for core average accumulated burnup of less than or equal to 3000 MWD/MTU, and

b. + 3% -12% for core average accumulated burnup of greater than 3000 MWD/MTU.

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER*.

ACTION:

a. With the indicated AFD outside of the above required target band about the target flux difference and with THERMAL POWER:

   1. Above 90% of RATED THERMAL POWER, within 15 minutes either:

      a) Restore the indicated AFD to within the target band limits, or

      b) Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

   2. Between 50% and 90% of RATED THERMAL POWER:

      a) POWER OPERATION may continue provided:

         1) The indicated AFD has not been outside of the above required target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

         2) The indicated AFD is within the limits shown on Figure 3.2-1. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

      b) Surveillance testing of the Power Range Neutron Flux channels may be performed pursuant to Specification 4.3.1.1 provided the indicated AFD is maintained within the limits of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

b. THERMAL POWER shall not be increased above 90% of RATED THERMAL POWER unless the indicated AFD is within the above required target band and ACTION a.2.a) 1), above has been satisfied.

---

*See Special Test Exception 3.10.2.

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POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the above required target band for more than 1 hour penalty deviation cumulative during the previous 24 hours. Power increases above 50% of RATED THERMAL POWER do not require being within the target band provided the accumulative penalty deviation is not violated.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

   1) At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

   2) At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its target band when two or more OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each 1 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 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

4.2.1.3 The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 The target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference pursuant to Specification 4.2.1.3 above or by linear interpolation between the most recently measured value and 0% at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

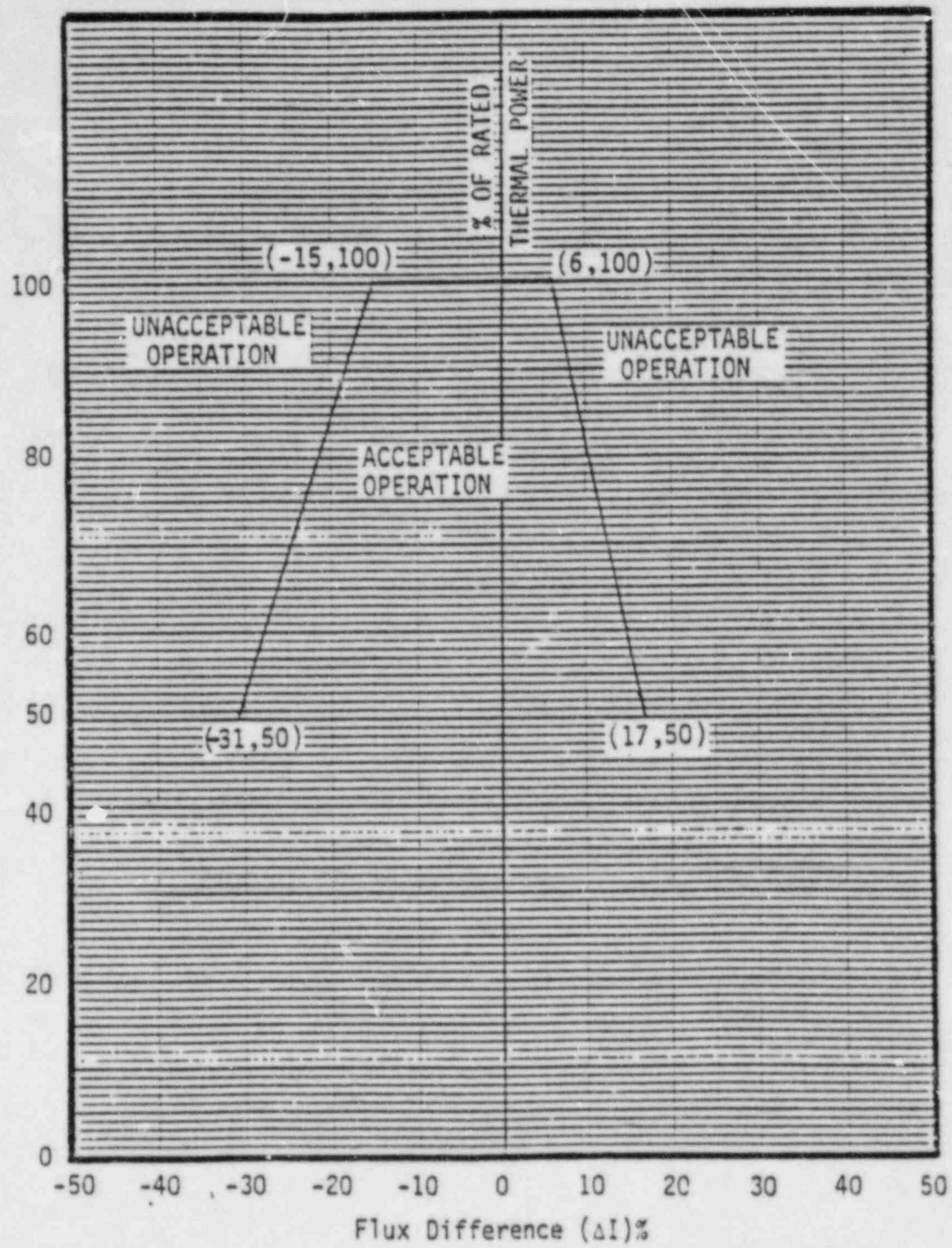

FIGURE 3.2-1~~a~~

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER ~~(UNIT 1)~~

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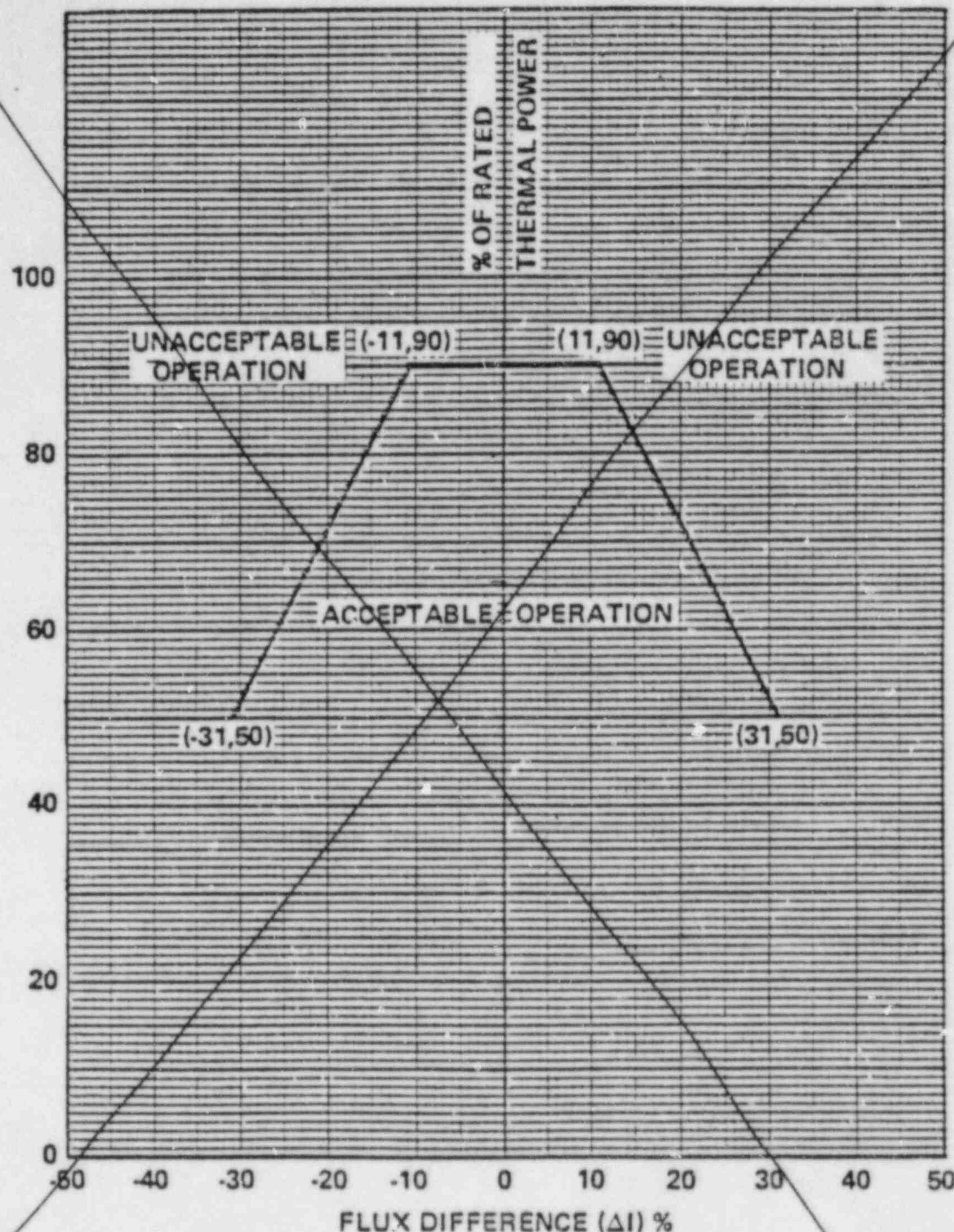

FIGURE 3.2-1b
AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER (UNIT 2)

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \le \frac{[\cancel{2.32}\ 2.26]}{P}\ [K(Z)] \text{ for } P > 0.5 \text{ (Unit 2)}$$

$$F_Q(Z) \le \frac{[2.15]}{P}\ [K(Z)] \text{ for } P > 0.5 \text{ (Unit 1)}$$

$$F_Q(Z) \le \frac{[\cancel{2.32}\ 2.26]}{0.5}\ [K(Z)] \text{ for } P \le 0.5 \text{ (Unit 2)}$$

$$F_Q(Z) \le \frac{[2.15]}{0.5}\ [K(Z)] \text{ for } P \le 0.5 \text{ (Unit 1)}$$

Where: $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

and K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower Delta T Trip Setpoints (value of K_4) have been reduced at least 1% (in ΔT span) for each 1% $F_Q(Z)$ exceeds the limit; and

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 For RAOC operation, $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. VERIFY THE REQUIREMENTS OF SPECIFICATION 3.2.2 ARE SATISFIED.

c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{2.15 \quad \times K(z)}{P \times W(z)} \text{ for } P > 0.5 \text{ (UNIT 1)}$$

$$F_Q^M(z) \leq \frac{2.26 \times K(z)}{P \times W(z)} \text{ FOR } P > 0.5 \text{ (UNIT 2)}$$

$$F_Q^M(z) \cancel{\times} \leq \frac{2.15 \quad \times K(z)}{W(z) \times 0.5} \text{ for } P \leq 0.5 \text{ (UNIT 1)}$$

$$F_Q^M(z) \leq \frac{2.26 \times K(z)}{W(z) \times 0.5} \text{ FOR } P \leq 0.5 \text{ (UNIT 2)}$$

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, 2.15 (UNIT 1) AND 2.26 (UNIT 2) is the F_Q limit, K(z) is given in Figure 3.2-2, P is the relative THERMAL POWER, and W(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. This function is given in the Peaking Factor Limit Report as per Specification 6.9.1.9.

d. Measuring $F_Q^M(z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(z)$ was last determined,* or

2. At least once per 31 Effective Full Power Days, whichever occurs first.

---

*During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

e. With measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)} \right)$$

has increased since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

1) $F_Q^M(z)$ shall be increased by 2% over that specified in Specification 4.2.2.2c. or

2) $F_Q^M(z)$ shall be measured at least once per 7 Effective Full Power Days until two successive maps indicate that

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)} \right) \text{ is not increasing.}$$

f. With the relationships specified in Specification 4.2.2.2c. above not being satisfied:

1) Calculate the percent $F_Q(z)$ exceeds its limit by the following expression:

$$\left\{ \left(\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z) \times W(z)}{\frac{2.15}{P} \times K(z)} \right] \right) - 1 \right\} \times 100 \quad \text{for } P \geq 0.5 \text{ (UNIT 1)}$$

SAME EXPRESSION EXCEPT USE 2.26 INSTEAD OF 2.15 FOR UNIT 2

$$\left\{ \left(\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z) \times W(z)}{\frac{2.15}{0.5} \times K(z)} \right] \right) - 1 \right\} \times 100 \quad \text{for } P < 0.5 \text{ (UNIT 1)}$$

SAME EXPRESSION EXCEPT USE 2.26 INSTEAD OF 2.15 FOR UNIT 2

2) One of the following actions shall be taken:

a) Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the AFD limits of 3.2-1 by 1% AFD for each percent $F_Q(z)$ exceeds its limits as determined in Specification 4.2.2.2f.1). Within 8 hours, reset the AFD alarm setpoints to these modified limits, or

b) Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated above, or

c) Verify that the requirements of Specification 4.2.2.3 for Base Load operation are satisfied and enter Base Load operation.

g. The limits specified in Specifications 4.2.2.2c, 4.2.2.2e., and 4.2.2.2f. above are not applicable in the following core plane regions:

1. Lower core region from 0 to 15%, inclusive.

2. Upper core region from 85 to 100%, inclusive.

4.2.2.3 Base Load operation is permitted at powers above APL^{ND} if the following conditions are satisfied:

a. Prior to entering Base Load operation, maintain THERMAL POWER above APL^{ND} and less than or equal to that allowed by Specification 4.2.2.2 for at least the previous 24 hours. Maintain Base Load operation surveillance (AFD within ± 3% (UNIT 1) AND ± 5% (UNIT 2) of target flux difference) during this time period. Base Load operation is then permitted providing THERMAL POWER is maintained between APL^{ND} and APL^{BL} or between APL^{ND} and 100% (whichever is most limiting) and F_Q surveillance is maintained pursuant to Specification 4.2.2.4. APL^{BL} is defined as:

$$APL^{BL} = \underset{\text{over } Z}{\text{minimum}} \left[\frac{(2.15 \times K(Z)}{F_Q^M(Z) \times W(Z)_{BL}} \right] \times 100\% \text{ (UNIT 1)}$$

SAME EQUATION EXCEPT USE 2.26 INSTEAD OF 2.15 FOR UNIT 2

where: $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty. The F_Q limit is 2.15 (UNIT 1) AND 2.26 (UNIT 2). K(z) is given in Figure 3.2-2. $W(z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during base load operation. The function is given in the Peaking Factor Limit Report as per Specification 6.9.1.9.

b. During Base Load operation, if the THERMAL POWER is decreased below APL^{ND} then the conditions of 4.2.2.3.a shall be satisfied before re-entering Base Load operation.

4.2.2.4 During Base Load Operation $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER above APL^{ND}.

b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. VERIFY THE REQUIREMENTS OF SPECIFICATION 3.2.2 ARE SATISFIED.

c. Satisfying the following relationship:

$$F_Q^M(Z) \leq \frac{2.15 \times K(Z)}{P \times W(Z)_{BL}} \text{ for } P > APL^{ND} \text{ (unit 1)}$$

(SAME EQUATION EXCEPT USE 2.16 INSTEAD OF 2.15 FOR UNIT 2)

where: $F_Q^M(Z)$ is the measured $F_Q(Z)$. The F_Q limit is 2.15 (unit 1) and 2.16 (unit 2). K(Z) is given in Figure 3.2-2. P is the relative THERMAL POWER. $W(Z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during normal operation. This function is given in the Peaking Factor Limit Report as per Specification 6.9.1.9.

d. Measuring $F_Q^M(Z)$ in conjunction with target flux difference determination according to the following schedule:

1. Prior to entering BASE LOAD operation after satisfying Section 4.2.2.3 unless a full core flux map has been taken in the previous 31 EFPD with the relative thermal power having been maintained above APL^{ND} for the 24 hours prior to mapping, and

2. At least once per 31 effective full power days.

e. With measurements indicating

$$\underset{\text{over } Z}{\text{maximum}} \left[\frac{F_Q^M(Z)}{K(Z)} \right]$$

has increased since the previous determination $F_Q^M(Z)$ either of the following actions shall be taken:

1. $F_Q^M(Z)$ shall be increased by 2 percent over that specified in 4.2.2.4.c, or

2. $F_Q^M(Z)$ shall be measured at least once per 7 EFPD until 2 successive maps indicate that

$$\underset{\text{over } Z}{\text{maximum}} \left[\frac{F_Q^M(Z)}{K(Z)} \right] \text{ is not increasing.}$$

f. With the relationship specified in 4.2.2.4.c above not being satisfied, either of the following actions shall be taken:

1. Place the core in an equilbrium condition where the limit in 4.2.2.2.c is satisfied, and remeasure $F_Q^M(Z)$, or

2. Comply with the requirements of Specification 3.2.2 for $F_Q(Z)$ exceeding its limit by the percent calculated with one of the following expressions:

$$[(\text{max. over } z \text{ of } [\frac{F_Q^M(Z) \times W(Z)_{BL}}{\frac{2.15}{P} \times K(Z)}]) -1] \times 100 \text{ for } P \geq APL^{ND} \text{ (unit 1)}$$

SAME EXPRESSION EXCEPT USE 2.26 INSTEAD OF 2.15 FOR UNIT 2

$$\cancel{[(\text{max. over } z \text{ of } [\frac{F_Q^M(Z) \times W(Z)}{\frac{2.15}{P} \times K(Z)}]) -1] \times 100 \text{ for } 0.5 < P < APL^{ND}}$$

g. The limits specified in 4.2.2.4.c, 4.2.2.4.e, and 4.2.2.4.f above are not applicable in the following core plan regions:

1. Lower core region 0 to 15 percent, inclusive.

2. Upper core region 85 to 100 percent, inclusive.

4.2.2.5 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of specification 4.2.2.2 an overall measured $F_Q(z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

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SURVEILLANCE REQUIREMENTS (UNIT 2)

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER,

b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties,

c. Comparing the F_{xy} computed (F_{xy}^C) obtained in Specification 4.2.2b., above, to:

1) The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in Specifications 4.2.2.2e. and f., below, and

2) The relationship:

$$F_{xy}^L = F_{xy}^{RTP} \; [1+0.2(1-P)],$$

Where F_{xy}^L is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP} and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

1) When F_{xy}^C is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^L relationship, additional power distribution maps shall be taken and F_{xy}^C compared to F_{xy}^{RTP} and F_{xy}^L either:

a) Within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^C was last determined, or

b) At least once per 31 EFPD, whichever occurs first.

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SURVEILLANCE REQUIREMENTS (UNIT 2)

2) When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided for all core planes containing Bank "D" control rods and all unrodded core planes in a Radial Peaking Factor Limit Report per Specification 6.9.1.9.

f. The F_{xy} limits of Specificatio.. 4.2.2.2e., above, are not applicable in the following core planes regions as measured in percent of core height from the bottom of the fuel:

1) Lower core region from 0 to 15%, inclusive,

2) Upper core region from 85 to 100%, inclusive,

3) Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2% and 74.9 ± 2%, inclusive, and

4) Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the Bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L}, the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits.

