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

March 22, 1985

Docket Nos: 50-369
and 50-370

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

Subject: Issuance of Amendment No. 42 to Facility Operating License NPF-9 and Amendment No. 23 to Facility Operating License NPF-17 - McGuire Nuclear Station, Units 1 and 2

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 42 to Facility Operating License NPF-9 and Amendment No. 23 to Facility Operating License NPF-17 for the McGuire Nuclear Station, Units 1 and 2. These amendments are issued in response to your application dated November 16, 1984. The other changes requested in that application were addressed in Amendment Nos. 39 and 20, respectively.

The amendments change the Technical Specifications to reflect the transition to the use of optimized fuel assemblies. The amendments are effective as of their dates of issuance.

A copy of the related safety evaluation supporting Amendment No. 42 to Facility Operating License NPF-9 and Amendment No. 23 to Facility Operating License NPF-17 is enclosed.

Sincerely,

Elinor G. Adensam

Elinor G. Adensam, Chief
Licensing Branch No. 4
Division of Licensing

Enclosures:
1. Amendment No. 42 to NPF-9
2. Amendment No. 23 to NPF-17
3. Safety Evaluation

cc w/encl:
See next page

DESIGNATED ORIGINAL
Certified By

McGuire

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

cc: Mr. A. Carr
Duke Power Company
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Mr. F. J. Twogood
Power Systems Division
Westinghouse Electric Corp.
P. O. Box 355
Pittsburgh, Pennsylvania 15230

Mr. Robert Gill
Duke Power Company
Nuclear Production Department
P. O. Box 33189
Charlotte, North Carolina 28242

J. Michael McGarry, III, Esq.
Bishop, Liberman, Cook, Purcell
and Reynolds
1200 Seventeenth Street, N.W.
Washington, D. C. 20036

Mr. Wm. Orders
Senior Resident Inspector
c/o U.S. Nuclear Regulatory
Commission
Route 4, Box 529
Hunterville, North Carolina 28078

J. Nelson Grace, Regional Admin.
U.S. Nuclear Regulatory Commission,
Region II
101 Marietta Street, N.W., Suite 2900
Atlanta, Georgia 30323

R. S. Howard
Operating Plants Projects
Regional Manager
Westinghouse Electric Corporation -
R&D 701
P. O. Box 2728
Pittsburgh, Pennsylvania 15230

Dr. John M. Barry
Department of Environmental Health
Mecklenburg County
1200 Blythe Boulevard
Charlotte, North Carolina 28203

County Manager of Mecklenburg County
720 East Fourth Street
Charlotte, North Carolina 28202

EIS Coordinator
U.S. Environmental Protection Agency
Region IV Office
345 Courtland Street, N.E.
Atlanta, Georgia 30365

Chairman, North Carolina
Utilities Commission
430 North Salisbury Street
Dobbs Building
Raleigh, North Carolina 27602

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

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

DUKE POWER COMPANY

DOCKET NO. 50-369

McGUIRE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 42
License No. NPF-9

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the McGuire Nuclear Station, Unit 1 (the facility) Facility Operating License No. NPF-9 filed by the Duke Power Company (licensee) dated November 16, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachments to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-9 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 42 , are hereby incorporated into this license.

The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam

Elinor G. Adensam, Chief
Licensing Branch No. 4
Division of Licensing

Attachment:
Technical Specification Changes

Date of Issuance: March 22, 1985

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

DUKE POWER COMPANY

DOCKET NO. 50-370

McGUIRE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 23
License No. NPF-17

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the McGuire Nuclear Station, Unit 2 (the facility) Facility Operating License No. NPF-17 filed by the Duke Power Company (licensee) dated November 16, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachments to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-17 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 23, are hereby incorporated into this license.

The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam

Elinor G. Adensam, Chief
Licensing Branch No. 4
Division of Licensing

Attachment:
Technical Specification Changes

Date of Issuance: March 22, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 42

FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

AND

TO LICENSE AMENDMENT NO. 23

FACILITY OPERATING LICENSE NO. NPF-17

DOCKET NO. 50-370

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are provided to maintain document completeness.