4.2.2.3 When $F_Q(Z)$ is measured for other than F_{xy} determinations, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

REPLACE FIG WITH THE FOLLOWING FIGURE (2)

UNIT 2

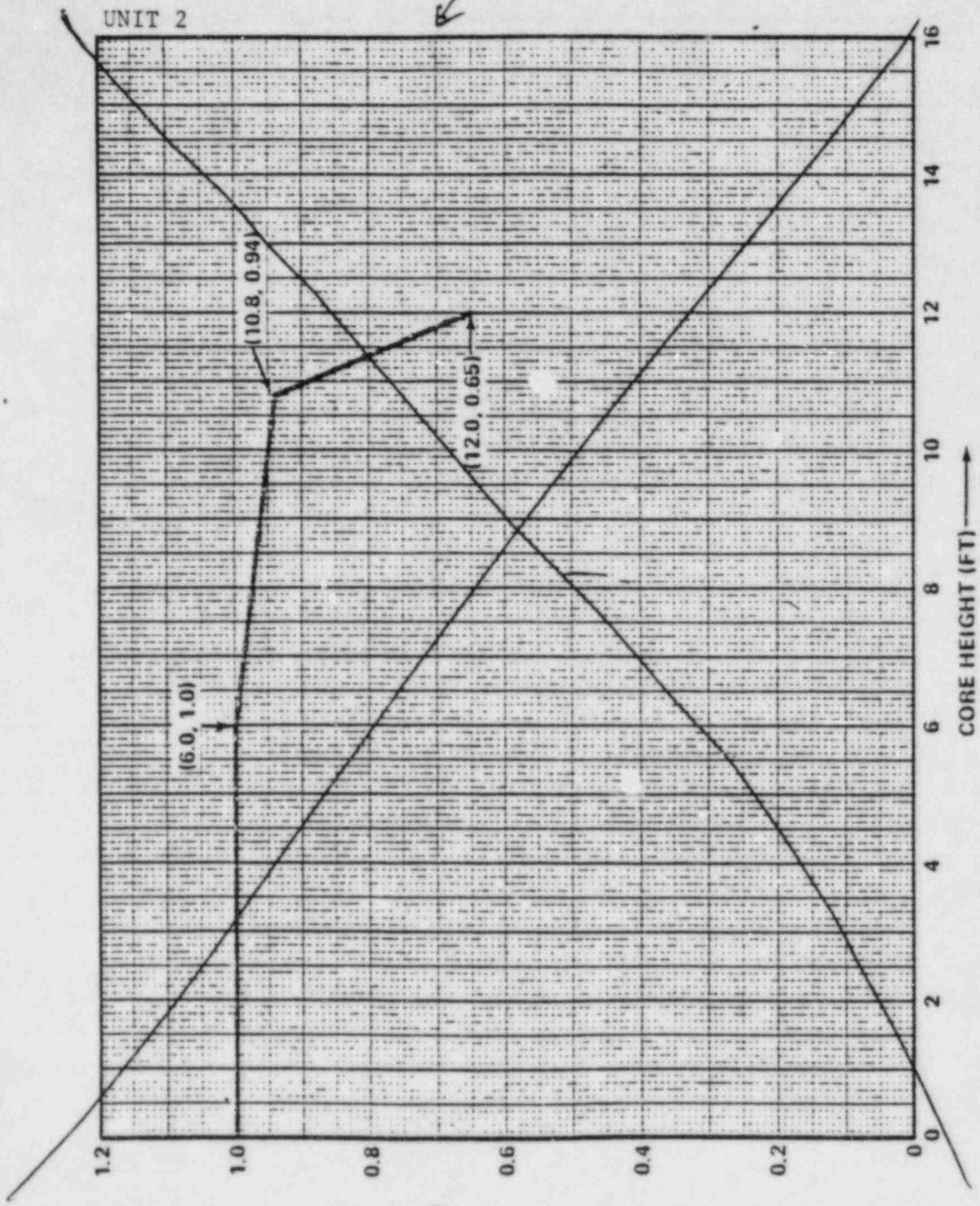
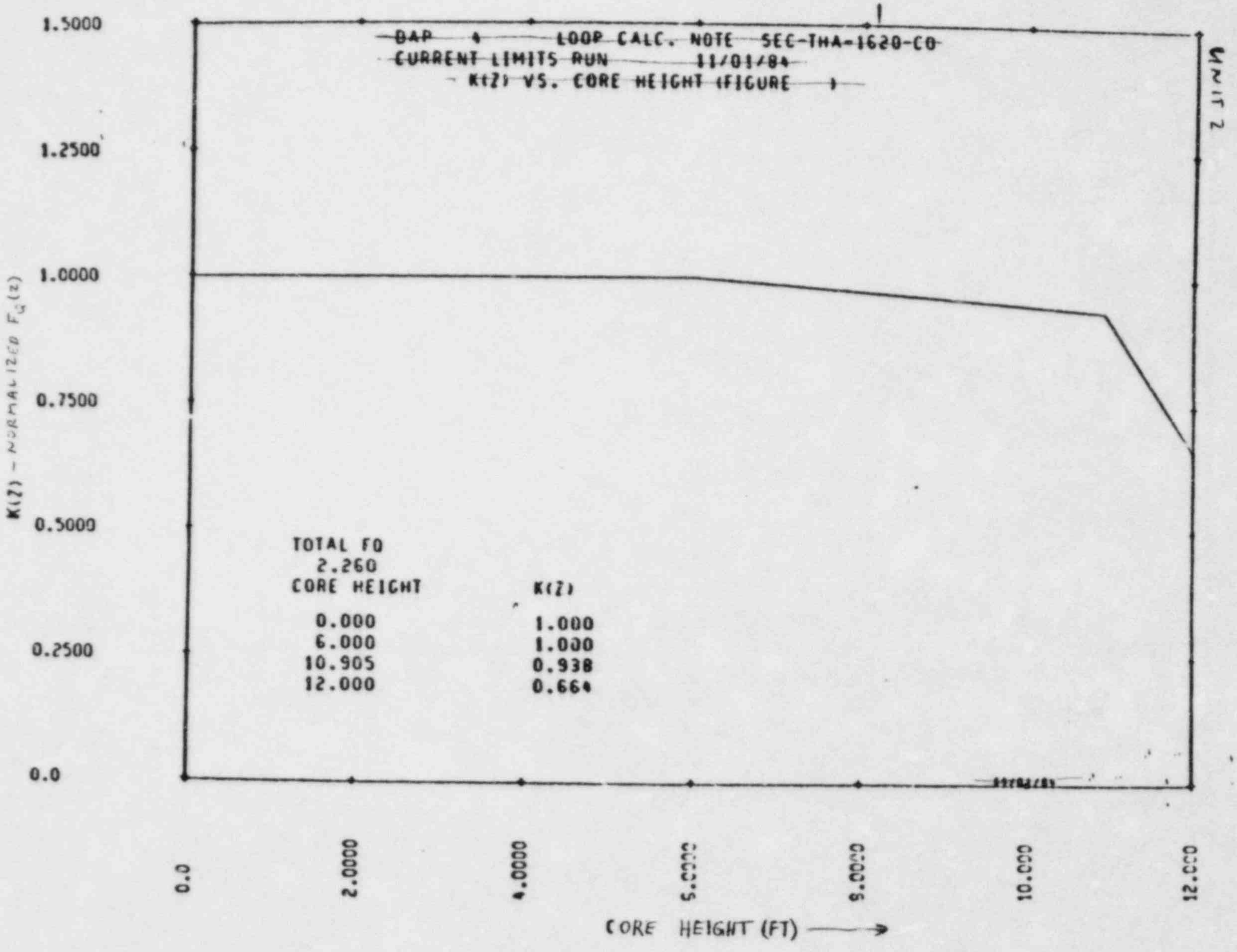


FIGURE 3.2-2b
K(Z) - NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT (UNIT 2)

FIGURE 2


UNIT 2
DAP 4 LOOP CALC. NOTE SEC-THA-1620-C0
CURRENT LIMITS RUN 11/01/84
K(Z) VS. CORE HEIGHT (FIGURE)
TOTAL FQ
2.260
CORE HEIGHT K(Z)
0.000 1.000
6.000 1.000
10.905 0.938
12.000 0.664
K(Z) - NORMALIZED $F_Q(z)$
1.5000
1.2500
1.0000
0.7500
0.5000
0.2500
0.0
0.0
2.0000
4.0000
6.0000
8.0000
10.000
12.000
CORE HEIGHT (FT)

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and $R_{\cancel{1}}$, ~~R_2~~ shall be maintained within the region of allowable operation shown on Figure 3.2-3 for four loop operation:

Where:

a. $R_{\cancel{1}}$ ~~(Unit 1)~~ $= \frac{F^N_{\Delta H}}{1.49\,[1.0 + 0.3\,(1.0 - P)]}$ ~~R_1 (Unit 2) $= \frac{F^N_{\Delta H}}{1.49\,[1.0 + 0.2\,(1.0}$~~

~~b. R_2 (Unit 1) = R_1, R_2 (Unit 2) = $\frac{R1}{[1-RBP(BU)]}$~~

b. ~~c.~~ $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, AND

c. ~~d.~~ $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R since Figure 3.2-3 includes penalties for undetected feedwater venturi fouling of 0.1% and for measurement uncertainties of 1.7% for flow and 4% for incore measurement of $F^N_{\Delta H}$. ~~and~~

~~e. RBP (BU) = Rod Bow Penalty as a function of region average burnup as shown in Figure 3.2-4, where a region is defined as those assemblies with the same loading date (reloads) or enrichment (first core). (Applies to Unit 2 only).~~

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and $R_{\cancel{1}}$, ~~R_2~~ outside the region of acceptable operation shown on Figure 3.2-3:

a. Within 2 hours either:

1. Restore the combination of RCS total flow rate and $R_{\cancel{1}}$, ~~R_2~~ to within the above limits, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of R_1 ~~R_2~~, and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b. above; subsequent POWER OPERATION may proceed provided that the combination of R_1 ~~R_2~~ and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation shown on Figure 3.2-3 prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,
2. A nominal 75% of RATED THERMAL POWER, and
3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate determined by process computer readings or digital voltmeter measurement and R_1 ~~and R_2~~ shall be within the region of acceptable operation of Figure 3.2.3:

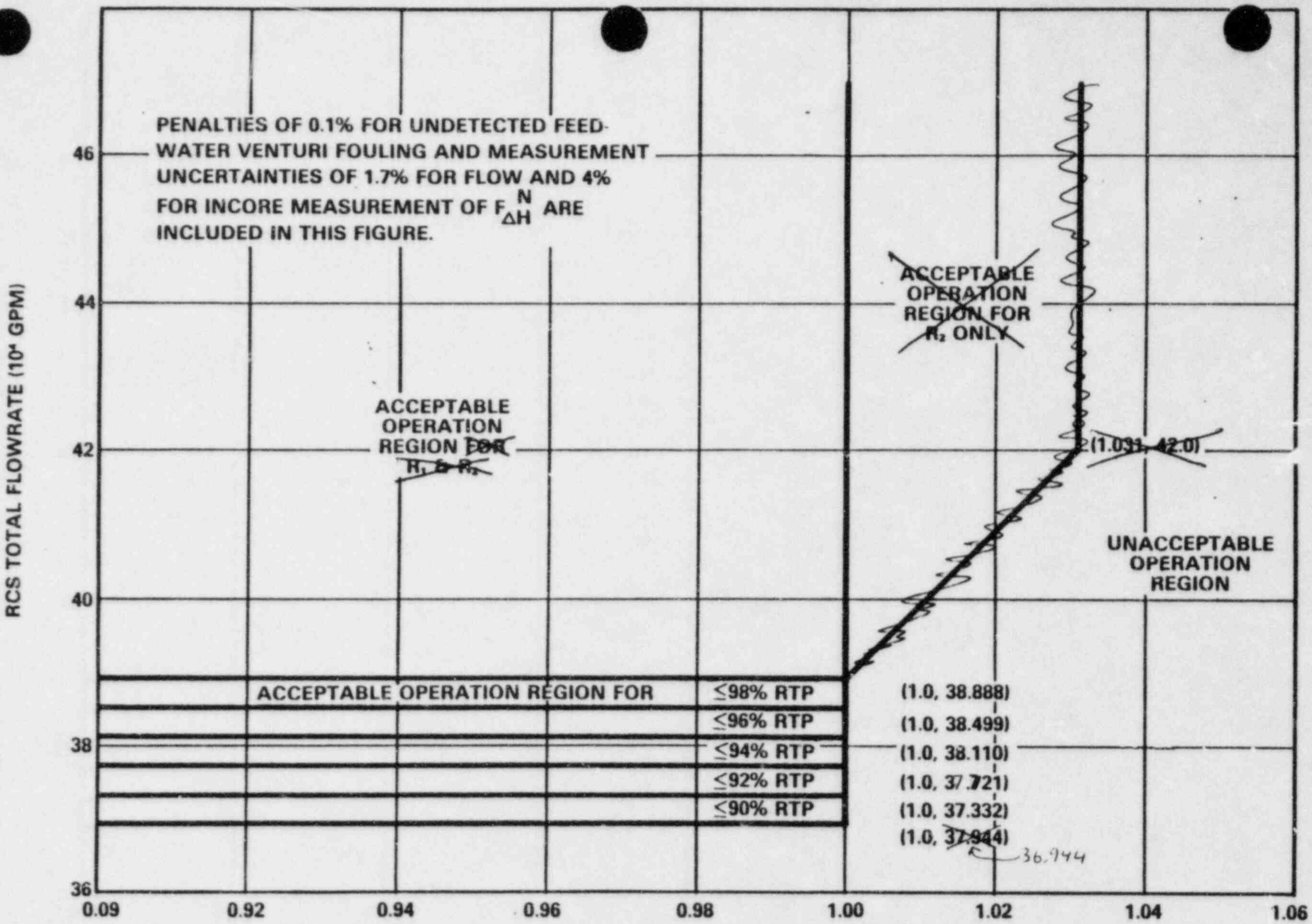
a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation of Figure 3.2-3 at least once per 12 hours when the most recently obtained value ~~s~~ of R_1 ~~and R_2~~, obtained per Specification 4.2.3.2, are assumed to exist.

4.2.3.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.3.5 The RCS total flow rate shall be determined by precision heat balance measurement at least once per 18 months.



$$R_X = F_{\Delta H}^{N}/1.49[1.0 + \overset{0.3}{\cancel{0.2}}\,(1.0\text{-}P)]$$

$$\cancel{R_2 = R_1/[1 - RBP(BU)]}$$

Figure 3.2-3b RCS FLOW RATE VERSUS R_X ~~and R_2~~ - FOUR LOOPS IN OPERATION (Unit 2)

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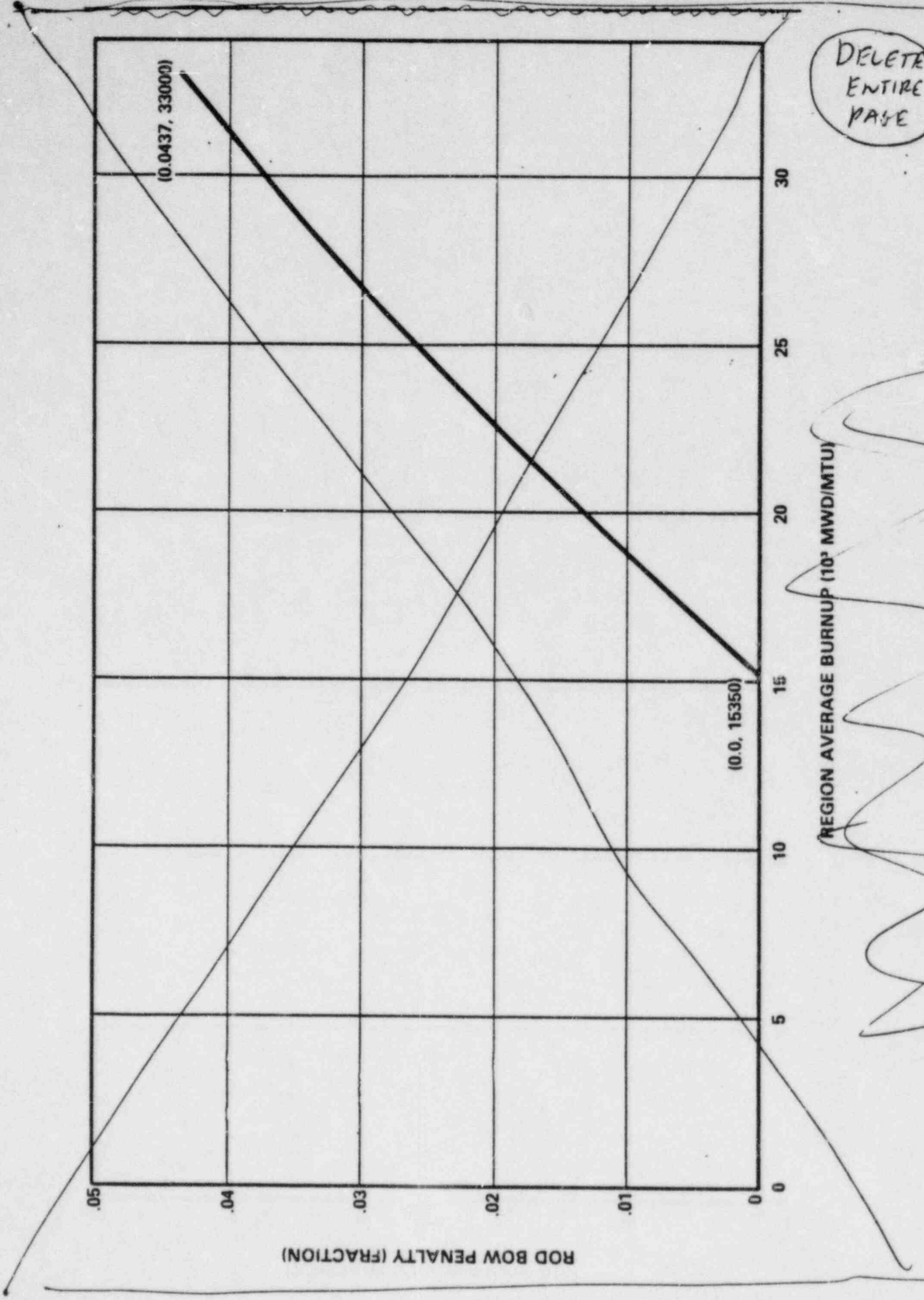

FIGURE 3.2-4 ROD BOW PENALTY AS A FUNCTION OF BURNUP (Unit 2)

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	N.A.
2. Power Range, Neutron Flux	≤ 0.5 second*
3. Power Range, Neutron Flux, High Positive Rate	N.A.
4. Power Range, Neutron Flux, High Negative Rate	≤ 0.5 second*
5. Intermediate Range, Neutron Flux	N.A.
6. Source Range, Neutron Flux	N.A.
7. Overtemperature ΔT	≤ 6.0 (unit 1), 8.0 (unit 2) seconds*
8. Overpower ΔT	≤ 6.0 (unit 1), 8.0 (unit 2) seconds*
9. Pressurizer Pressure--Low	≤ 2.0 seconds
10. Pressurizer Pressure--High	≤ 2.0 seconds
11. Pressurizer Water Level--High	N.A.

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. Low Reactor Coolant Flow	
a. Single Loop (Above P-8)	≤ 1.0 second
b. Two Loops (Above P-7 and below P-8)	≤ 1.0 second
13. Steam Generator Water Level--Low-Low	≤ 2.0 (unit 1), 3.5 (unit 2) seconds
14. Undervoltage-Reactor Coolant Pumps	< 1.5 seconds
15. Underfrequency-Reactor Coolant Pumps	< 0.6 second
16. Turbine Trip	
a. Low Fluid Oil Pressure	N.A.
b. Turbine Stop Valve Closure	N.A.
17. Safety Injection Input from ESF	N.A.
18. Reactor Trip System Interlocks	N.A.
19. Reactor Trip Breakers	N.A.
20. Automatic Trip and Interlock Logic	N.A.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
7. Auxiliary Feedwater		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Steam Generator Water Level--Low-Low		
1) Start Motor-Driven Pumps	(unit 1), 40.0% (unit 2) ≥ 12% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ 54.9% of span at 100% of RATED THERMAL POWER.	(unit 1), 39.0% (unit 2) ≥ 11% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ 53.9% of span at 100% of RATED THERMAL POWER.
2) Start Turbine-Driven Pumps	≥ 12% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ 54.9% of span at 100% of RATED THERMAL POWER. (unit 1), 40.0% (unit 2)	≥ 11% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ 53.9% of span at 100% of RATED THERMAL POWER. (unit 1), 39.0% (unit 2)
d. Auxiliary Feedwater Suction Pressure - Low (Suction Supply Automatic Realignment)	≥ 2 psig	≥ 1 psig
e. Safety Injection - Start Motor-Driven Pumps	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values	
f. Station Blackout - Start Motor-Driven Pumps and Turbine-Driven Pump	3464 ± 173 volts with a 8.5 ± 0.5 second time delay	≥ 3200 volts
g. Trip of Main Feedwater Pumps - Start Motor-Driven Pumps	N.A.	N.A.

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The Movable Incore Detection System shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of two detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABILITY: When the Movable Incore Detection System is used for:

a. Recalibration of the Excore Neutron Flux Detection System,

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^{N}_{\Delta H}$, AND $F_Q(Z)$. ~~and F_{xy}~~

ACTION:

With the Movable Incore Detection System inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The Movable Incore Detection System shall be demonstrated OPERABLE at least once per 24 hours by normalizing each detector output when required for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^{N}_{\Delta H}$, AND $F_Q(Z)$~~, and F_{xy}~~.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1.1 Each cold leg injection accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between ~~1)~~ 8022 and 8256 gallons, ~~(Unit 1),~~
~~2) 8261 and 8496 gallons (Unit 2),~~

c. A boron concentration of between 1900 and 2100 ppm,

d. A nitrogen cover-pressure of between 430 and 484 psig ~~(Unit 1)~~, AND ~~400 and 454 psig (Unit 2), and~~

e. A water level and pressure channel OPERABLE.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one cold leg injection accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one cold leg injection accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 1 hour and in HOT SHUTDOWN within the following 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1.1 Each cold leg injection accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each cold leg injection accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

EMERGENCY CORE COOLING SYSTEMS

UPPER HEAD INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1.2 Each Upper Head Injection Accumulator System shall be OPERABLE with:

a. The isolation valves open,

b. The water-filled accumulator containing a minimum of 1850 cubic feet of borated water having a concentration of between 1900 and 2100 ppm of boron, and

c. The nitrogen bearing accumulator pressurized to between 1206 and 1264 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

REPLACE WITH INSERT Y

~~Above 46% RATED THERMAL POWER:~~

~~a. With the Upper Head Injection Accumulator System inoperable, except as a result of a closed isolation valve(s), restore the Upper Head Injection Accumulator System to OPERABLE status within 1 hour or be at less than or equal to 46% RATED THERMAL POWER and close the isolation valves within the next 6 hours.~~

~~b. With the Upper Head Injection Accumulator System inoperable due to the isolation valve(s) being closed, either immediately open the isolation valve(s) or be at less than or equal to 46% RATED THERMAL POWER and close the remaining isolation valves within 1 hour.~~

~~Less than or equal to 46% RATED THERMAL POWER:~~

~~a. With the Upper Head Injection Accumulator System inoperable, POWER OPERATION may continue provided the isolation valves are closed within 6 hours.~~

~~b. The provisions of Specification 3.0.4 are not applicable.~~

SURVEILLANCE REQUIREMENTS

4.5.1.2 Each Upper Head Injection Accumulator System shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the contained borated water volume and nitrogen pressure in the accumulators, and

2) Verifying that each accumulator isolation valve is open.

*Pressurizer Pressure above 1900 psig.

INSERT Y

a. With the Upper Head Injection Accumulator System inoperable, except as a result of a closed isolation valve(s), restore the Upper Head Injection Accumulator System to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the Upper Head Injection Accumulator System inoperable due to the isolation valve(s) being closed, either immediately open the isolation valve(s) or be in HOT STANDBY within 1 hour and be in HOT SHUTDOWN within the next 12 hours.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of ~~1.6% of delta k/k (Unit 2),~~ 1.3% delta k/k ~~(Unit 1)~~ is required to control the reactivity transient.
Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% delta k/k SHUTDOWN MARGIN provides adequate protection.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value -4.1×10^{-4} delta k/k/°F. The MTC value of -3.2×10^{-4} delta k/k/°F represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value of -4.1×10^{-4} k/k/°F.

BASES

MODERATOR TEMPERATURE COEFFICIENT (Continued)

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within it analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, (5) associated Heat Tracing Systems, and (6) an emergency power supply from OPERABLE diesel generators.

1.3%

With the RCS average temperature above 200°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of ~~1.6%~~ delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 16,321 gallons of 7000-ppm borated water from the boric acid storage tanks or 75,000 gallons of 2000-ppm borated water from the refueling water storage tank (RWST).

With the RCS temperature below 200°F, one Boron Injection System is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single Boron Injection System becomes inoperable.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below 300°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the calculated DNBR in the core at or above the design limit during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding 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.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ 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;

$F^N_{\Delta H}$ 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; and

~~Unit No. 2~~

$F_{xy}(Z)$ ~~Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.~~

3/4.2.1 AXIAL FLUX DIFFERENCE

2.26

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of ~~2.32~~ (Unit 2), 2.15 (Unit 1) times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full-length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

AXIAL FLUX DIFFERENCE (Continued)

Unit No. 2

Although it is intended that the plant will be operated with the AFD within the target band required by Specification 3.2.1 about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band.

Unit No. 1

~~For Unit 1, a~~At power levels below APL^{ND}, the limits on AFD are defined by Figures 3.2-1, i.e. that defined by the RAOC operating procedure and limits. These limits were calculated in a manner such that expected operational transients, e.g. load follow operations, would not result in the AFD deviating outside of those limits. However, in the event such a deviation occurs, the short period of time allowed outside of the limits at reduced power levels will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prevent operation in the vicinity of the APL^{ND} power level.

At power levels greater than APL^{ND}, two modes of operation are permissible; 1) RAOC, the AFD limit of which are defined by Figure 3.2-1, and 2) Base Load operation, which is defined as the maintenance of the AFD within a ±3% (UNIT 1), ±5% (UNIT 2) band about a target value. The RAOC operating procedure above APL^{ND} is the same as that defined for operation below APL^{ND}. However, it is possible when following extended load following maneuvers that the AFD limits may result in restrictions in the maximum allowed power or AFD in order to guarantee operation with $F_Q(z)$ less than its limiting value. To allow operation at the maximum permissible value, the Base Load operating procedure restricts the

AXIAL FLUX DIFFERENCE (Continued)

indicated AFD to relatively small target band and power swings (AFD target band of ~~±3%,~~ ±3% (unit 1), ±5% (unit 2), $APL^{ND} \leq$ power $\leq APL^{BL}$ or 100% Rated Thermal Power, whichever is lower). For Base Load operation, it is expected that the plant will operate within the target band. Operation outside of the target band for the short time period allowed will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prohibit continued operation in the power region defined above. To assure there is no residual xenon redistribution impact from past operation on the Base Load operation, a 24 hour waiting period at a power level above APL^{ND} and allowed by RAOC is necessary. During this time period load changes and rod motion are restricted to that allowed by the Base Load procedure. After the waiting period extended Base Load operation is permissible.

~~For Unit 1,~~ The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are: 1) outside the allowed ΔI power operating space (for RAOC operation), or 2) outside the allowed ΔI target band (for Base Load operation). These alarms are active when power is greater than: 1) 50% of RATED THERMAL POWER (for RAOC operation), or 2) APL^{ND} (for Base Load operation). Penalty deviation minutes for Base Load operation are not accumulated based on the short period of time during which operation outside of the target band is allowed.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flow rate, and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded, and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

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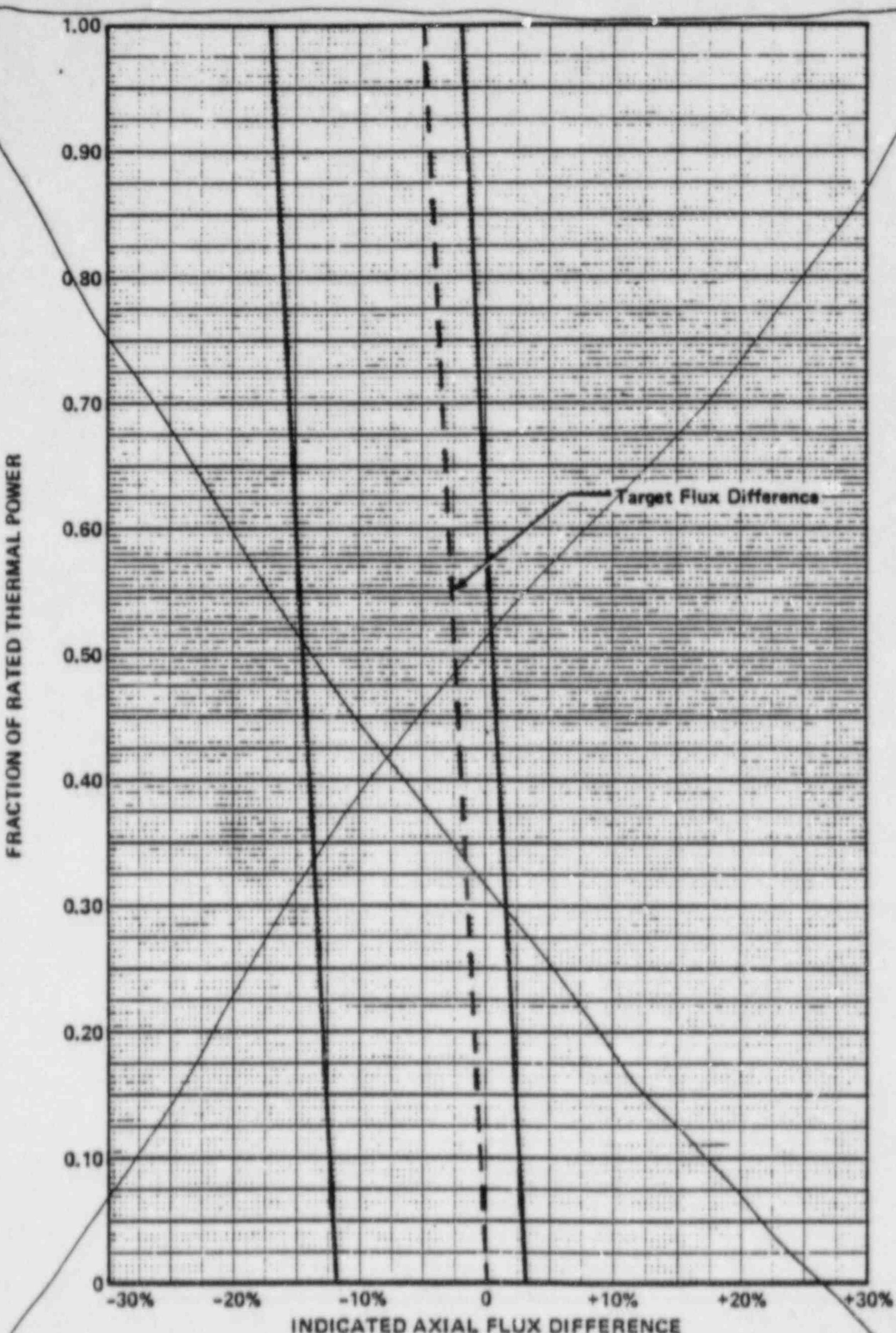

FIGURE B 3/4 2-1.
TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

AMENDMENT NO. (UNIT 1)
AMENDMENT NO. (UNIT 2)

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 13 steps from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. As noted on Figure~~s~~ 3.2-3 ~~and 3.2-4~~, RCS flow rate and ~~$F^N_{\Delta H}$~~ POWER may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the ~~measured $F^N_{\Delta H}$ is also low~~ POWER LEVEL IS DECREASED) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R_x as calculated in Specification 3.2.3 and used in Figure 3.2-3, accounts for $F^N_{\Delta H}$ less than or equal to 1.49. This value is used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed. ~~R_2, as defined,~~ allows for the inclusion of a penalty for Rod Bow on DNBR only. Thus, knowing the "as measured" values of $F^N_{\Delta H}$ and RCS flow allows for "tradeoffs" in excess of R equal to 1.0 for the purpose of offsetting the Rod Bow DNBR penalty.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to partially offset this reduction. This credit comes from a generic or plant-specific design margin. For McGuire Unit 2, the margin used to partially offset rod bow penalties is 9.1%. This margin breaks down as follows:

1)	Design limit DNBR	1.6%
2)	Grid spacing K_s	2.9%
3)	Thermal Diffusion Coefficient	1.2%
4)	DNBR Multiplier	1.7%
5)	Pitch Reduction	1.7%

HEAT FLUX HOT CHANNEL FACTOR and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

~~However, the margin used to partially offset rod bow penalties is 5.9% with the remaining 3.2% used to trade off against measured flow being as much as 2% lower than thermal design flow plus uncertainties. The penalties applied to $F^N_{\Delta H}$ to account for rod bow (Figure 3.2-4) as a function of burnup are consistent with those described in Mr. John F. Stolz's (NRC) letter to T. M. Anderson (Westinghouse) dated April 5, 1979 with the difference being due to the amount of margin each unit uses to partially offset rod bow penalties.~~

~~For McGuire Unit 1,~~ Margin between the safety analysis limit DNBRs (1.47 and 1.49 for thimble and typical cells, respectively) and the design limit DNBRs (1.32 and 1.34 for thimble and typical cells, respectively) is maintained. A fraction of this margin is utilized to accommodate the transition core DNBR penalty (2%) and the appropriate fuel rod bow DNBR penalty (WCAP - 8691, Rev. 1)

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of Figure~~s~~ 3.2-3, ~~and 3.2-4~~. Measurement errors of 1.7% for RCS total flow rate and 4% for $F^N_{\Delta H}$ have been allowed for in determination of the design DNBR value.

The measurement error for RCS total flow rate is based upon performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a non-conservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi is included in Figure 3.2-3. Any fouling which might bias the RCS flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e., either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The 12-hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation shown on Figure 3.2-3.

RADIAL PEAKING FACTOR LIMIT REPORT

Unit No. 2

6.9.1.9 The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided to the Regional Administrator of the NRC Regional Office, with a copy to the Director, Nuclear Reactor Regulation, Attention: Chief, Core Performance Branch, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555 for all core planes containing Bank "D" control rods and all unrodded core planes at least 60 days prior to cycle initial criticality. In the event that the limit would be submitted at some other time during core life, it shall be submitted 60 days prior to the date the limit would become effective unless otherwise exempted by the Commission.

Any information needed to support F_{xy}^{RTP} will be by request from the NRC and need not be included in this report.

Unit No. 1

6.9.1.9 The W(z) functions for RAOC and Base Load operation and the value for APL^{ND} (as required) shall be provided to the Director, Nuclear Reactor Regulations, Attention: Chief, Core Performance Branch, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 at least 60 days prior to cycle initial criticality. In the event that these values would be submitted at some other time during core life, it will be submitted 60 days prior to the date the values would become effective unless otherwise exempted by the Commission.

Any information needed to support W(z), $W(z)_{BL}$ and APL^{ND} will be by request from the NRC and need not be included in this report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report.

ATTACHMENT 2
JUSTIFICATION AND SAFETY ANALYSIS

Mr. H. B. Tucker's (DPC) November 14, 1983 letter to Mr. H. R. Denton (NRC/ONRR) described planned changes in the fuel design for McGuire Nuclear Station, Units 1 and 2. McGuire Unit 2 has been operating with a Westinghouse 17x17 low-parasitic (STD) fueled core. It is planned to refuel Unit 2 with Westinghouse 17x17 Reconstitutable Optimized Fuel Assembly (OFA) regions. As a result, future core loadings would range from an approximately 1/3 OFA - 2/3 STD transition core to eventually an all OFA fueled core. McGuire Unit 1 is currently operating with the first such OFA reload region (Cycle 2), with the second OFA region scheduled for the upcoming Cycle 3 refueling. The OFA fuel has similar design features compared to the STD fuel which has had substantial operating experience in a number of nuclear plants. The major differences are the use of six intermediate (mixing vane) Zircaloy grids for the OFA fuel versus six intermediate (mixing vane) Inconel grids for STD fuel and a reduction in fuel rod diameter. Major advantages for utilizing the OFA are: (1) increased efficiency of the core by reducing the amount of parasitic material and (2) reduced fuel cycle costs due to an optimization of the water to uranium ratio.

The above letter provided a Reference Safety Evaluation Report summarizing the evaluation/analysis performed on the region-by-region reload transition from the McGuire Units 1 and 2 STD fueled cores to cores with all optimized fuel. The report examined the differences between the Westinghouse OFA and STD designs and evaluated the effects of these differences for the transition to an all OFA core. The evaluation considered the standard reload design methods described in WCAP-9272 and 9273, "Westinghouse Reload Safety Evaluation Methodology," and the transition effects described for mixed cores in Chapter 18 of WCAP-9500-A, "Reference Core Report - 17x17 Optimized Fuel Assembly." Consistent with the Westinghouse STD reload methodology for analyzing cycle specific reloads, parameters were chosen to maximize the applicability of the transition evaluations for each reload cycle and to facilitate subsequent determination of the applicability of 10 CFR 50.59. Subsequent cycle specific reload safety evaluations will verify that applicable safety limits are satisfied based on the reference evaluation/analyses established in the reference report. A summary of the mechanical, nuclear, thermal and hydraulic, and accident evaluations for the McGuire Units 1 and 2 transitions to an all OFA core are given in the reference report.

WCAP-8183, "Operational Experience with Westinghouse Cores," presents the operating experience through December 31, 1983 of six 17x17 OFA demonstration assemblies (two in each of three reactors) which have the McGuire 1 and 2 design features. During 1983 four assemblies operated in their fourth cycle and were expected to achieve burnups of 39,000 and 35,000 MWD/MTU respectively during the first quarter of 1984, and two others completed their second cycle of irradiation with a burnup of 22,000 MWD/MTU and were operating in their third cycle. All demonstration 17x17 OFAs examined were in good or excellent condition. This provides evidence of favorable operation of Zircaloy grids and reduced fuel rod diameters which are the major new design features of the 17x17 OFA. In addition, Maanshan Unit 1 was scheduled to begin irradiating a full core of 17x17 OFAs during the first half of 1984, and McGuire Unit 1 has operated nearly a full cycle with an OFA reload region (60 17x17 OFA assemblies).

The results of evaluation/analysis and tests described in the Reference Safety Evaluation Report lead to the following conclusions:

a. The Westinghouse OFA reload fuel assemblies for McGuire 1 and 2 are mechanically compatible with the current STD design, control rods, and reactor internals interfaces. Both fuel assemblies satisfy the current design bases for the McGuire units.

b. Changes in the nuclear characteristics due to the transition from STD to OFA fuel will be within the range normally seen from cycle to cycle due to fuel management effects.

c. The reload OFAs are hydraulically compatible with the current STD design.

d. The accident analyses for the OFA transition core were shown to provide acceptable results by meeting the applicable criteria, such as, minimum DNBR, peak pressure, and peak clad temperature, as required. The previously reviewed and licensed safety limits are met. Analyses in support of this safety evaluation establish a reference design on which subsequent reload safety evaluations involving OFA reloads can be based. (Attachment 2A of H. B. Tucker's December 12, 1983 Unit 1/Cycle 2 OFA reload submittal presents those detailed non-LOCA and LOCA accident analyses of the McGuire Units 1 and 2 FSAR impacted by the proposed changes as determined in Section 6.0 of the Reference Safety Evaluation Report. The information contained within was prepared using the NRC Standard Format and Content Guide, Regulatory Guide 1.70, Revision 3 as it applies to McGuire Nuclear Station Units 1 and 2).

e. Plant operating limitations given in the Technical Specifications affected by use of the OFA design and positive MTC will be satisfied with the proposed changes noted in Section 7.0 of the report.

Attachment 2A is the cycle-specific Reload Safety Evaluation (RSE) for McGuire Unit 2/Cycle 2 including Fq surveillance and RAOC/Base Load Technical Specifications. The RSE presents an evaluation for McGuire Unit 2, Cycle 2, which demonstrates that the core reload will not adversely affect the safety of the plant. This evaluation was performed utilizing the methodology described in WCAP-9273, "Westinghouse Reload Safety Evaluation Methodology." In addition, the NRC has previously approved a similar OFA reload for McGuire Unit 1 via Ms. E. G. Adensam's (NRC/ONRR) April 20, 1984 letter to H. B. Tucker (note that base load operation technical specifications were previously approved for Unit 1 by Ms. Adensam's letters dated June 21 and September 13, 1984).

McGuire Unit 2 is operating in Cycle 1 with all Westinghouse 17x17 low parasitic (STD) fuel assemblies. For Cycle 2 and subsequent cycles, it is planned to refuel the McGuire Unit 2 core with Westinghouse 17x17 optimized fuel assembly (OFA) regions. In the OFA transition licensing submittal to the NRC (Reference Safety Evaluation, November 14, 1983 letter) an analyses of the safety aspects of the transition from STD fuel design to OFA design was provided. This licensing submittal justified the compatibility of the OFA design with the STD design in a transition core as well as a full OFA core. The OFA transition licensing submittal contained mechanical, nuclear, thermal-hydraulic, and accident evaluations which are applicable to the Cycle 2 safety evaluation.