Amended Page	Overleaf Page
2-2a	
2-5	
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Amended Page	Overleaf Page
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B3/4 1-2	
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B3/4 2-2	
B3/4 2-2a	
B3/4 2-3	
B3/4 2-4	
B3/4 2-5	
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Delete page B3/4 2-6.

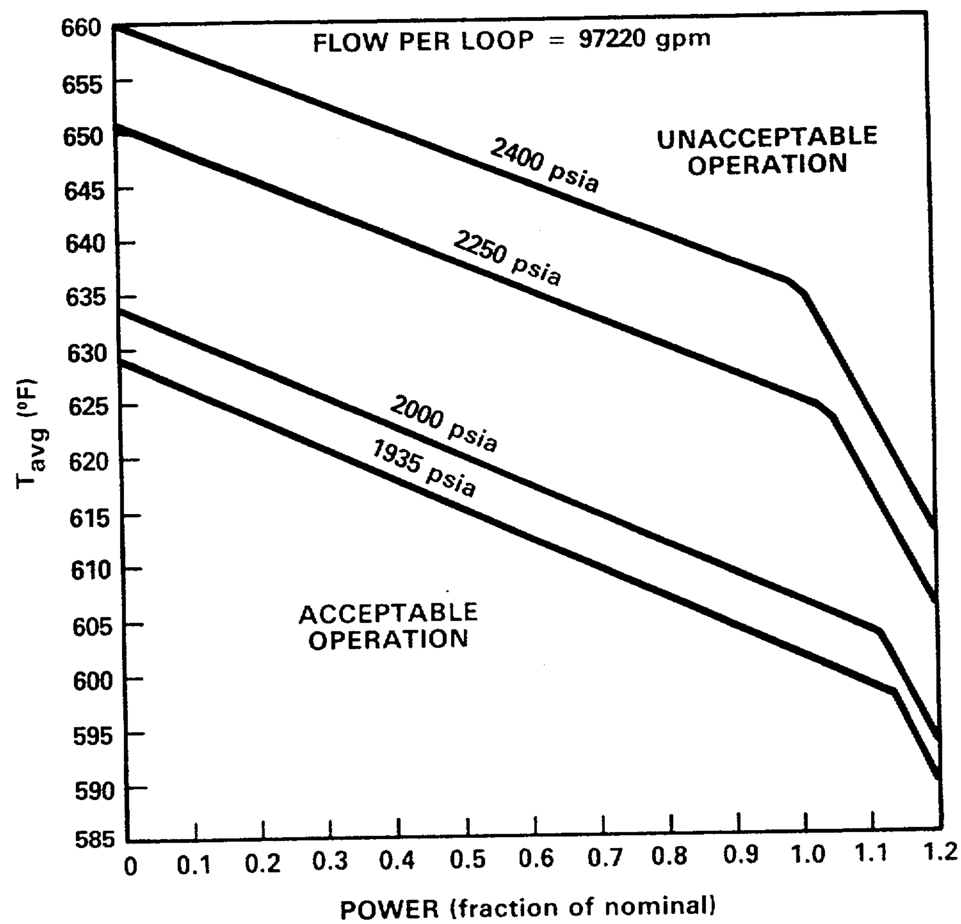

FIGURE 2.1-1b

UNIT 2

REACTOR CORE SAFETY LIMIT - 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 High Setpoint - ≤ 109% of RATED THERMAL POWER	Low Setpoint - ≤ 26% 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 x 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 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% (Unit 1), 40% (Unit 2) of span at 100% of RATED THERMAL POWER.	≥ 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	≥ 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

$$\Delta T\ \left(\frac{1+\tau_1 S}{1+\tau_2 S}\right)\left(\frac{1}{1+\tau_3 S}\right) \leq \Delta T_0\ \left\{K_1 - K_2\left(\frac{1+\tau_4 S}{1+\tau_5 S}\right)\left[T\left(\frac{1}{1+\tau_6 S}\right)-T'\right] + K_3(P-P') - f_1(\Delta I)\right\}$$