All of the accidents comprising the licensing bases which could potentially be affected by the fuel reload have been reviewed for the Cycle 2 design described herein. The results of new analyses are included in the above-mentioned licensing submittal and in the cycle-specific Reload Safety Evaluation, and the justification or the applicability of previous results for the remaining analyses is presented.

The McGuire Unit 2, Cycle 2 reactor core will be comprised of 193 fuel assemblies arranged in the core loading pattern configuration shown in Figure 1 of the Cycle 2 Reload Safety Evaluation. During the Cycle 1/2 refueling, 60 STD fuel assemblies will be replaced with 60 Region 4 optimized fuel assemblies. A summary of the Cycle 2 fuel inventory is given in Table 1 of the Cycle 2 Reload Safety Evaluation.

From the evaluation presented in the Cycle 2 Reload Safety Evaluation, it is concluded that the Cycle 2 design does not cause the previously acceptable safety limits to be exceeded. This conclusion is based on the following:

1. Cycle 1 burnup is between 14400 and 15400 MWD/MTU.

2. Cycle 2 burnup is limited to 10700 MWD/MTU including a coastdown.

3. There is adherence to all plant operating limitations given in the Technical Specifications as revised by the proposed changes submitted in support of the OFA transition licensing submittal and the changes given in Appendix A of the Cycle 2 RSE.

To ensure plant operation consistent with the design and safety evaluation conclusion statements made in the Cycle 2 RSE and to ensure that these conclusions remain valid, several Technical Specifications changes will be needed for Cycle 2. These changes are those outlined in Section 7.0 of the OFA transition licensing submittal and the changes given in Appendix A of the cycle-specific RSE. Differences between the cycle-specific RSE Technical Specification changes to those given in the OFA transition licensing submittal are discussed in the cycle-specific RSE, along with any necessary justifications. In addition to these changes, Technical Specification 3.5.1.2 is revised to reflect the fact that the analysis performed to allow operation at less than or equal to 46% rated thermal power with the upper head injection accumulator system inoperable which was the bases for a recent Technical Specification change (Amendment Nos. 37 and 18 to McGuire Nuclear Station Units 1 and 2 Facility Operating Licenses NPF-9 and NPF-17, respectively) is valid only for the STD fuel design, and thus will not be applicable once the OFA reload occurs. Consequently, the specification is revised back to the way it was prior to Amendment Nos. 37/18. Note that Amendment Nos. 37/18 inadvertently revised the specification to be applicable to both Units 1 and 2 although McGuire Unit 1 has already had an OFA reload, invalidating the change for Unit 1 (i.e. the specification should have indicated that the change applies to Unit 2 only). Therefore, the revision changes the application to Unit 1 also (this change is conservative). In the interim the additional provisions of Amendment Nos. 37/18 will not be applied to McGuire Unit 1 through the use of administrative controls. Attachment 1 provides copies of these specifications as they presently appear in the McGuire Units 1 and 2 Technical Specifications with the appropriate changes noted. Certain changes are made such that they affect McGuire Unit 1 as well as Unit 2 (as opposed to indicating that they apply to Unit 2 only), but these constitute only administrative-type changes (corrections of minor errors/typos, clarifications, etc.) or are improvements incorporated for the Unit 2 specifications which are more conservative than the existing Unit 1 specifications. There are no changes which solely affect Unit 1.

The Peaking Factor Limit Report for McGuire Unit 2/Cycle 2 which will be submitted in accordance with the proposed Unit 2 Technical Specification 6.9.1.9 as given in Attachment 1 provides the elevation dependent W(z) values that are to be used as inputs to define the appropriate fitting coefficients for W(z) interpolations to be performed as a function of cycle burnup and axial elevation for RAOC and Base Load Operation, and the value for APL^{ND}. The appropriate W(z) function is used to confirm that the Heat Flux Hot Channel Factor, $F_Q(z)$, will be limited to the values specified in the Technical Specifications.

ATTACHMENT 2A

RELOAD SAFETY EVALUATION

MCGUIRE NUCLEAR STATION

UNIT 2 CYCLE 2

October, 1984

Edited by: P. Schueren

Approved: L. H. Boman

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1.0 INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

This report presents an evaluation for McGuire Unit 2, Cycle 2, which demonstrates that the core reload will not adversely affect the safety of the plant. This evaluation was performed utilizing the methodology described in WCAP-9273, "Westinghouse Reload Safety Evaluation Methodology"(1).

McGuire Unit 2 is operating in Cycle 1 with all Westinghouse 17x17 low parasitic (STD) fuel assemblies. For Cycle 2 (expected startup early 1985) and subsequent cycles, it is planned to refuel the McGuire Unit 2 core with Westinghouse 17x17 optimized fuel assembly (OFA) regions. In the OFA transition licensing submittal(2) to the NRC, approval was requested for the transition from the STD fuel design to the OFA design and the associated proposed changes to the McGuire Units 1 and 2 Technical Specifications. The licensing submittal justifies the compatibility of the OFA design with the STD design in a transition core as well as a full OFA core. The OFA transition licensing submittal(2) contains mechanical, nuclear, thermal-hydraulic, and accident evaluations which are applicable to the Cycle 2 safety evaluation.

All of the accidents comprising the licensing bases(2,3) which could potentially be affected by the fuel reload have been reviewed for the Cycle 2 design described herein. The results of new analyses are included in the above mentioned licensing submittal and in this evaluation, and the justification for the applicability of previous results for the remaining analyses is presented.

1.2 GENERAL DESCRIPTION

The McGuire Unit 2, Cycle 2 reactor core will be comprised of 193 fuel assemblies arranged in the core loading pattern configuration shown in Figure 1. During the Cycle 1/2 refueling, 60 STD fuel assemblies will be replaced with 60 Region 4 optimized fuel assemblies. A summary of the Cycle 2 fuel inventory is given in Table 1.

Nominal core design parameters utilized for Cycle 2 are as follows:

Core Power (MWt)	3411
System Pressure (psia)	2250
Core Inlet Temperature (°F)	558.5
Thermal Design Flow (gpm)	382,000
Average Linear Power Density (kw/ft) (based on 144" active fuel length)	5.43

1.3 CONCLUSIONS

From the evaluation presented in this report, it is concluded that the Cycle 2 design does not cause the previously acceptable safety limits to be exceeded. This conclusion is based on the following:

1. Cycle 1 burnup is between 14400 and 15400 MWD/MTU.

2. Cycle 2 burnup is limited to 10700 MWD/MTU including a coastdown.

3. The analyses and proposed Technical Specification changes submitted in support of the OFA transition licensing submittal(2) are approved by the NRC prior to Cycle 2 startup.

4. With the changes submitted in support of the OFA transition licensing submittal(2) and the Technical Specification changes given in Appendix A, there is adherence to all plant operating limitations in the Technical Specification.

2.0 REACTOR DESIGN

2.1 MECHANICAL DESIGN

The new Region 4 fuel assemblies are Westinghouse OFAs. The mechanical description and justification of their compatibility with the Westinghouse STD design in a transition core is presented in the OFA transition licensing submittal.[2]

The OFAs and Core Components are designed to be handled by existing handling tools. The control rods, thimble plugs, burnable absorber rods, and source rods are compatible with both the STD and OFA designs.

Table 1 presents a comparison of pertinent design parameters of the various fuel regions. The Region 4 fuel has been designed according to the fuel performance model[4]. The fuel is designed and operated so that clad flattening will not occur, as predicted by the Westinghouse clad flattening model[5]. For all fuel regions, the fuel rod internal pressure design basis, which is discussed and shown acceptable in Reference 6, is satisfied.

Westinghouse has had considerable experience with Zircaloy clad fuel. This experience is described in WCAP-8183, "Operational Experience with Westinghouse Cores."[7] Operating experience for Zircaloy grids has also been obtained from six demonstration 17x17 OFAs and four demonstration 14x14 OFAs. This experience is summarized in the OFA transition licensing submittal.[2]

2.2 NUCLEAR DESIGN

The Cycle 2 core loading is designed to meet a $F_Q(z) \times P$ ECCS limit of $\leq 2.26 \times K(z)$.

Relaxed Axial Offset Control (RAOC) will be employed in Cycle 2 to enhance operational flexibility during non-steady state operation. RAOC makes use of available margin by expanding the allowable ΔI band, particularly at reduced power. The RAOC methodology and application is fully described in Reference 8. The analysis for Cycle 2 indicates that no change to the safety parameters is required for RAOC operation. During operation at or near steady state equilibrium conditions, core peaking factors are significantly reduced due to the limited amount of xenon skewing possible under these operating conditions. The Cycle 2 Technical Specifications recognize this reduction in core peaking factors through the use of a Base Load Technical Specification.

Adherence to the F_Q limit is obtained by using the F_Q Surveillance Technical Specification, also described in Reference 8. F_Q surveillance replaces the previous F_{xy} surveillance by comparing a measured F_Q, increased to account for expected plant maneuvers, to the F_Q limit. This provides a more convenient form of assuring plant operation below the F_Q limit while retaining the intent of using a measured parameter to verify operation below Technical Specification limits. F_Q surveillance is only a change to the plant's surveillance requirements and as such has no impact on the results of the Cycle 2 analysis or safety parameters.

Table 2 provides a summary of Cycle 2 kinetics characteristics compared with the OFA transition current limits based on previously submitted accident analyses.

Table 3 provides the control rod worths and requirements at the most limiting condition during the cycle (end-of-life) for the standard burnable absorber design. The required shutdown margin is based on previously submitted accident analysis. The available shutdown margin exceeds the minimum required.

The loading pattern contains 64 burnable absorber (BA) rods located in 16 BA rod assemblies. Location of the BA rods are shown in Figure 1.

2.3 THERMAL AND HYDRAULIC DESIGN

The thermal hydraulic methodology, DNBR correlation and core DNB limits used for Cycle 2, are consistent with the OFA transition licensing submittal[2]. The thermal hydraulic safety analyses used for Cycle 2 are based on a reduced design flow rate in comparison to Reference 2. No significant variations in thermal margins will result from the Cycle 2 reload.

The thermal-hydraulic methods used to analyze axial power distributions generated by the RAOC methodology are similar to those used in the Constant Axial Offset Control (CAOC) methodology. Normal operation power distributions are evaluated relative to the assumed limiting normal operation power distribution used in the accident analysis. Limits on allowable operating axial flux imbalance as a function of power level from these considerations were found to be less restrictive than those resulting from LOCA F_Q considerations.

The Condition II analyses were evaluated relative to the axial power distribution assumptions used to generate DNB core limits and resultant Overtemperature Delta-T setpoints (including the $f(\Delta I)$ function). No changes in these limits are required for RAOC operation.

3.0 POWER CAPABILITY AND ACCIDENT EVALUATION

3.1 POWER CAPABILITY

The plant power capability has been evaluated considering the consequences of those incidents examined in the FSAR[3] using the previously accepted design basis. It is concluded that the core reload will not adversely affect the ability to safely operate at the design power level (Section 1.0) during Cycle 2. For the overpower transient, the fuel centerline temperature limit of $4700^{\circ}F$ can be accommodated with margin in the Cycle 2 core. The time dependent densification model[9] was used for fuel temperature evaluations. The LOCA limit at rated power can be met by maintaining $F_Q(z)$ at or below 2.26 x K(z).

3.2 ACCIDENT EVALUATION

The effects of the reload on the design basis and postulated incidents analyzed in the FSAR[3] were examined. In all cases, it was found that the effects were accommodated within the conservatism of the initial assumptions used in the previous applicable safety analysis, the safety analysis performed in support of the OFA transition licensing submittal[2], or reanalysis as described in Section 3.3.

A core reload can typically affect accident analysis input parameters in the following areas: core kinetic characteristics, control rod worths, and core peaking factors. Cycle 2 parameters in each of these three areas were examined as discussed in the following subsections to ascertain whether new accident analyses (in addition to the OFA analyses) were required.

3.2.1 KINETICS PARAMETERS

Table 2 is a summary of the OFA transition kinetics parameters current limits along with the associated Cycle 2 calculated values. All of the kinetics values fall within the bounds of the OFA current limits.

3.2.2 CONTROL ROD WORTHS

Changes in control rod worths may affect differential rod worths, shutdown margin, ejected rod worths, and trip reactivity. Table 2 shows that the maximum differential rod worth of two RCCA control banks moving together in their highest worth region for Cycle 2 meets the OFA transition current limit. As noted in the OFA transition licensing submittal,(2) Table 3 shows that the Cycle 2 shutdown margin requirement has been changed from 1.6%$\Delta\rho$ to 1.3%$\Delta\rho$. The reduced shutdown margin was shown to be acceptable by the results of the OFA transition safety analyses.(2) Table 4 is a summary of OFA transition current limit control rod ejection analysis parameters and the corresponding Cycle 2 values. The ejected rod worths are within the OFA transition limits.

3.2.3 CORE PEAKING FACTORS

Peaking factors for the dropped RCCA incidents were evaluated based on the NRC approved dropped rod methodology described in Reference 10. Results show that DNB design basis is met for all dropped rod events initiated from full power.

The peaking factors for steamline break and control rod ejection have been evaluated and are within the bounds of the limits of the OFA transition licensing submittal(2) analysis.

3.3 REDUCED RCS FLOW

The safety analyses performed in support of the OFA transition licensing submittal(2) assumed a Thermal Design Flow of 386,000 gpm. For Cycle 2, the TDF will be 382,000 gpm. This represents an approximate 1 percent reduction in the RCS flow used for the OFA transition licensing submittal(2).

The following safety evaluation confirms the acceptability of operation at 100 percent of rated thermal power and 99 percent of the RCS flow assumed in the OFA transition analyses. All of the affected FSAR Chapter 15 accidents and protection system setpoints have been reviewed to determine the impact of the proposed reduction in flow requirement. In addition, Technical Specification changes required to support the reduced flow are included in Section 4.0.

3.3.1 DNB CONSIDERATIONS

The core DNB limits have been verified to be unchanged from the OFA transition values, and the conclusion that the DNB basis is met for the following transients remains valid:

- Excessive Heat Removal Due to Feedwater System Malfunction
- Excessive Load Increase
- Main Steamline Depressurization
- Main Steamline Rupture
- Loss of Load/Turbine Trip
- Partial Loss of Forced Reactor Coolant Flow
- Complete Loss of Forced Reactor Coolant Flow
- Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition
- Uncontrolled RCCA Bank Withdrawal at Power
- Startup of an Inactive Reactor Coolant Loop
- Inadvertent ECCS Operation at Power
- Reactor Coolant System Depressurization

3.3.2 NON-DNB CONSIDERATIONS

In addition to the DNB concern, the following evaluations are presented for those accidents which are not DNB related or for which DNBR is not the only safety criterion of interest.

Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition

A control rod assembly withdrawal incident when the reactor is subcritical results in an uncontrolled addition of reactivity leading to a power excursion (Section 15.2.1 of the FSAR). The nuclear power response is characterized by a very fast rise terminated by the reactivity feedback of the negative fuel temperature coefficient. The power excursion caused a heatup of the moderator. However, since the power rise is rapid and is followed by an immediate reactor trip, the moderator temperature rise is small. Thus, nuclear power response is primarily a function of the Doppler temperature coefficient. An increase in temperature due to reduced RCS flow would result in more Doppler feedback, thus reducing the nuclear power excursion as calculated in the OFA transition analysis which partially compensates for the flow reduction.

The OFA transition analysis shows that for a reactivity insertion rate of 75×10^{-5} delta-K/sec, the peak hot spot heat flux achieved is 179.4 percent of nominal with a resultant peak fuel average temperature of 2242°F, and a peak clad temperature of 726°F. A 1 percent reduction of reactor coolant flow would degrade heat transfer from the fuel by a maximum of 1 percent. Thus, peak fuel and clad temperatures would also increase by a maximum 1 percent, yielding maximum fuel and clad temperatures which are still significantly below fuel melt (4800°F) and zirconium-H_2O reaction (1800°F) limits. Therefore, the conclusions presented in the OFA transition licensing submittal[2] are still valid.

Boron Dilution

The results of the boron dilution transient will remain unchanged for all modes of operation due to a reduction in reactor coolant flow. The maximum dilution flow rate, RCS active volumes, and RCS boron concentrations are not impacted by a reduction in flow. Since these parameters determine the amount of time available to the operator to terminate the dilution event, the results presented in the FSAR remain unchanged.

Loss of Load

The loss of load accident is presented in Section 15.2.7 of the FSAR and can result from either loss of external electrical load or a turbine trip. The result of a loss of load is an increase in core power which exceeds the secondary system power extraction, thus causing an increase in core water temperature. A reduction in RCS flow will result in a more rapid pressure rise than that calculated in the OFA transition analysis. The effect will be minor, however, since the reactor is tripped on high pressurizer pressure. Thus, the time to trip will be decreased, which will result in a lower total energy input to the coolant. The analysis shows a peak pressurizer pressure of 2567 psia. A 1 percent reduction in flow will lead to a conservative increase in system pressure to less than 2580 psia. The pressurizer will not fill, and the maximum pressures are within the design limits. Therefore, operation at reduced flow will not violate safety limits following a loss of load accident.

Loss of Normal Feedwater/Station Blackout

This transient is analyzed to demonstrate that the peak RCS pressure does not exceed allowable limits and that the core remains covered with water. These criteria are assured by applying the more stringent requirement that the pressurizer must not be filled with water. The effect of reducing initial core flow results in an initial more rapid heatup of the RCS. The resultant coolant density decrease increases the volume of water in the pressurizer. These transients have been reanalyzed with the reduced flow assumption. In addition, the low-low steam generator level setpoint will be revised and a filter added to the channels to help prevent unnecessary reactor trips as a result of load rejections. These changes have been incorporated into the reanalysis, and appropriate Technical Specification changes are identified in Section 4.0. The results show considerable margin to filling the pressurizer. Therefore, all safety criteria are met for the events.

Steamline Break

The steamline break transient is analyzed at hot zero power, end-of-life conditions for the following cases:

- Inadvertent opening of a steam dump, safety, or relief valve (Section 15.2.13 of the FSAR)

- Main steam pipe rupture with and without offsite power available (Section 15.4.2 of the FSAR)

A steamline break results in a rapid depressurization of the steam generators and primary side cooldown. This causes a large reactivity insertion due to the presence of a negative moderator temperature coefficient. A reduction in reactor coolant flow will result in a reduction in heat transfer from the fuel to the coolant. Therefore, the reactivity insertion and return to power in the double-ended rupture case for reduced flow conditions would be less limiting that the cases presented in the FSAR. For the double-ended rupture case, the time of safety injection actuation is unaffected by reduced coolant flow. This, coupled with a slower return to power would result in a significant reduction in peak average power from the FSAR results. The main steam depressurization case is bounded by the double-ended rupture. Since the return to power is less severe and the DNB evaluations remain valid as previously stated, the conclusions presented in the OFA transition licensing submittal[2] are still valid for a 1 percent reduction in reactor coolant flow.

Rupture of a Main Feedwater Line

This transient is analyzed to demonstrate that the peak RCS pressure does not exceed allowable limits and that the core remains covered with water. These criteria are assured by applying the more stringent requirement that bulk voiding does not occur at the outlet of the core. The effect of reducing initial core flow results in an initial more rapid heatup of the reactor coolant system (RCS). This transient has been reanalyzed with the reduced flow assumption. In addition, the low-low steam generator level setpoint will be revised and a filter added to the channels to help prevent unnecessary reactor trips as a result of load rejections. These changes have been incorporated into the reanalysis, and appropriate Technical Specification changes are identified in Section 4.0. The results show considerable margin to hot leg saturation. Therefore, all safety criteria are met for the event.

Locked Rotor

Following a locked rotor, reactor coolant system temperature rises until shortly after reactor trip. A reduction in RCS flow will not affect the time to DNB since DNB is conservatively assumed to occur at the beginning of the transient. The flow reduction in the affected loop is so rapid that the time of reactor trip on low flow does not change due to the 1 percent reduction in reactor coolant flow. Therefore, the nuclear power and heat flux transients will not change from those presented in the FSAR. However, the reduction in flow will result in slightly higher system pressures and clad temperatures. The peak RCS pressure calculated in the OFA transition analysis was 2593 psia. A 1 percent reduction in reactor coolant flow would cause a conservative increase in pressure to less than 2620 psia, which is still significantly below the pressure at which vessel stress limits are exceeded. The peak clad temperature calculated in the OFA transition analysis is 1964°F, well below the limit of 2700°F, and shows that a slight increase in this parameter due to reduced RCS flow can be easily accommodated. Therefore, the conclusions presented in the OFA transition licensing submittal[2] are still valid.

Control Rod Ejection

The rod ejection transient is analyzed at full power and hot standby for both beginning and end-of-life conditions (Sections 15.4.6 of the FSAR). A reduction in core flow will result in a reduction in heat transfer to the coolant, which will increase peak clad and fuel temperatures and peak fuel stored energy. However, all cases have margin to fuel failure limits. The effect of reducing reactor coolant flow is to increase the peak clad temperatures. The analysis shows that, for the worst case, there is sufficient conservatism in the analysis assumptions and margin in the results such that the peak clad temperature limit (2700°F) is not violate with the reduced flow. This was verified by a reanalysis of the limiting end-of-life zero power case. The peak clad temperature calculated for this case in the OFA transition analysis was 2685°F. The reanalysis of this case assumed the reduced RCS flow, but used shorter time steps to remove some conservatism in the calculation of the nuclear power transient. The result was a peak clad temperature of 2683°F. Thus, the limit is not violated. The fuel temperatures and peak fuel stored energy will also increase slightly due to the 1 percent decrease in reactor coolant flow. However, there is sufficient margin between the analysis results and the limits to accommodate the effects of the reduced flow. Therefore, the conclusions presented in the OFA transition licensing submittal[2] are still valid.

LOCA Analysis

A LOCA analysis has been performed for McGuire Unit 2 that uses the reduced Thermal Design Flow. Results of the analysis are given in Section 3.4.

Technical Specification Changes

The necessary revisions to the Technical Specifications to support operation at the reduced flow are included in Section 4.0. Each Technical Specification change from the OFA transition submittal(2) is discussed below.

2.1 Safety Limits

A new reactor core safety limits curve is provided. As discussed above, the DNB limits of the figure are unchanged. However, the Vessel Exit Boiling limits become more restrictive since flow is reduced for a given power.

2.2 Limiting Safety System Settings

The protection system setpoints have been reviewed for the reduced flow. The only setpoints which are impacted by the flow reduction are the Overtemperature Delta-T and Overpower Delta-T functions. These setpoints are designed to protect the core by tripping the reactor before the core safety limits (Figure 2.1-1) are exceeded. The setpoint equations have been recalculated for the reduced flow with OFA in addition to the introduction of ITDP and steam generator low-low level setpoint changes.

In addition, the time constants in the equations have been updated. Specifically, the lag time constants in the delta-T and Tavg channels have been increased from 2 to 6 seconds, to accommodate operational considerations. The effect of this change has been evaluated by reanalyzing the limiting events that rely on Overtemperature Delta-T and Overpower Delta-T protection.

The limiting RCCA Withdrawal at Power cases from the OFA transition analyses have been reanalyzed with the increased time constants in the Overtemperature Delta-T setpoint equation. The results show that the DNB design basis is met.

The Overpower Delta-T trip is not relied upon for protection for any of the FSAR accident analyses. However, a spectrum of steamline breaks were analyzed at various power levels in Reference 11 to determine the limiting cases that are presented in the FSAR. Some of the small steamline breaks at power analyzed in this generic study rely on Overpower Delta-T for protection.

A McGuire-specific analysis was performed that verifies that the DNB design basis is met for small breaks at full power with the increased time constants in the Overpower Delta-T setpoint equation.

Also, the lead-lag compensation on Tavg is changed from 33/4 to 28/4. The 28/4 compensation was used in the accident analyses and affords the plant more margin to an Overtemperature Delta-T trip on a load rejection.

3/4.2.3 RCS Flow Rate and F-delta-H, and Bases

A new RCS flow vs. R figure is provided for Unit 2 to reflect the reduced flow, introduction of OFA and ITDP, and removal of rod bow parameter, R_2.

3.4 LOCA Analysis

The large break LOCA analysis applicable for transition and full OFA core cycles of McGuire 1 and 2 was performed utilizing the OFA design. This is consistent with the methodology given in Reference 2 for the OFA transition. The currently approved UHI Large Break ECCS Evaluation Model modified to incorporate BART(12) core reflood heat transfer models was utilized for the analysis. BART(13) has been approved for use on non-UHI plants. Four cold leg breaks were reanalyzed.

Evaluation of hydraulic mismatches of less than 10% have shown an insignificant effect on blowdown cooling, such that the impact on reflood cooling alone needs to be considered.

Since the overall resistance of the two types of fuel is essentially identical, only the crossflows during core reflood due to the smaller rod size and different grid designs need be evaluated. The maximum flow reduction due to crossflow calculated to occur in the OFA is ~2.9%. Analyses have been performed which demonstrate that a 5% reduction leads to a maximum PCT increase of 19°F. Therefore, the PCT increase due to crossflow between adjacent OFA and STD assemblies would be approximately 11°F. This effect can be offset in the McGuire 1 and 2 transition cores by considering the favorable UHI quench characteristics of the STD design. Quenching of fuel throughout the core during blowdown is calculated using UHIPOWERREGIONS LOCTA, with computed parameters then being input to UHIWREFLOOD. If the STD design is modeled the quench parameters significantly improve, leading to a faster reflooding of the core than is true for the OFA case. The magnitude of this benefit is several times the 11° penalty identified for transition cycles; because of this benefit no transition core penalty need be applied. Two further reasons why this method is indeed conservative for transition cores are:

1. The increase in core flow are associated with OFA due to the smaller rod diameter has an important impact on flooding rates during reflood. Full OFA core representation decreases core flooding rates, which reduces heat transfer coefficients.

2. The OFA design has a higher volumetric heat generation rate than STD design. The analysis assumes that the OFA has the hottest rod and maximum $F_{\Delta H}$ which maximizes the calculated PCT.

For breaks up to and including the double-ended severance of a reactor coolant pipe, the emergency core cooling system will meet the acceptance criteria as presented in 10 CFR 50.46. That is:

1. The calculated peak fuel element clad temperature is below the requirement of 2200°.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1% of the total amount of Zircaloy in the reactor.

3. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limit of 17% is not exceeded during or after quenching.

4. The core remains amenable to cooling during and after the break.

5. The core temperature is reduced and decay heat is removed for an extended period of time as required by the long-lived radioactivity remaining in the core.

The Large Break LOCA analysis for McGuire 1 and 2 utilizing the currently approved UHI Evaluation Models modified to incorporate BART technology resulted in a PCT of 2157°F at 2.26 F_Q for the $C_D = 0.6$ (perfect mixing) DECLG break. The small impact for transition core cycles is offset by the presence of STD fuel in the core so that margin to 10 CFR 50.46 limits remains in transition cycles.

4.0 TECHNICAL SPECIFICATION CHANGES

To ensure that plant operation is consistent with the design and safety evaluation conclusion statements made in this report and to ensure that these conclusions remain valid, several technical specifications changes will be needed for Cycle 2. These changes are summarized below.

(1) Technical Specification changes outlined in the OFA transition licensing submittal.[2]

(2) Technical Specification changes given in Appendix A.

5.0 REFERENCES

1. Bordelon, F.M., et. al., "Westinghouse Reload Safety Evaluation Methodology", WCAP-9273, March 1978.

2. Duke Power Company Transmittal to NRC, "Safety Evaluation for McGuire Units 1 and 2 Transition to Westinghouse 17x17 Optimized Fuel Assemblies."

3. "McGuire Final Safety Analysis Report."

4. Miller, J.V., (Ed.), "Improved Analytical Model used in Westinghouse Fuel Rod Design Computations", WCAP-8785, October 1976.

5. George, R.A., (et. al.), "Revised Clad Flattening Model", WCAP-8381, July 1974.

6. Risher, D. H., (et. al.), "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis," WCAP-8964, June 1977.

7. Skaritka, J., Iorii, J.A., "Operational Experience with Westinghouse Cores", WCAP-8183, Revision 13, September, 1984.

8. Miller, R. W., (et al.), "Relaxation of Constant Axial Offset Control-F_Q Surveillance Technical Specification," WCAP-10217-A, June 1983.

9. Hellman, J.M. (Ed.), "Fuel Densification Experimental Results and Model for Reactor Operation", WCAP-8219-A, March 1975.

10. Letter from NRC, C. O. Thomas to E. P. Rahe, Jr., Westinghouse, "Acceptance for Referencing of Licensing Topical Report WCAP-10297-(P), WCAP-10298 (NS-EPR-2545) Entitled Dropped Rod Methodology for Negative Flux Rate Trip Plants", March 31, 1983.

11. Hollingsworth, S. D. and Wood, D. C., "Reactor Core Response to Excessive Secondary Steam Releases," WCAP-9226, Revision 1, (Proprietary), January, 1978, and WCAP-9227, Revision 1, (Non-Proprietary), January, 1978.

12. Schwartz, W. R., "Addendum to BART-A1: A Computer Code for the Best Estimate Analysis of Reflood Transients," WCAP-9561, Addendum 1, November 1984. (Westinghouse Proprietary)

13. Young, M. Y., et al., "BART-A1: A Computer Code for the Best Estimate Analysis of Reflood Transients," WCAP-9561-P-A, March 1984. (Westinghouse Proprietary)

TABLE 1

MCGUIRE UNIT 2 - CYCLE 2

FUEL ASSEMBLY DESIGN PARAMETERS

Region	1	2	3	4*
Enrichment (w/o U-235)+	2.093	2.566	3.089	3.20
Density(% Theoretical)+	94.77	94.41	94.87	95.0
Number of Assemblies	5	64	64	60
Approximate Burnup at++ Beginning of Cycle 2 (MWD/MTU)	14698	16750	11640	0
Approximate Burnup at++ End of Cycle 2 (MWD/MTU)	23793	26635	23250	10692

* Optimized Fuel - Zirc grid

+ All fuel region values are as-built except Region 4 values which are nominal.

++Based on EOC1 = 14900 MWD/MTU, EOC2 = 10700 MWD/MTU (coastdown included)

TABLE 2

MCGUIRE UNIT 2 - CYCLE 2
KINETICS CHARACTERISTICS

	OFA Transition Current Limits(2)	Cycle 2 Design
Minimum Moderator Temperature Coefficient (pcm/°F)*	+5 < 70% of RTP 0 ≥ 70% of RTP	+5 <70% of RTP 0 ≥70% of RTP
Doppler Temperature Coefficient (pcm/°F)*	-2.9 to -0.91	-2.9 to -0.91
Least Negative Doppler-Only Power Coefficient, Zero to Full Power, (pcm/% power)*	-9.55 to -6.05	-9.55 to -6.05
Most Negative Doppler Only Power Coefficient, Zero to Full Power (pcm/% power)*	-19.4 to -12.6	-19.4 to -12.6
Minimum Delayed Neutron Fraction β_{eff}, (%)	.44	>.44
Minimum Delayed Neutron Fraction β_{eff}, (%) [Ejected Rod at BOL]	.50	>.50
Maximum Differential Rod Worth of Two Banks Moving Together (pcm/in)*	100	<100

*pcm = 10^{-5} $\Delta\rho$

TABLE 3

END-OF-CYCLE SHUTDOWN REQUIREMENTS AND MARGINS
MCGUIRE UNIT 2 - CYCLE 2

Control Rod Worth (%$\Delta\rho$)	Cycle 1	Cycle 2
All Rods Inserted	7.86	7.11
All Rods Inserted Less Worst Stuck Rod	6.73	6.07
(1) Less 10%	6.06	5.46
Control Rod Requirements		
Reactivity Defects (Doppler, T_{avg}, Void, Redistribution)	3.1	2.98
Rod Insertion Allowance	0.50	0.50
(2) Total Requirements	3.60	3.48
Shutdown Margin [(1) - (2)] (%$\Delta\rho$)	2.46	1.98
Required Shutdown Margin (%$\Delta\rho$)	1.60	1.30

TABLE 4

MCGUIRE UNIT 2 - CYCLE 2
CONTROL ROD EJECTION ACCIDENT PARAMETERS

HZP-BOC	OFA Transition Current Limit	Cycle 2
Maximum ejected rod worth, %$\Delta\rho$	0.75	<0.75
Maximum F_Q (ejected)	11.5	<11.5
HFP-BOC		
Maximum ejected rod worth, %$\Delta\rho$	0.23	<0.23
Maximum F_Q (ejected)	5.3	<5.3
HZP-EOC		
Maximum ejected rod worth, %$\Delta\rho$	0.90	<0.90
Maximum F_Q (ejected)	20.0	<20.0
HFP-EOC		
Maximum ejected rod worth, %$\Delta\rho$	0.23	<0.23
Maximum F_Q (ejected)	5.9	<5.9

180°

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
					4	4	4	4	4	4	4					1
			4	4	4 4	2	4 4	1	4 4	2	4 4	4	4			2
		4	3	3	2	2	2	2 SS	2	2	2	3	3	4		3
		4	3	3	3	3	2	2	2	3	3	3	3	4		4
	4	4 4	2	3	3	2	3	2	3	2	3	3	2	4 4	4	5
	4	2	2	3	2	3	3	3	3	3	2	3	2	2	4	6
	4	4 4	2	2	3	3	3	2	3	3	3	2	2	4 4	4	7
90°	4	1	2	2	2	3	2	1	2	3	2	2	2	1	4	8 270°
	4	4 4	2	2	3	3	3	2	3	3	3	2	2	4 4	4	9
	4	2	2	3	2	3	3	3	3	3	2	3	2	2	4	10
	4	4 4	2	3	3	2	3	2	3	2	3	3	2	4 4	4	11
		4	3	3	3	3	2	2	2	3	3	3	3	4		12
		4	3	3	2	2	2	2 SS	2	2	2	3	3	4		13
			4	4	4 4	2	4 4	1	4 4	2	4 4	4	4			14
					4	4	4	4	4	4	4					15

0°

X Y	region number BA'S

SS Secondary Source

Figure 1

CORE LOADING PATTERN

MCGUIRE UNIT 2, CYCLE 2

APPENDIX A

TECHNICAL SPECIFICATION

PAGE CHANGES

(In addition to proposed changes submitted in support of the OFA transition licensing submittal[2])

Delete Pages 3/4 2-2
3/4 2-3
3/4 2-5 (Figure 3.2-1B)
3/4 2-10
3/4 2-11

(Reference to Amendment 32 (Unit 1), Amendment 13 (Unit 2))

MODIFICATIONS TO 3/4.2.1

AXIAL FLUX DIFFERENCE LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE AFD ~~(UNIT 1)~~

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within:

a. the allowed operational space defined by Figure 3.2-1 for RAOC operation, or

b. within a ± ~~X~~ 5 percent target band about the target flux difference during base load operation.

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER*.

ACTION:

a. For RAOC operation with the indicated AFD outside of the Figure 3.2-1 limits,

   1. Either restore the indicated AFD to within the Figure 3.2-1 limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux - High Trip setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. For Base Load operation above APL^{ND}** with the indicated AXIAL FLUX DIFFERENCE outside of the applicable target band about the target flux difference:

   1. Either restore the indicated AFD to within the target band limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than APL^{ND} of RATED THERMAL POWER and discontinue Base Load operation within 30 minutes.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the Figure 3.2-1 limits.

*See Special Test Exception 3.10.2.

**APL^{ND} is the minimum allowable power level for base load operation and will be provided in the Peaking Factor Limit Report per Specification 6.9.1.9.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2. At least once per hour for the first 24 hours after restoring the AFD Monitoring Alarm to OPERABLE status.

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its limits when at least two OPERABLE excore channels are indicating the AFD to be outside the limits.

4.2.1.3 When in Base Load operation, the target axial flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 When in Base Load operation, the target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference ~~pursuant to 4.2.1.3~~ above or by linear interpolation between the most recently measured value and ~~0 percent~~ at the end of cycle life. The provisions of Specification 4.0.4 are not applicable.

the calculated value

in conjuction with the surveillance requirements of Specification 3/4.2.2

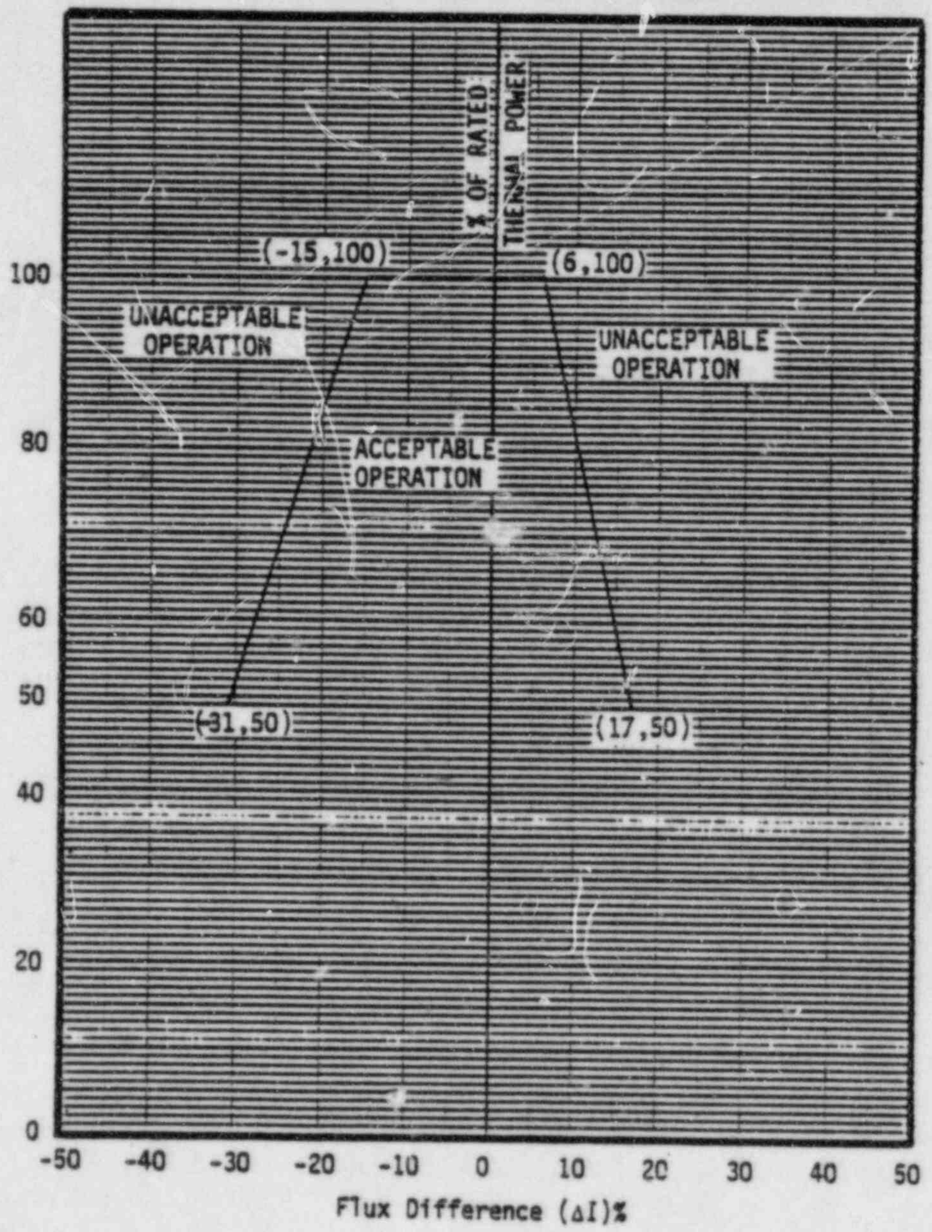


FIGURE 3.2-1

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

MODIFICATIONS TO 3/4.2.2

HEAT FLUX HOT CHANNEL FACTOR LIMITS

POWER DISTRIBUTION LIMITS

HEAT FLUX HOT CHANNEL FACTOR-FQ(Z)

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(z)$ shall be limited by the following relationships:

$$F_Q(z) \leq \left[\frac{\overset{2.26}{\cancel{2.15}}}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(z) \leq \left[\frac{\overset{2.26}{\cancel{2.15}}}{0.5}\right] [K(Z)] \text{ for } P \leq 0.5$$

where $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

and $K(z)$ is the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1

ACTION:

With $F_Q(z)$ exceeding its limit:

1. Reduce THERMAL POWER at least 1 percent for each 1 percent $F_Q(z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints (value of K_4) have been reduced at least 1 percent (in ΔT span) for each 1 percent $F_Q(z)$ exceeds the limit.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; THERMAL POWER may then be increased provided $F_Q(z)$ is demonstrated through incore mapping to be within its limit.