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, $\tau_1 \geq 8$ sec., $\tau_2 \leq 3$ sec.,
$\frac{1}{1+\tau_3}$	=	Lag compensator on measured ΔT,
τ_3	=	Time constants utilized in the lag compensator for ΔT, $\tau_3 \leq 2$ sec. (Unit 1), 6 sec. (Unit 2)
ΔT_0	=	Indicated ΔT at RATED THERMAL POWER,
K_1	$\leq$	1.200 (Unit 2), 1.4060 (Unit 1),
K_2	=	0.0222
$\frac{1+\tau_4 S}{1+\tau_5 S}$	=	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}, $\tau_4 \geq 28$ sec, $\tau_5 \leq 4$ 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, $\tau_6 \leq 2$ sec (Unit 1), 6 sec (Unit 2)

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

K_3 = 0.001095,

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 -29% and +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 -29% (Unit 2), -41% (Unit 1), the ΔT Trip Setpoint shall be automatically reduced by 3.151% of its value at RATED THERMAL POWER; and

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds +9.0% (Unit 2), -4.0% (Unit 1), the ΔT Trip Setpoint shall be automatically reduced by 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

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

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,
$\frac{1}{1 + \tau_6 S}$	=	As defined in Note 1,
τ_6	=	As defined in Note 1,
K_6	=	0.00169/°F 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"	=	$\leq$ 588.2°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. 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 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 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.

3/4.1.1 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 1.3% delta k/k for four loop operation.

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

ACTION:

With the SHUTDOWN MARGIN less than 1.3% delta k/k, 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 1.3% delta k/k:

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.

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than the limits shown in Figure 3.1-0, and
b. 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. - MODES 1 and 2* only.#
Specification 3.1.1.3b. - MODES 1, 2, and 3 only.#

ACTION:

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

   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. 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. 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 3.1.1.3b. 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 3.1.1.3a., 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 3.1.1.3b., at least once per 14 EFPD during the remainder of the fuel cycle.