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 For RAOC operation, $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. *Verify the requirements of Specification 3.2.2 are satisfied.*

c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{\overset{2.26}{\cancel{2.15}} \times K(z)}{P \times W(z)} \text{ for } P > 0.5$$

$$F_Q^M(z) \leq \frac{\overset{2.26}{\cancel{2.15}} \times K(z)}{W(z) \times 0.5} \text{ for } P \leq 0.5$$

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, $\overset{2.26}{\cancel{2.15}}$ is the F_Q limit, K(z) is given in Figure 3.2-2, P is the relative THERMAL POWER, and W(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. This function is given in the Peaking Factor Limit Report as per Specification 6.9.1.9.

d. Measuring $F_Q^M(z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(z)$ was last determined,* or

2. At least once per 31 Effective Full Power Days, whichever occurs first.

*During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

e. With measurements indicating

$$\begin{array}{l}\text{maximum}\\ \text{over } z\end{array} \left(\frac{F_Q^M(z)}{K(z)}\right)$$

has increased since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

1) $F_Q^M(z)$ shall be increased by 2% over that specified in Specification 4.2.2.2c. or

2) $F_Q^M(z)$ shall be measured at least once per 7 Effective Full Power Days until two successive maps indicate that

$$\begin{array}{l}\text{maximum}\\ \text{over } z\end{array} \left(\frac{F_Q^M(z)}{K(z)}\right) \text{ is not increasing.}$$

f. With the relationships specified in Specification 4.2.2.2c. above not being satisfied:

1) Calculate the percent $F_Q(z)$ exceeds its limit by the following expression:

$$\left\{\left(\begin{array}{l}\text{maximum}\\ \text{over } z\end{array} \; 2.26\left[\frac{F_Q^M(z) \times W(z)}{\frac{\cancel{2.15}}{P} \times K(z)}\right]\right)-1\right\} \times 100 \quad \text{for } P \geq 0.5$$

$$\left\{\left(\begin{array}{l}\text{maximum}\\ \text{over } z\end{array} \; 2.26\left[\frac{F_Q^M(z) \times W(z)}{\frac{\cancel{2.15}}{0.5} \times K(z)}\right]\right)-1\right\} \times 100 \quad \text{for } P < 0.5$$

2) One of the following actions shall be taken:

a) Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the AFD limits of 3.2-1 by 1% AFD for each percent $F_Q(z)$ exceeds its limits as determined in Specification 4.2.2.2f.1). Within 8 hours, reset the AFD alarm setpoints to these modified limits, or

b) Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated above, or

c) Verify that the requirements of Specification 4.2.2.3 for Base Load operation are satisfied and enter Base Load operation.

g. The limits specified in Specifications 4.2.2.2c, 4.2.2.2e., and 4.2.2.2f. above are not applicable in the following core plane regions:

1. Lower core region from 0 to 15%, inclusive.

2. Upper core region from 85 to 100%, inclusive.

4.2.2.3 Base Load operation is permitted at powers above APL^{ND} if the following conditions are satisfied:

a. Prior to entering Base Load operation, maintain THERMAL POWER above APL^{ND} and less than or equal to that allowed by Specification 4.2.2.2 for at least the previous 24 hours. Maintain Base Load operation surveillance (AFD within ±5% of target flux difference) during this time period. Base Load operation is then permitted providing THERMAL POWER is maintained between APL^{ND} and APL^{BL} or between APL^{ND} and 100% (whichever is most limiting) and F_Q surveillance is maintained pursuant to Specification 4.2.2.4. APL^{BL} is defined as:

$$APL^{BL} = \underset{\text{over } Z}{\text{minimum}} \left[\frac{(\overset{2.26}{\cancel{2.15}} \times K(Z)}{F_Q^M(Z) \times W(Z)_{BL}} \right] \times 100\%$$

where: $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty. The F_Q limit is $\overset{2.26}{\cancel{2.15}}$. K(z) is given in Figure 3.2-2. $W(z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during base load operation. The function is given in the Peaking Factor Limit Report as per Specification 6.9.1.9.

b. During Base Load operation, if the THERMAL POWER is decreased below APL^{ND} then the conditions of 4.2.2.3.a shall be satisfied before re-entering Base Load operation.

4.2.2.4 During Base Load Operation $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER above APL^{ND}.

b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. *Verify the requirements of Specification 3.2.2 are satisfied.*

c. Satisfying the following relationship:

$$F_Q^M(Z) \le \frac{\overset{2.26}{\cancel{2.15}} \times K(Z)}{P \times W(Z)_{BL}} \text{ for } P > APL^{ND}$$

where: $F_Q^M(Z)$ is the measured $F_Q(Z)$. The F_Q limit is ~~2.15~~ 2.26.

$K(Z)$ is given in Figure 3.2-2. P is the relative THERMAL POWER. $W(Z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during normal operation. This function is given in the Peaking Factor Limit Report as per Specification 6.9.1.9.

d. Measuring $F_Q^M(Z)$ in conjunction with target flux difference determination according to the following schedule:

1. Prior to entering BASE LOAD operation after satisfying Section 4.2.2.3 unless a full core flux map has been taken in the previous 31 EFPD with the relative thermal power having been maintained above APL^{ND} for the 24 hours prior to mapping, and

2. At least once per 31 effective full power days.

e. With measurements indicating

$$\underset{\text{over } Z}{\text{maximum}} \left[\frac{F_Q^M(Z)}{K(Z)} \right]$$

has increased since the previous determination $F_Q^M(Z)$ either of the following actions shall be taken:

1. $F_Q^M(Z)$ shall be increased by 2 percent over that specified in 4.2.2.4.c, or

2. $F_Q^M(Z)$ shall be measured at least once per 7 EFPD until 2 successive maps indicate that

$$\underset{\text{over } Z}{\text{maximum}} \left[\frac{F_Q^M(Z)}{K(Z)} \right] \text{ is not increasing.}$$

f. With the relationship specified in 4.2.2.4.c above not being satisfied, either of the following actions shall be taken:

1. Place the core in an equilbrium condition where the limit in 4.2.2.2.c is satisfied, and remeasure $F_Q^M(Z)$, or

2. Comply with the requirements of Specification 3.2.2 for $F_Q(Z)$ exceeding its limit by the percent calculated with one of the following expressions:

$$[(\text{max. over z of } [\frac{F_Q^M(Z) \times W(Z)_{BL}}{\frac{2.26}{P} \times K(Z)}]) - 1] \times 100 \text{ for } P \geq APL^{ND}$$

$$\cancel{[(\text{max. over z of } [\frac{F_Q^M(Z) \times W(Z)}{\frac{2.15}{P} \times K(Z)}]) - 1] \times 100 \text{ for } 0.5 \leq P < APL^{ND}}$$

g. The limits specified in 4.2.2.4.c, 4.2.2.4.e, and 4.2.2.4.f above are not applicable in the following core plan regions:

1. Lower core region 0 to 15 percent, inclusive.

2. Upper core region 85 to 100 percent, inclusive.

4.2.2.5 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of specification 4.2.2.2 an overall measured $F_Q(z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

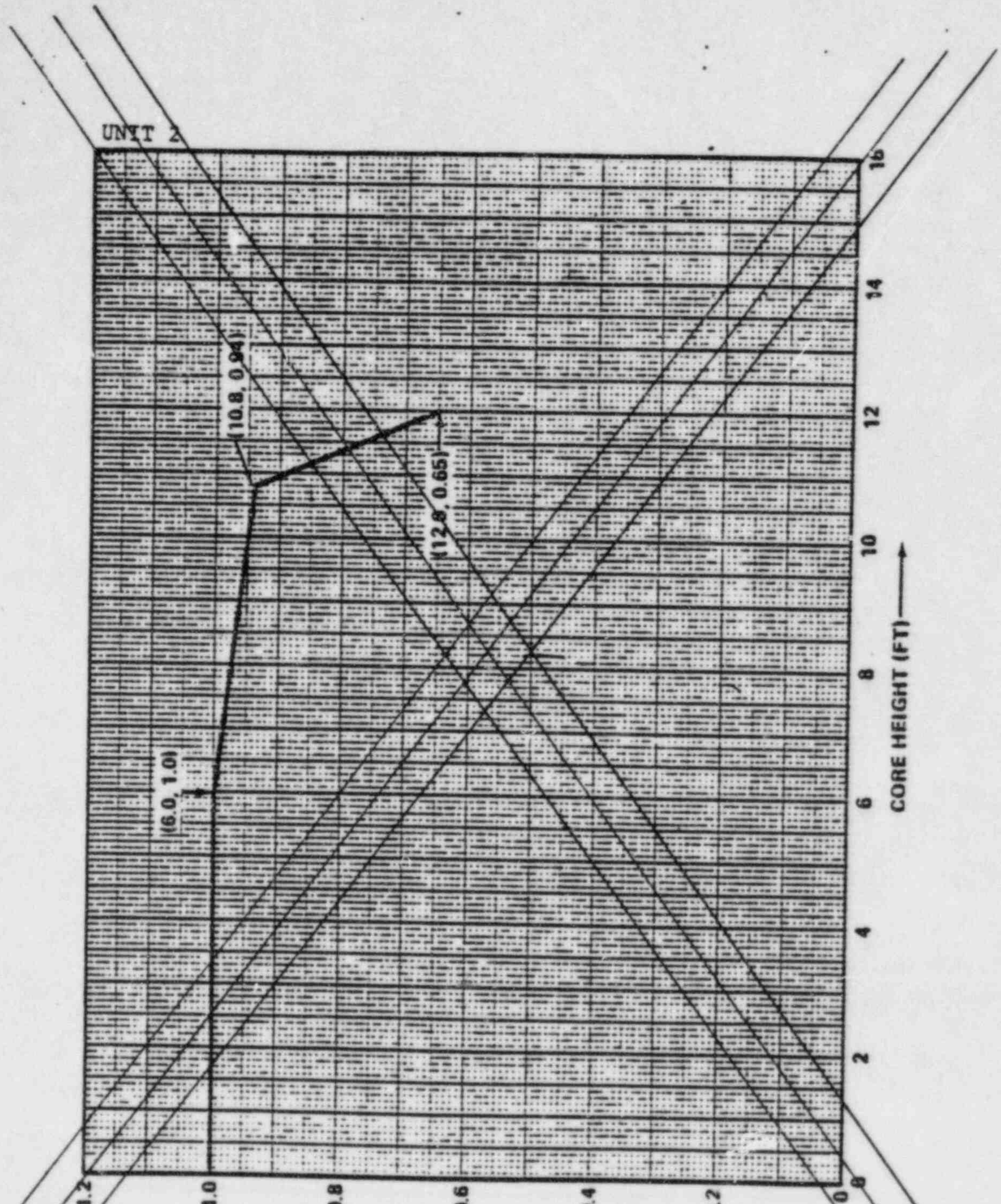
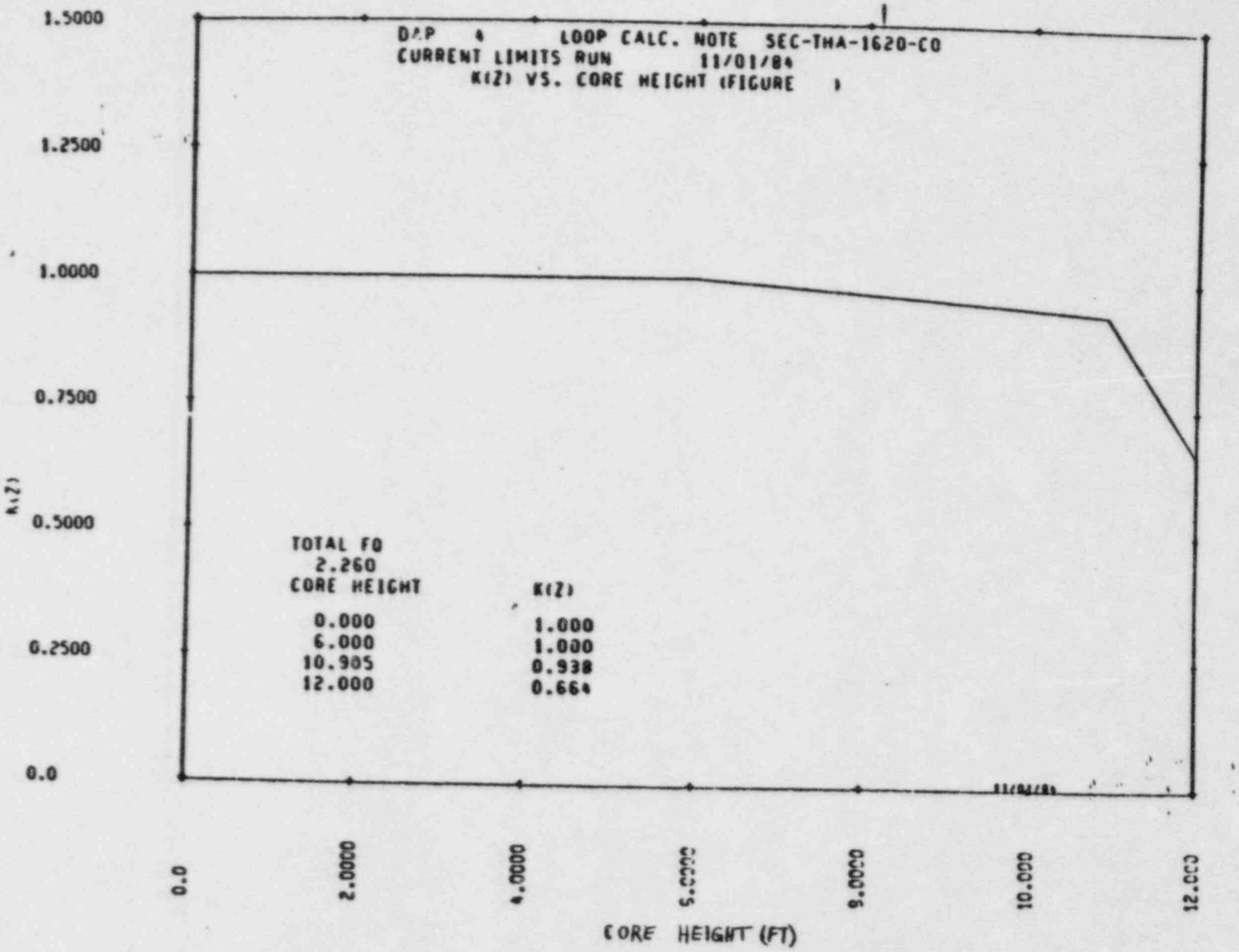


FIGURE 3.2-2b
K(Z) - NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT (UNIT 2)

Replace with
following



D.P 4 LOOP CALC. NOTE SEC-THA-1620-CO
CURRENT LIMITS RUN 11/01/84
K(Z) VS. CORE HEIGHT (FIGURE)
TOTAL FQ
2.260
CORE HEIGHT K(Z)
0.000 1.000
6.000 1.000
10.905 0.938
12.000 0.664
1.5000
1.2500
1.0000
0.7500
0.5000
0.2500
0.0
K(Z)
0.0
2.0000
4.0000
6.0000
8.0000
10.000
12.000
CORE HEIGHT (FT)

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the calculated DNBR in the core at or above the design limit during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding 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.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ 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;

$F^N_{\Delta H}$ 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; and

~~Unit No. 2~~

~~$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.~~

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of 2.26 ~~2.32 (Unit 2), 2.15 (Unit 1)~~ times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full-length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

BASES

AXIAL FLUX DIFFERENCE (Continued)

Although it is intended that the plant will be operated with the AFD within the target band required by Specification 3.2.1 about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

~~Unit No. 2~~

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band.

~~For Unit 1, at~~ At power levels below APL^{ND}, the limits on AFD are defined by Figures 3.2-1, i.e. that defined by the RAOC operating procedure and limits. These limits were calculated in a manner such that expected operational transients, e.g. load follow operations, would not result in the AFD deviating outside of those limits. However, in the event such a deviation occurs, the short period of time allowed outside of the limits at reduced power levels will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prevent operation in the vicinity of the APL^{ND} power level.

~~Unit No. 1~~

At power levels greater than APL^{ND}, two modes of operation are permissible; 1) RAOC, the AFD limit of which are defined by Figure 3.2-1, and 2) Base Load operation, which is defined as the maintenance of the AFD within a ±~~3~~ 5% band about a target value. The RAOC operating procedure above APL^{ND} is the same as that defined for operation below APL^{ND}. However, it is possible when following extended load following maneuvers that the AFD limits may result in restrictions in the maximum allowed power or AFD in order to guarantee operation with $F_Q(z)$ less than its limiting value. To allow operation at the maximum permissible value, the Base Load operating procedure restricts the

AXIAL FLUX DIFFERENCE (Continued)

indicated AFD to relatively small target band and power swings (AFD target band of ±5%, APL^{ND} ≤ power ≤ APL^{BL} or 100% Rated Thermal Power, whichever is lower). For Base Load operation, it is expected that the plant will operate within the target band. Operation outside of the target band for the short time period allowed will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prohibit continued operation in the power region defined above. To assure there is no residual xenon redistribution impact from past operation on the Base Load operation, a 24 hour waiting period at a power level above APL^{ND} and allowed by RAOC is necessary. During this time period load changes and rod motion are restricted to that allowed by the Base Load procedure. After the waiting period extended Base Load operation is permissible.

~~For Unit 1,~~ The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are: 1) outside the allowed ΔI power operating space (for RAOC operation), or 2) outside the allowed ΔI target band (for Base Load operation). These alarms are active when power is greater than: 1) 50% of RATED THERMAL POWER (for RAOC operation), or 2) APL^{ND} (for Base Load operation). Penalty deviation minutes for Base Load operation are not accumulated based on the short period of time during which operation outside of the target band is allowed.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flow rate, and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded, and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

BASES

HEAT FLUX HOT CHANNEL FACTOR and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

The hot channel factor $F_Q^M(z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to either RAOC or Base Load operation, $W(z)$ or $W(z)_{BL}$, to provide assurance that the limit on the hot channel factor, $F_Q(z)$, is met. $W(z)$ accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. $W(z)_{BL}$ accounts for the more restrictive operating limits allowed by Base Load operation which result in less severe transient values. The $W(z)$ function for normal operation is provided in the Peaking Factor Limit Report per Specification 6.9.1.9.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3% from RATED THERMAL POWER for each percent of tilt in excess of 1.0.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a design limit DNBR throughout each analyzed transient.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

THIS FIGURE DELETED

Figure B 3/4 2-1 TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

ADDITIONS TO 6.0

ADMINISTRATIVE CONTROLS

RADIAL PEAKING FACTOR LIMIT REPORT

~~6.9.1.9 The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided to the Regional Administrator of the NRC Regional Office, with a copy to the Director, Nuclear Reactor Regulation, Attention: Chief, Core Performance Branch, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555 for all core planes containing Bank "D" control rods and all unrodded core planes at least 60 days prior to cycle initial criticality. In the event that the limit would be submitted at some other time during core life, it shall be submitted 60 days prior to the date the limit would become effective unless otherwise exempted by the Commission.~~

~~Any information needed to support F_{xy}^{RTP} will be by request from the NRC and need not be included in this report.~~

~~Units No. 2~~

The W(z) functions for RAOC and Base Load operation and the value for APL^{ND} (as required) shall be provided to the Director, Nuclear Reactor Regulations, Attention: Chief, Core Performance Branch, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 at least 60 days prior to cycle initial criticality. In the event that these values would be submitted at some other time during core life, it will be submitted 60 days prior to the date the values would become effective unless otherwise exempted by the Commission.

Any information needed to support W(z), $W(z)_{BL}$ and APL^{ND} will be by request from the NRC and need not be included in this report.

~~Units No. 1~~

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report.

FURTHER MODIFICATIONS
DUE TO :

Reduced RCS Flow

Shutdown Margin

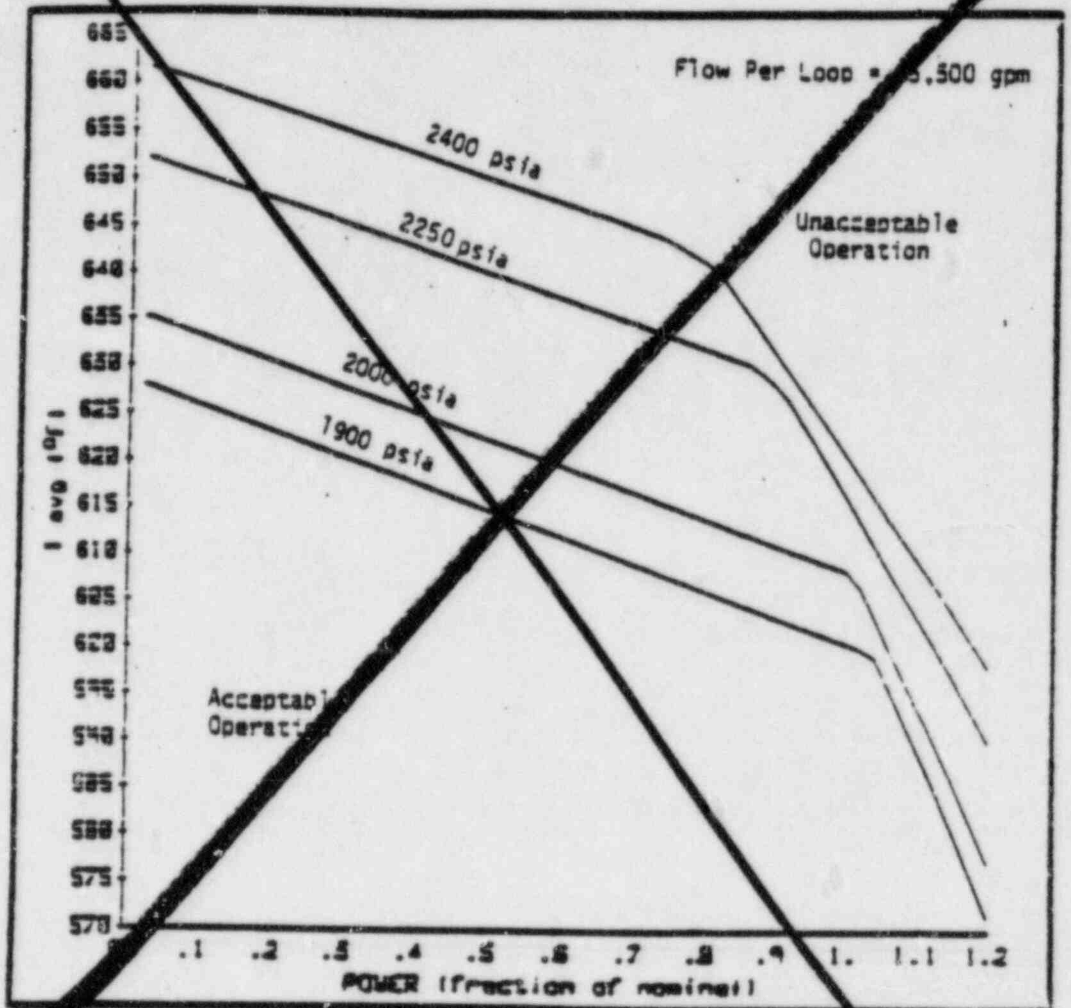
Positive MTC

New Rod Bow Methodology

Increased τ in OTΔT and OPΔT Equations

Improved Thermal Design Procedure

Revised SG Low-Low Level Setpoint

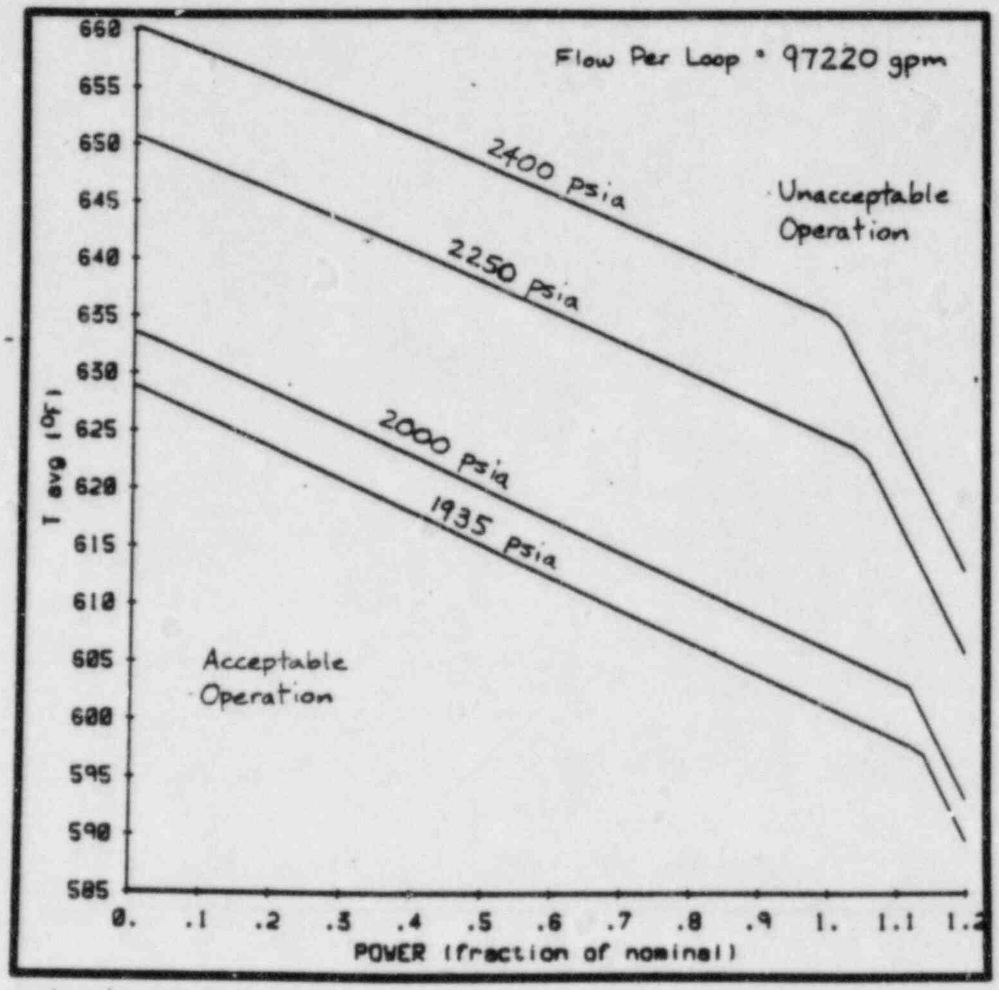


Replace with following

FIGURE 2.1-1b

UNIT 2

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION



REACTOR CORE SAFETY LIMITS

FOUR LOOPS IN OPERATION

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	N.A.	N.A.
2. Power Range, Neutron Flux	Low Setpoint - ≤ 25% of RATED THERMAL POWER	Low Setpoint - ≤ 26% of RATED THERMAL POWER
	High Setpoint - ≤ 109% of RATED THERMAL POWER	High Setpoint - ≤ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds
5. Intermediate Range, Neutron Flux	≤ 25% of RATED THERMAL POWER	≤ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	≤ 10^5 counts per second	≤ 1.3×10^5 counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 3
9. Pressurizer Pressure--Low	≥ 1945 psig	≥ 1935 psig
10. Pressurizer Pressure--High	≤ 2385 psig	≤ 2395 psig
11. Pressurizer Water Level--High	≤ 92% of instrument span	≤ 93% of instrument span
12. Low Reactor Coolant Flow	≥ 90% of design flow per loop*	≥ 89% of design flow per loop*

*Design flow is ~~98,400 gpm per loop for Unit 1 and 95,600~~ 97,220 gpm per loop for Unit 2.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level--Low-Low	≥ 12% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ ~~54.9%~~ 40.0% of span at 100% of RATED THERMAL POWER.	≥ 11% of span from 0 to 30% of RATED THERMAL POWER, increasing to ~~53.9%~~ 39.0% of span at 100% of RATED THERMAL POWER.
14. Undervoltage-Reactor Coolant Pumps	≥ 5082 volts-each bus	≥ 5016 volts-each bus
15. Underfrequency-Reactor Coolant Pumps	≥ 56.4 Hz - each bus	≥ 55.9 Hz - each bus
16. Turbine Trip		
a. Low Trip System Pressure	≥ 45 psig	≥ 42 psig
b. Turbine Stop Valve Closure	≥ 1% open	≥ 1% open
17. Safety Injection Input from ESF	N.A.	N.A.
18. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6, Enable Block Source Range Reactor Trip	≥ 1×10^{-10} amps	≥ 6×10^{-11} amps
b. Low Power Reactor Trips Block, P-7		
1) P-10 Input	10% of RATED THERMAL POWER	≥ 9%, ≤ 11% of RATED THERMAL POWER
2) P-13 Input	≤ 10% RTP Turbine Impulse Pressure Equivalent	≤ 11% RTP Turbine Impulse Pressure Equivalent

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \left(\frac{1 + \tau_1 S}{1 + \tau_2 S}\right) \left(\frac{1}{1 + \tau_3 S}\right) \le \Delta T_o \left\{K_1 - K_2 \left(\frac{1 + \tau_4 S}{1 + \tau_5 S}\right)\left[T\left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where: ΔT = Measured ΔT by RTD Manifold Instrumentation,

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT,

τ_1, τ_2 = Time constants utilized in the lead-lag controller for ΔT, $\tau_1 \ge 8$ sec., $\tau_2 \le 3$ sec.,

$\frac{1}{1 + \tau_3}$ = Lag compensator on measured ΔT,

τ_3 = Time constants utilized in the lag compensator for ΔT, $\tau_3 \le$ ~~1~~ 6 sec.,

ΔT_o = Indicated ΔT at RATED THERMAL POWER,

K_1 $\le$ ~~1.0962 (Unit 2), 1.4060 (Unit 1),~~ 1.200

K_2 = ~~0.0133 (Unit 2)~~, 0.0222 ~~(Unit 1),~~ ✓

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation,

τ_4, τ_5 = Time constants utilized in the lead-lag controller for T_{avg}, ~~τ_4 = 28 sec (Unit 1)~~, $\tau_4 \ge$ ~~33~~ 28 sec. ~~(Unit 2)~~, $\tau_5 \le 4$ sec.,

T = Average temperature, °F,

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg},

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 1: (Continued)

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 \leq$ ~~2~~ 6 sec ~~(Units 1 & 2)~~,

T' $\leq$ 588.2°F Reference T_{avg} at RATED THERMAL POWER,

K_3 = ~~0.000647 (Unit 2)~~, 0.001095 ~~(Unit 1)~~,

P = Pressurizer pressure, psig,

P' = 2235 psig (Nominal RCS operating pressure),

S = Laplace transform operator, sec^{-1},

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between ~~-36%~~ -29% and ~~+8.0% (Unit 2), -41% and -4.0% (Unit 1)~~ +9.0%; $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds ~~-36% (Unit 2), -41% (Unit 1)~~ -29%, the ΔT Trip Setpoint shall be automatically reduced by ~~1.173% (Unit 2)~~, 3.151% ~~(Unit 1)~~ of its value at RATED THERMAL POWER; and

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds ~~+8.0% (Unit 2), -4.0% (Unit 1)~~ +9.0%, the ΔT Trip Setpoint shall be automatically reduced by ~~0.901% (Unit 2), 1.447% (Unit 1)~~ 1.50% of its value at RATED THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 2: OVERPOWER ΔT

$$\Delta T \left(\frac{1 + \tau_1 S}{1 + \tau_2 S}\right) \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left[T\left(\frac{1}{1 + \tau_6 S}\right) - T''\right] - f_2(\Delta I)\right\}$$

Where:

ΔT = As defined in Note 1,

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = As defined in Note 1

τ_1, τ_2 = As defined in Note 1

$\frac{1}{1 + \tau_3 S}$ = As defined in Note 1,

ΔT_0 = As defined in Note 1,

K_4 $\leq$ 1.090~~8~~ ~~(Unit 2), 1.0788 (Unit 1)~~,

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature,

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag controller for T_{avg} dynamic compensation,

τ_7 = Time constant utilized in the rate-lag controller for T_{avg}, $\tau_7 \geq 5$ sec ~~(Units 1 & 2)~~,

$\frac{1}{1 + \tau_6 S}$ = As defined in Note 1,

τ_6 = As defined in Note 1,

K_6 = ~~0.00126/°F (Unit 2)~~, 0.00169/°F ~~(Unit 1)~~ for T > T" and $K_6 = 0$ for T ≤ T".

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

T = As defined in Note 1,

T" = $\leq$ 588.~~1~~2°F Reference T_{avg} at RATED THERMAL POWER,

S = As defined in Note 1, and

$f_2(\Delta I)$ = 0 for all ΔI.

Note 3: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2%.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and reactor coolant temperature and pressure have been related to DNB through the WRB-1 correlation. The WRB-1 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

~~The minimum value of the DNBR during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30 (based upon W-3 correlation). This value corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.~~

~~The curves of Figures 2.1-1 and 2.1-2 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.~~

~~These curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:~~

$$\cancel{F^N_{\Delta H} = 1.55 [1 + 0.2 (1-P)]}$$

~~Where P is the fraction of RATED THERMAL POWER.~~

~~These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the Setpoints to provide protection consistent with core Safety Limits.~~

~~For Unit 1~~, The DNB design basis is as follows: there must be at least a 95% probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 correlation in this application). The correlation DNBR set such that there is a 95% probability with 95% confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95% confidence that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure, and average temperature below which the calculated DNBR is no less than the design DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

The curves are based on a nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$ of 1.49 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.49\ [1 + 0.3\ (1-P)]$$

Where P is the fraction of RATED THERMAL POWER.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

Power Range, Neutron Flux (Continued)

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power, a rod drop accident of a single or multiple rods could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBR's will be greater than ~~[illegible]~~ the design limit DNBR value.

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^{5} counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} >200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to ~~1.6% delta k/k (Unit 2)~~, 1.3% delta k/k ~~(Unit 1)~~ for four loop operation.

APPLICABILITY: MODES 1, 2*, 3, and 4.

ACTION:

With the SHUTDOWN MARGIN less than ~~1.6% delta k/k (Unit 2)~~, 1.3% delta k/k ~~(Unit 1)~~, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to ~~1.6% delta k/k (Unit 2)~~, 1.3% delta k/k ~~(Unit 1)~~:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0 at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1e., below, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

*See Special Test Exception 3.10.1.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. ~~For Unit 1~~, L ~~l~~ess positive than the limits shown in Figure 3.1-0, and
~~b. For Unit 2, less positive than 0 delta k/k/°F for the all rods withdrawn, beginning of cycle life (BOL), hot zero THERMAL POWER condition~~; and

b. ~~c~~. ~~For Units 1 and 2~~, L ~~l~~ess negative than -4.1 x 10-⁴ delta k/k/°F for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition

APPLICABILITY: Specification~~s~~ 3.1.1.3a. ~~and 3.1.1.3b~~. - MODES 1 and 2* only.#
Specification 3.1.1.3~~c~~b - MODES 1, 2, and 3 only.#

ACTION:

a. With the MTC more positive than the limit of Specification~~s~~ 3.1.1.3a. ~~or 3.1.1.3b~~, above, operation in MODES 1 and 2 may proceed provided:

1. ~~For Unit 1~~, C ~~c~~ontrol rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the limits shown in Figure 3.1-0 within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

~~2. For Unit 2, control rod withdrawal limits are established an maintained sufficient to restore the MTC to less positive than 0 delta k/k/°F within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;~~

2 ~~3~~. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

3 ~~4~~. A Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of Specification 3.1.1.3~~c~~b above, be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.0.

#See Special Test Exception 3.10.3.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit of Specification~~s~~ 3.1.1.3a. ~~and 3.1.1.3b.~~, above, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading; and

b. The MTC shall be measured at any THERMAL POWER and compared to -3.2×10^{-4} delta k/k/°F (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than -3.2×10^{-4} delta k/k/°F, the MTC shall be remeasured, and compared to the EOL MTC limit of Specification 3.1.1.3~~c~~b, at least once per 14 EFPD during the remainder of the fuel cycle.

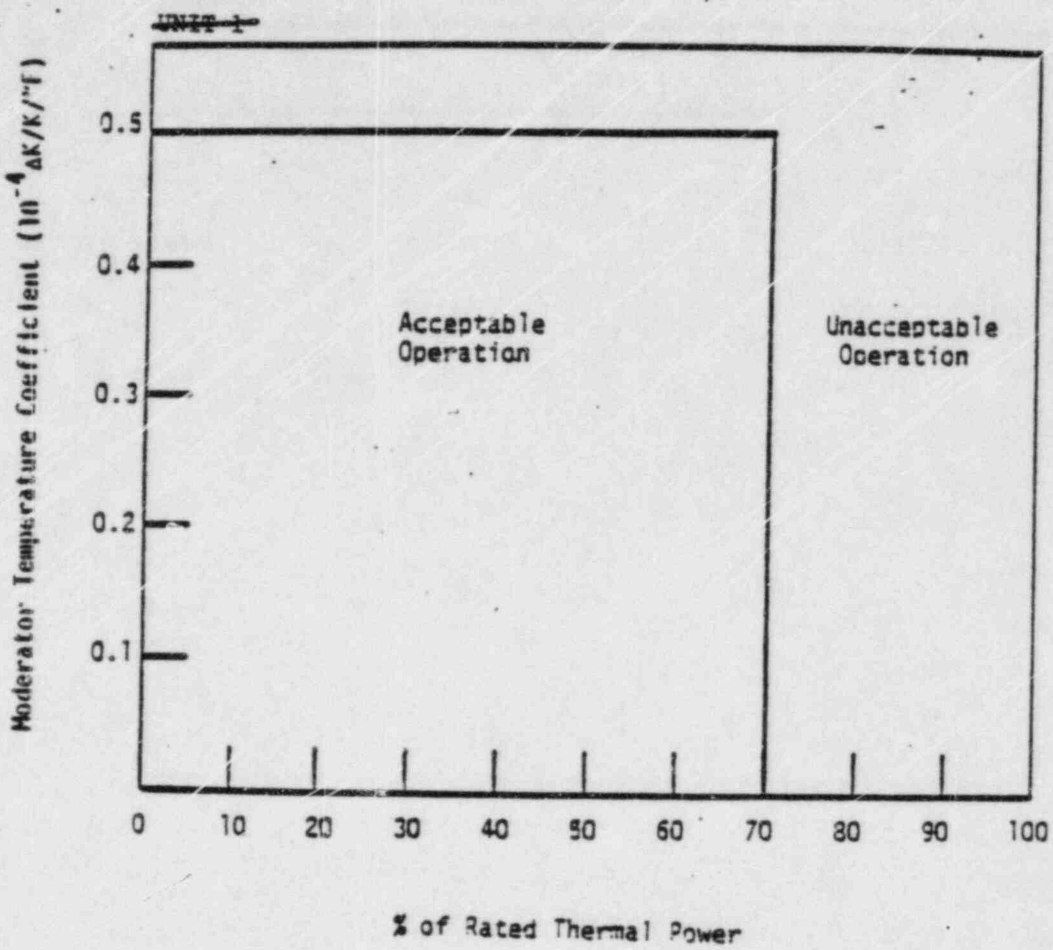


FIGURE 3.1-0

MODERATOR TEMPERATURE COEFFICIENT VS POWER LEVEL ~~(UNIT 1)~~

POWER DISTRIBUTION LIMITS

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and $R_{\cancel{1}}$ ~~R_2~~ shall be maintained within the region of allowable operation shown on Figure 3.2-3 for four loop operation:

Where:

a. $R_{\cancel{1}}$ ~~(Unit 1)~~ $= \dfrac{F^N_{\Delta H}}{1.49\ [1.0 + 0.3\ (1.0 - P)]}$ ~~R_1 (Unit 2) $= \dfrac{F^N_{\Delta H}}{1.49\ [1.0 + 0.2\ (1.0 -}$~~