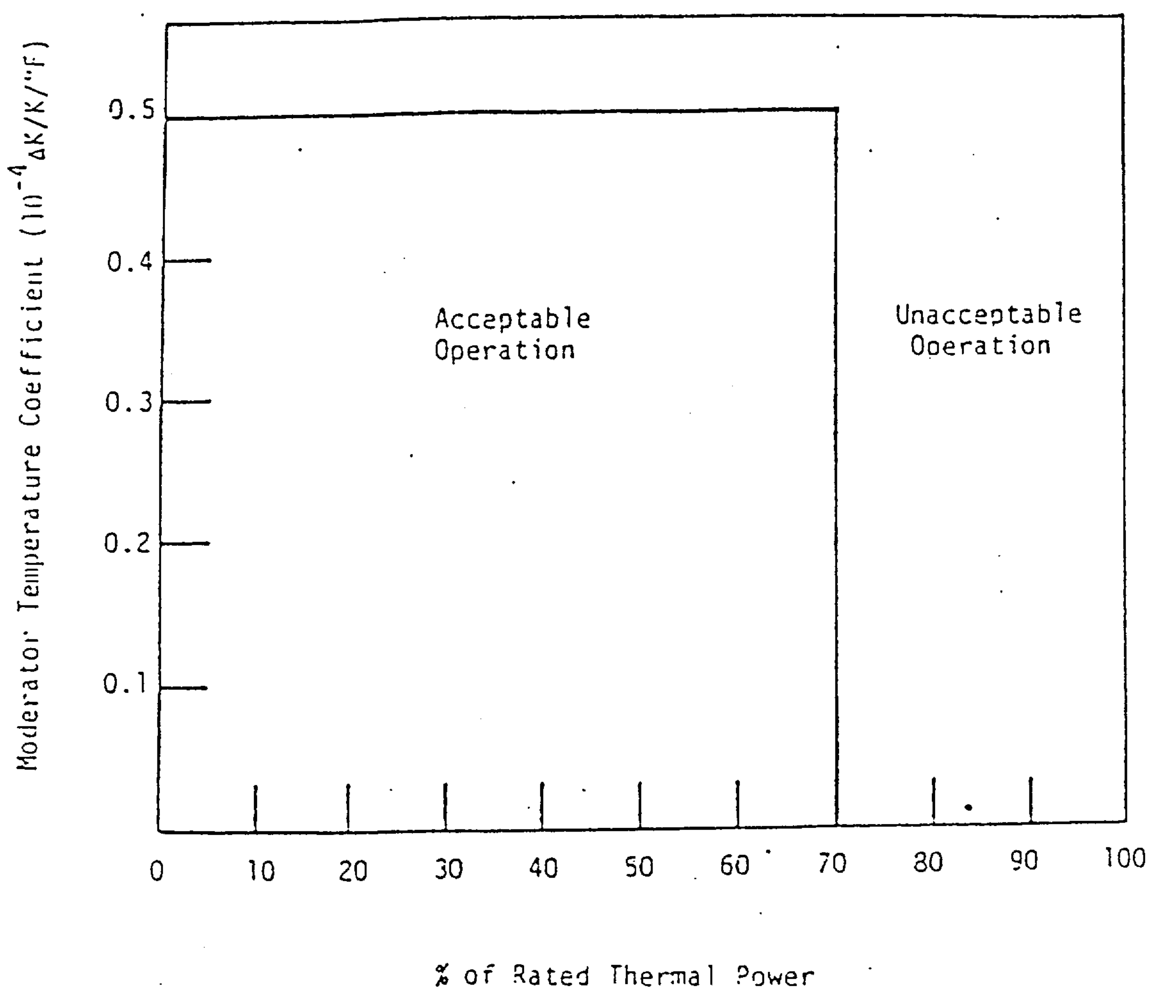

FIGURE 3.1-0

MODERATOR TEMPERATURE COEFFICIENT VS POWER LEVEL

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (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.

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 in conjunction with the surveillance requirements of Specification 3/4.2.2 or by linear interpolation between the most recently measured value and the calculated value at the end of cycle life. The provisions of Specification 4.0.4 are not applicable.

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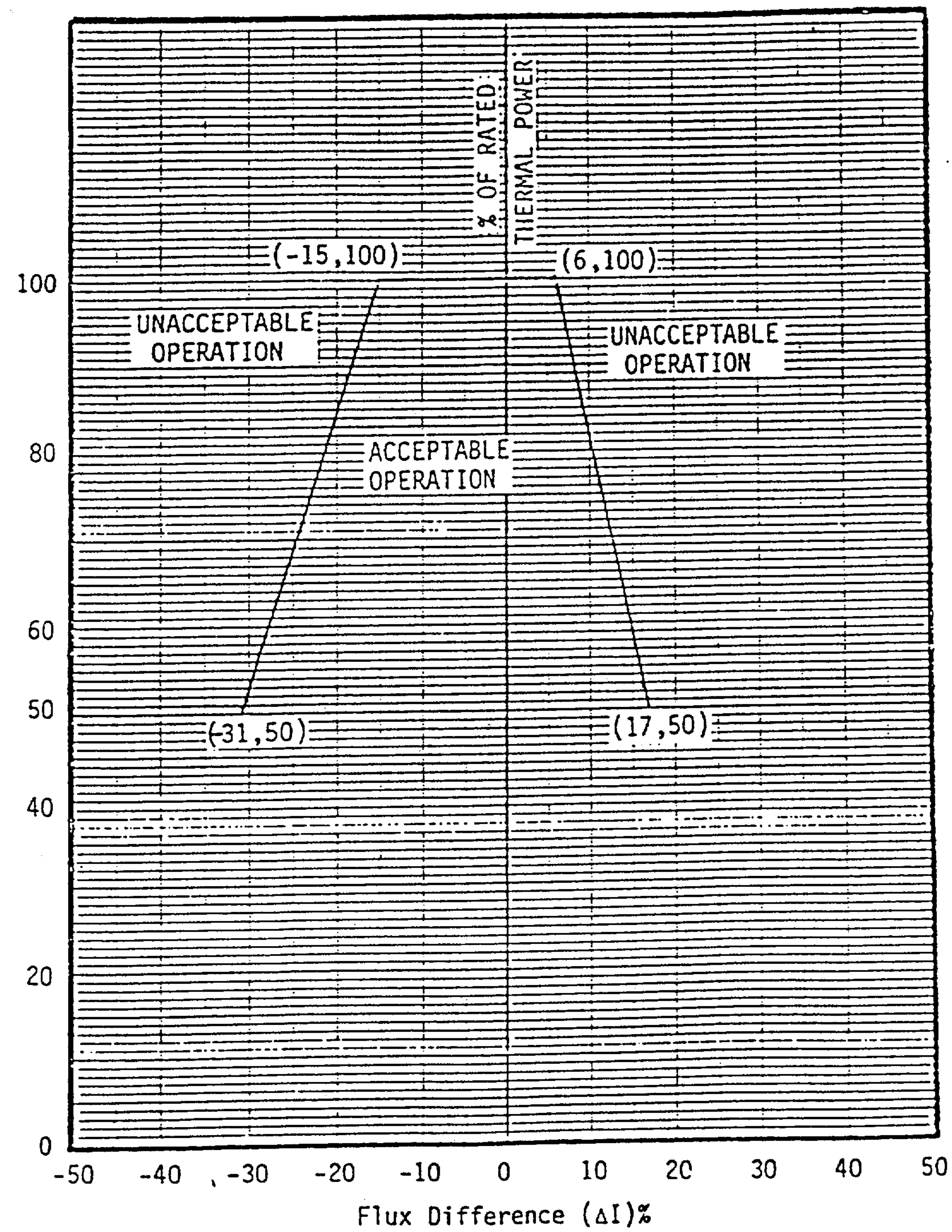

FIGURE 3.2-1
AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

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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:

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

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

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

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

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.

SURVEILLANCE REQUIREMENTS

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:

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

$$F_Q^M(z) \leq \frac{2.26 \quad x\ K(z)}{P\ x\ W(z)} \text{ for } P > 0.5 \text{ (Unit 2)}$$

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

$$F_Q^M(z) \leq \frac{2.26 \quad x\ K(z)}{W(z)\ x\ 0.5} \text{ for } P \leq 0.5 \text{ (Unit 2)}$$

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

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

$$\left\{ \left(\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z) \times W(z)}{\frac{2.26}{P} \times K(z)} \right] \right) - 1 \right\} \times 100 \quad \text{for } P \geq 0.5 \text{ (Unit 2)}$$

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

$$\left\{ \left(\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z) \times W(z)}{\frac{2.26}{0.5} \times K(z)} \right] \right) - 1 \right\} \times 100 \quad \text{for } P < 0.5 \text{ (Unit 2)}$$

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.

SURVEILLANCE REQUIREMENTS (Continued)

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 FQ surveillance is maintained pursuant to Specification 4.2.2.4. APL^{BL} is defined as:

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

$$APL^{BL} = \underset{\text{over } Z}{\text{minimum}} \left[\frac{(2.26 \times K(Z)}{F_Q^M(Z) \times W(Z)_{BL}} \right] \times 100\% \text{ (Unit 2)}$$

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:

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

$$F_Q^M(Z) \leq \frac{2.26 \times K(Z)}{P \times W(Z)_{BL}} \text{ for } P > APL^{ND} \text{ (Unit 2)}$$

where: $F_Q^M(Z)$ is the measured $F_Q(Z)$. The F_Q limit is 2.15 (Unit 1) and 2.26 (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

SURVEILLANCE REQUIREMENTS (Continued)

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:

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

$$[(\text{max. over z of } [\frac{F_Q^M(Z) \times W(Z)_{BL}}{\frac{2.26}{P} \times K(Z)}]) - 1] \times 100 \text{ for } P \geq APL^{ND} \text{ (Unit 2)}$$