~~b. R_2 (Unit 1) $= R_1$, R_2 (Unit 2) $= \dfrac{R1}{[1 - RBP(BU)]}$~~

b. ~~c.~~ $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

c. ~~d.~~ $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R since Figure 3.2-3 includes penalties for undetected feedwater venturi fouling of 0.1% and for measurement uncertainties of 1.7% for flow and 4% for incore measurement of $F^N_{\Delta H}$. ~~and~~

~~e. RBP (BU) = Rod Bow Penalty as a function of region average burnup as shown in Figure 3.2-4, where a region is defined as those assemblies with the same loading date (reloads) or enrichment (first core). (Applies to Unit 2 only).~~

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and $R_{\cancel{1}}$ ~~R_2~~ outside the region of acceptable operation shown on Figure 3.2-3:

a. Within 2 hours either:

1. Restore the combination of RCS total flow rate and $R_{\cancel{1}}$ ~~R_2~~ to within the above limits, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of $R_{\cancel{1}}$ $\cancel{R_2}$ and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b. above; subsequent POWER OPERATION may proceed provided that the combination of $R_{\cancel{1}}$ $\cancel{R_2}$ and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation shown on Figure 3.2-3 prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,
2. A nominal 75% of RATED THERMALPOWER, and
3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate determined by process computer readings or digital voltmeter measurement and $R_{\cancel{1}}$ ~~and~~ $\cancel{R_2}$ shall be within the region of acceptable operation of Figure 3.2.3:

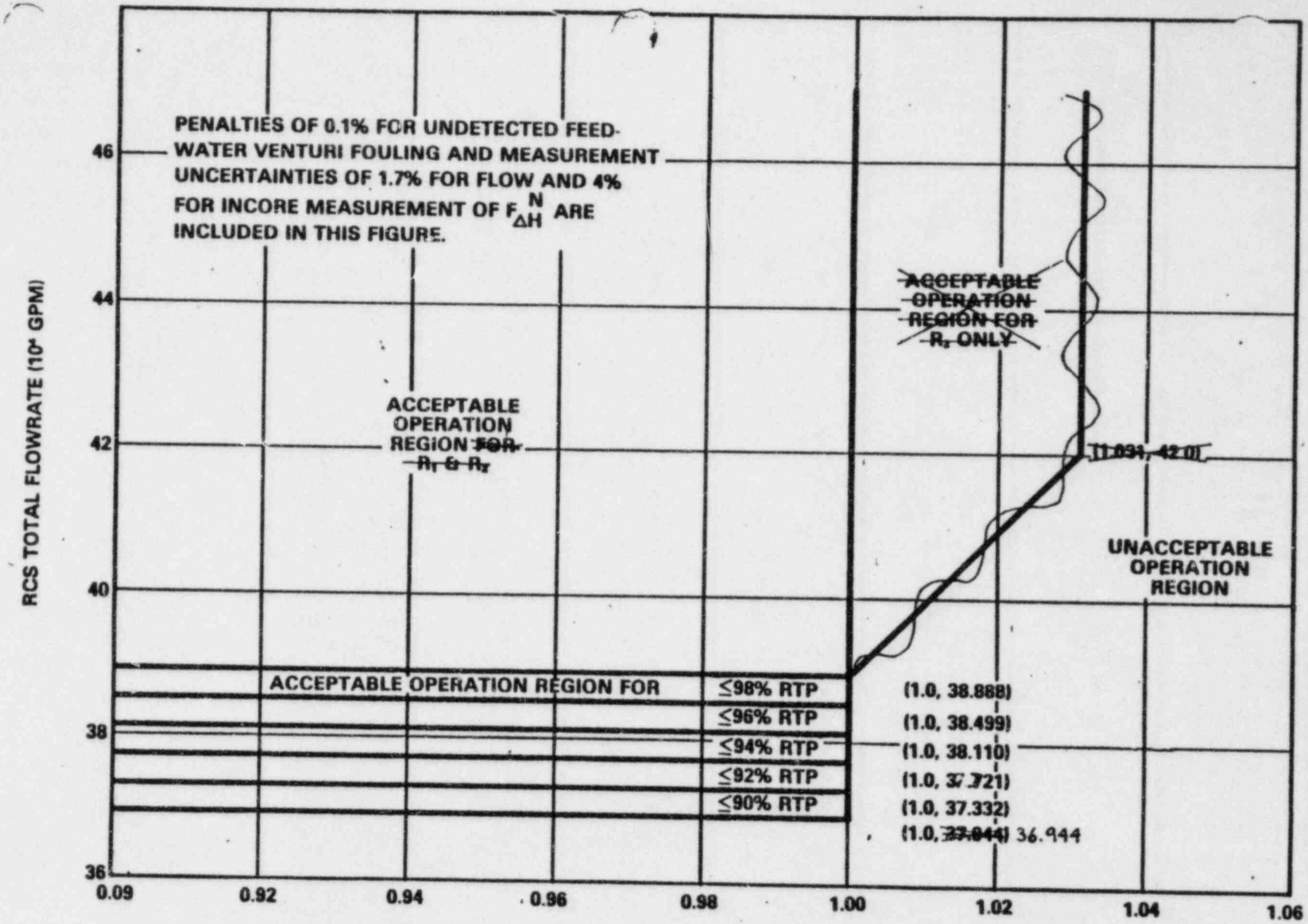
a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation of Figure 3.2-3 at least once per 12 hours when the most recently obtained value~~s~~ of $R_{\cancel{1}}$ ~~and~~ $\cancel{R_2}$, obtained per Specification 4.2.3.2, are assumed to exist.

4.2.3.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.3.5 The RCS total flow rate shall be determined by precision heat balance measurement at least once per 18 months.



$$R_x = F^N_{\Delta H}/1.49[1.0 + \cancel{0.2}\ 0.3\,(1.0-P)]$$

~~$R_2 = R_1/[1 - RBP(BU)]$~~

Figure 3.2-3b RCS FLOW RATE VERSUS R_x ~~and R_2~~ - FOUR LOOPS IN OPERATION (Unit 2)

DELETE

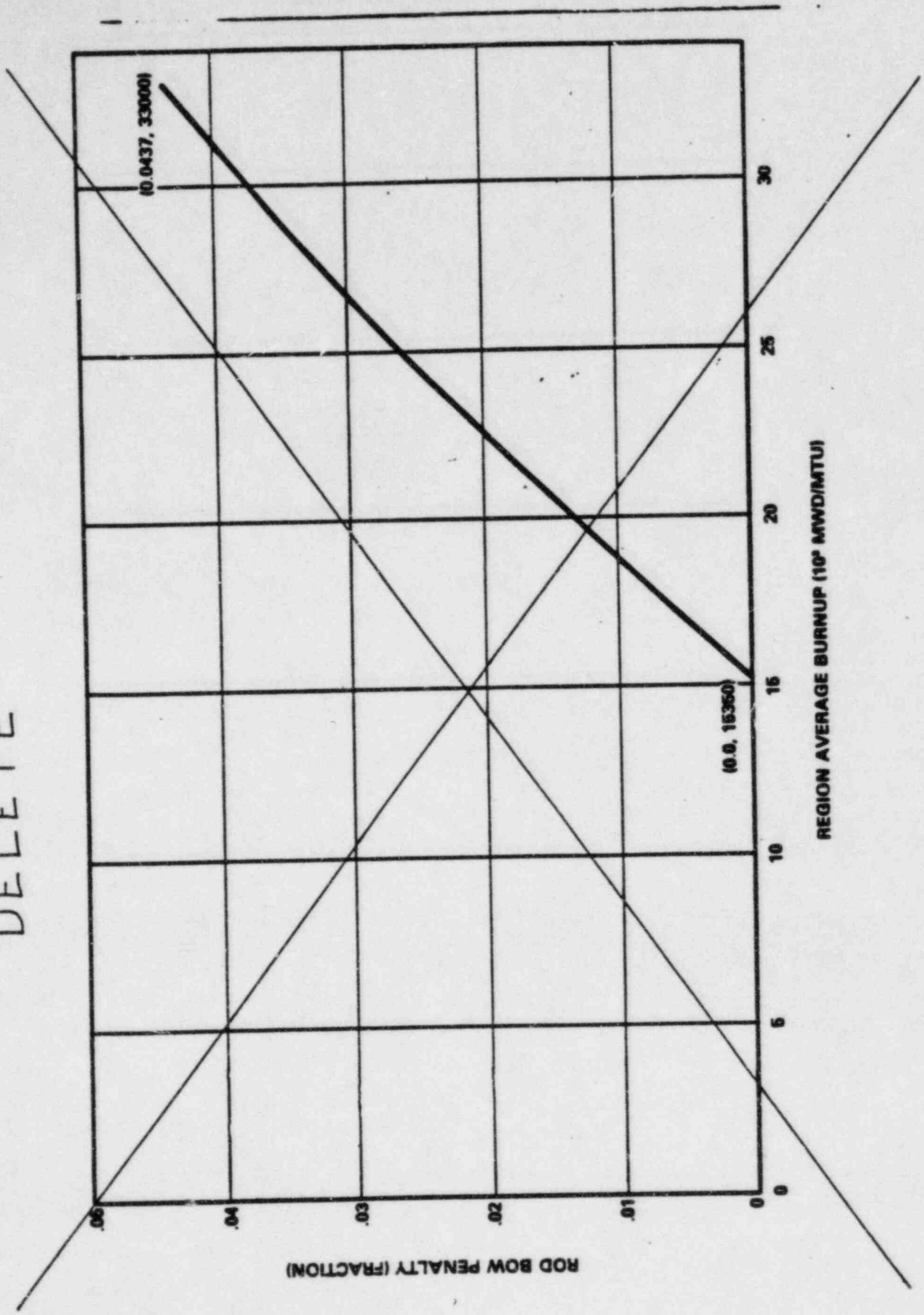


FIGURE 3.2-4 ROD BOW PENALTY AS A FUNCTION OF BURNUP (Unit 2)

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	N.A.
2. Power Range, Neutron Flux	≤ 0.5 second*
3. Power Range, Neutron Flux, High Positive Rate	N.A.
4. Power Range, Neutron Flux, High Negative Rate	≤ 0.5 second*
5. Intermediate Range, Neutron Flux	N.A.
6. Source Range, Neutron Flux	N.A.
7. Overtemperature ΔT	≤ ~~6.0~~ 8.0 seconds*
8. Overpower ΔT	≤ ~~6.0~~ 8.0 seconds*
9. Pressurizer Pressure--Low	≤ 2.0 seconds
10. Pressurizer Pressure--High	≤ 2.0 seconds
11. Pressurizer Water Level--High	N.A.

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. Low Reactor Coolant Flow	
a. Single Loop (Above P-8)	$\leq$ 1.0 second
b. Two Loops (Above P-7 and below P-8)	$\leq$ 1.0 second
13. Steam Generator Water Level--Low-Low	$\leq$ ~~2.0~~ 3.5 seconds
14. Undervoltage-Reactor Coolant Pumps	< 1.5 seconds
15. Underfrequency-Reactor Coolant Pumps	< 0.6 second
16. Turbine Trip	
a. Low Fluid Oil Pressure	N.A.
b. Turbine Stop Valve Closure	N.A.
17. Safety Injection Input from ESF	N.A.
18. Reactor Trip System Interlocks	N.A.
19. Reactor Trip Breakers	N.A.
20. Automatic Trip and Interlock Logic	N.A.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
7. Auxiliary Feedwater		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Steam Generator Water Level--Low-Low		
1) Start Motor-Driven Pumps	> 12% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to > ~~54.9%~~ 40.0% of span at 100% of RATED THERMAL POWER.	> 11% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to > ~~53.9%~~ 39.0% of span at 100% of RATED THERMAL POWER.
2) Start Turbine-Driven Pumps	> 12% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to > ~~54.9%~~ 40.0% of span at 100% of RATED THERMAL POWER.	> 11% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to > ~~53.9%~~ 39.0% of span at 100% of RATED THERMAL POWER.
d. Auxiliary Feedwater Suction Pressure - Low (Suction Supply Automatic Realignment)	≥ 2 psig	≥ 1 psig
e. Safety Injection - Start Motor-Driven Pumps	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values	
f. Station Blackout - Start Motor-Driven Pumps and Turbine-Driven Pump	3464 ± 173 volts with a 8.5 ± 0.5 second time delay	≥ 3200 volts
g. Trip of Main Feedwater Pumps - Start Motor-Driven Pumps	N.A.	N.A.

INSTRUMENTATION

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The Movable Incore Detection System shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of two detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABILITY: When the Movable Incore Detection System is used for:

a. Recalibration of the Excore Neutron Flux Detection System,

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$ ~~and~~ $\cancel{F_{xy}}$

ACTION:

With the Movable Incore Detection System inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The Movable Incore Detection System shall be demonstrated OPERABLE at least once per 24 hours by normalizing each detector output when required for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$, ~~and~~ $\cancel{F_{xy}}$

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1.1 Each cold leg injection accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between;
~~1)~~ 8022 and 8256 gallons ~~(Unit 1),~~
~~2) 8261 and 8496 gallons (Unit 2),~~

c. A boron concentration of between 1900 and 2100 ppm,

d. A nitrogen cover-pressure of between 430 and 484 psig ~~(Unit 1), 400 and 454 psig (Unit 2),~~ and

e. A water level and pressure channel OPERABLE.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one cold leg injection accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one cold leg injection accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 1 hour and in HOT SHUTDOWN within the following 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1.1 Each cold leg injection accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each cold leg injection accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of ~~1.6% of delta k/k (Unit 2)~~, 1.3% delta k/k ~~(Unit 1)~~ is required to control the reactivity transient.
Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% delta k/k SHUTDOWN MARGIN provides adequate protection.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value -4.1×10^{-4} delta k/k/°F. The MTC value of -3.2×10^{-4} delta k/k/°F represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value of -4.1×10^{-4} k/k/°F.

MODERATOR TEMPERATURE COEFFICIENT (Continued)

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within it analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, (5) associated Heat Tracing Systems, and (6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of ~~1.6%~~ 1.3% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 16,321 gallons of 7000-ppm borated water from the boric acid storage tanks or 75,000 gallons of 2000-ppm borated water from the refueling water storage tank (RWST).

With the RCS temperature below 200°F, one Boron Injection System is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single Boron Injection System becomes inoperable.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below 300°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 13 steps from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. As noted on Figure~~s~~ 3.2-3 ~~and 3.2-4~~, RCS flow rate and power ~~$F^N_{\Delta H}$~~ may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the power level is decreased ~~measured $F^N_{\Delta H}$ is also low~~) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R_x as calculated in Specification 3.2.3 and used in Figure 3.2-3, accounts for $F^N_{\Delta H}$ less than or equal to 1.49. This value is used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed. ~~R_2, as defined,~~ allows for the inclusion of a penalty for Rod Bow on DNBR only. Thus, knowing the "as measured" values of $F^N_{\Delta H}$ and RCS flow allows for "tradeoffs" in excess of R equal to 1.0 for the purpose of offsetting the Rod Bow DNBR penalty.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to partially offset this reduction. This credit comes from a generic or plant-specific design margin. For McGuire Unit 2, the margin used to partially offset rod bow penalties is 9.1%. This margin breaks down as follows:

1)	Design limit DNBR	1.6%
2)	Grid spacing K_s	2.9%
3)	Thermal Diffusion Coefficient	1.2%
4)	DNBR Multiplier	1.7%
5)	Pitch Reduction	1.7%

HEAT FLUX HOT CHANNEL FACTOR and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

~~However, the margin used to partially offset rod bow penalties is 5.9% with the remaining 3.2% used to trade off against measured flow being as much as 2% lower than thermal design flow plus uncertainties. The penalties applied to $F^N_{\Delta H}$ to account for rod bow (Figure 3.2-4) as a function of burnup are consistent with those described in Mr. John F. Stolz's (NRC) letter to T. M. Anderson (Westinghouse) dated April 5, 1979 with the difference being due to the amount of margin each unit uses to partially offset rod bow penalties.~~

~~For McGuire Unit 1,~~ Margin between the safety analysis limit DNBRs (1.47 and 1.49 for thimble and typical cells, respectively) and the design limit DNBRs (1.32 and 1.34 for thimble and typical cells, respectively) is maintained. A fraction of this margin is utilized to accommodate the transition core DNBR penalty (2%) and the appropriate fuel rod bow DNBR penalty (WCAP - 8691, Rev. 1)

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of Figure~~s~~ 3.2-3 ~~and 3.2-4~~. Measurement errors of 1.7% for RCS total flow rate and 4% for $F^N_{\Delta H}$ have been allowed for in determination of the design DNBR value.

The measurement error for RCS total flow rate is based upon performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a non-conservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi is included in Figure 3.2-3. Any fouling which might bias the RCS flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e., either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The 12-hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation shown on Figure 3.2-3.

Attachment 3

Analysis of Significant Hazards Consideration

As required by 10 CFR 50.91, this analysis is provided concerning whether the proposed amendments involve significant hazards considerations, as defined by 10 CFR 50.92. Standards for determination that a proposed amendment involves no significant hazards considerations are if operation of the facility in accordance with the proposed amendment would not: 1) involve a significant increase in the probability or consequences of an accident previously evaluated; or 2) create the possibility of a new or different kind of accident from any accident previously evaluated; or 3) involve a significant reduction in a margin of safety.

The proposed amendments would change plant operating limitations given in the Technical Specifications affected by the use of the optimized fuel assembly design in McGuire Unit 2 Cycle 2. The reference safety evaluation report submitted by Mr. H. B. Tucker's November 14, 1983 letter to Mr. H. R. Denton summarizes the evaluation performed on the region-by-region reload transition from the McGuire Units 1 and 2 standard (STD) fueled cores to cores with all optimized fuel (OFA). The report examines the differences between the Westinghouse STD design and OFA design and evaluates the effects of these differences for the transition to an all OFA core. The report justifies the compatibility of the OFA design with the STD design in a transition core as well as a full OFA core. The report also contains summaries of the mechanical, nuclear, thermal-hydraulic, and accident evaluations.

The McGuire Unit 2/Cycle 2 reload safety evaluation (Attachment 2A) presents an evaluation which demonstrates that the core reload will not adversely affect the safety of the plant. All of the accidents comprising the licensing bases which could potentially be affected by the fuel reload were reviewed for the Unit 2 Cycle 2 design. The results of new analyses are included in the reference safety evaluation report and the Unit 2/Cycle 2 RSE, and the justification for the applicability of previous results for the remaining analyses is presented. The results of evaluation/analysis and tests lead to the following conclusions:

a. The Westinghouse OFA reload fuel assemblies for McGuire 1 and 2 are mechanically compatible with the current STD design, control rods, and reactor internals interfaces. Both fuel assemblies satisfy the current design bases for the McGuire units.

b. Changes in the nuclear characteristics due to the transition from STD to OFA fuel will be within the range normally seen from cycle to cycle due to fuel management effects.

c. The reload OFAs are hydraulically compatible with the current STD design.

d. The accident analyses for the OFA transition core were shown to provide acceptable results by meeting the applicable criteria, such as, minimum DNBR, peak pressure, and peak clad temperature, as required. The previously reviewed and licensed safety limits are met.

e. Plant operating limitations given in the Technical Specifications will be satisfied with the proposed changes.

From these evaluations, it is concluded that the Unit 2 Cycle 2 design does not cause the previously acceptable safety limits to be exceeded.

The commission has provided examples of amendments likely to involve no significant hazards considerations (48 FR 14870). One example of this type is (vi), "A change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where results of the change are clearly within all acceptable criteria with respect to the system or component specified in the standard review plan: for example, a change resulting from the application of a small refinement of a previously used calculational model or design method". Because the evaluations previously discussed show that all of the accidents comprising the licensing bases which could potentially be affected by the fuel reload were reviewed for the Unit 2 Cycle 2 design and conclude that the reload design does not cause the previously acceptable safety limits to be exceeded, the above example can be applied to this situation. In addition, the NRC has previously concluded that similar changes for McGuire Unit 1 Cycle 2 did not involve significant hazards considerations (these changes were subsequently approved via Ms. E. G. Adensam's (NRC/ONRR) letters to Mr. H. B. Tucker dated April 20, June 21, and September 13, 1984).

Based upon the preceding analyses, Duke Power Company concludes that the proposed amendments do not involve a significant hazards consideration.