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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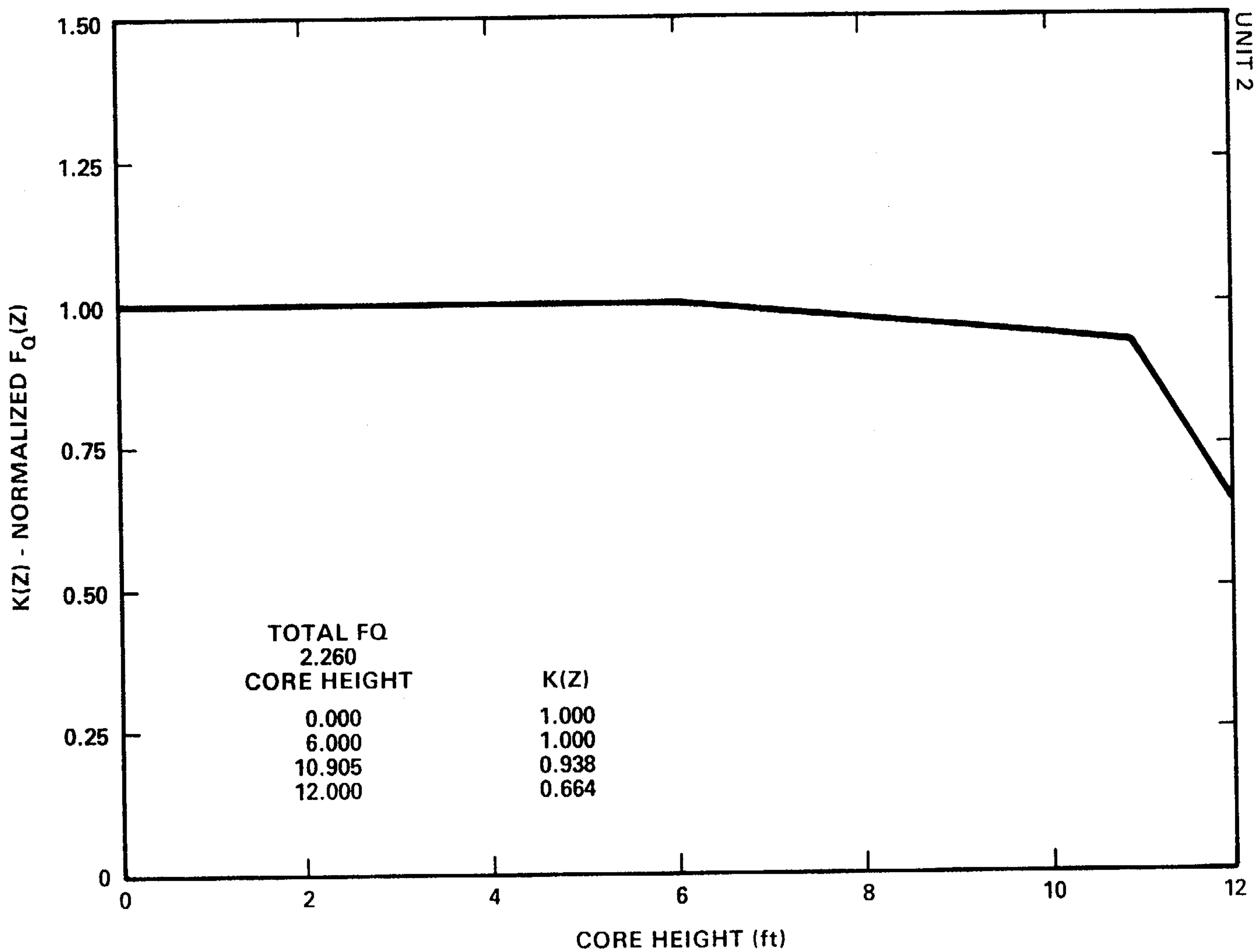

FIGURE 3.2-2b

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

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 shall be maintained within the region of allowable operation shown on Figure 3.2-3 for four loop operation:

Where:

a. $R = \dfrac{F^N_{\Delta H}}{1.49\,[1.0 + 0.3\,(1.0 - P)]}$,

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

c. $F^N_{\Delta H}$ = 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 $F^N_{\Delta H}$.

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and R 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 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 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 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 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 of R obtained per Specification 4.2.3.2, is 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.

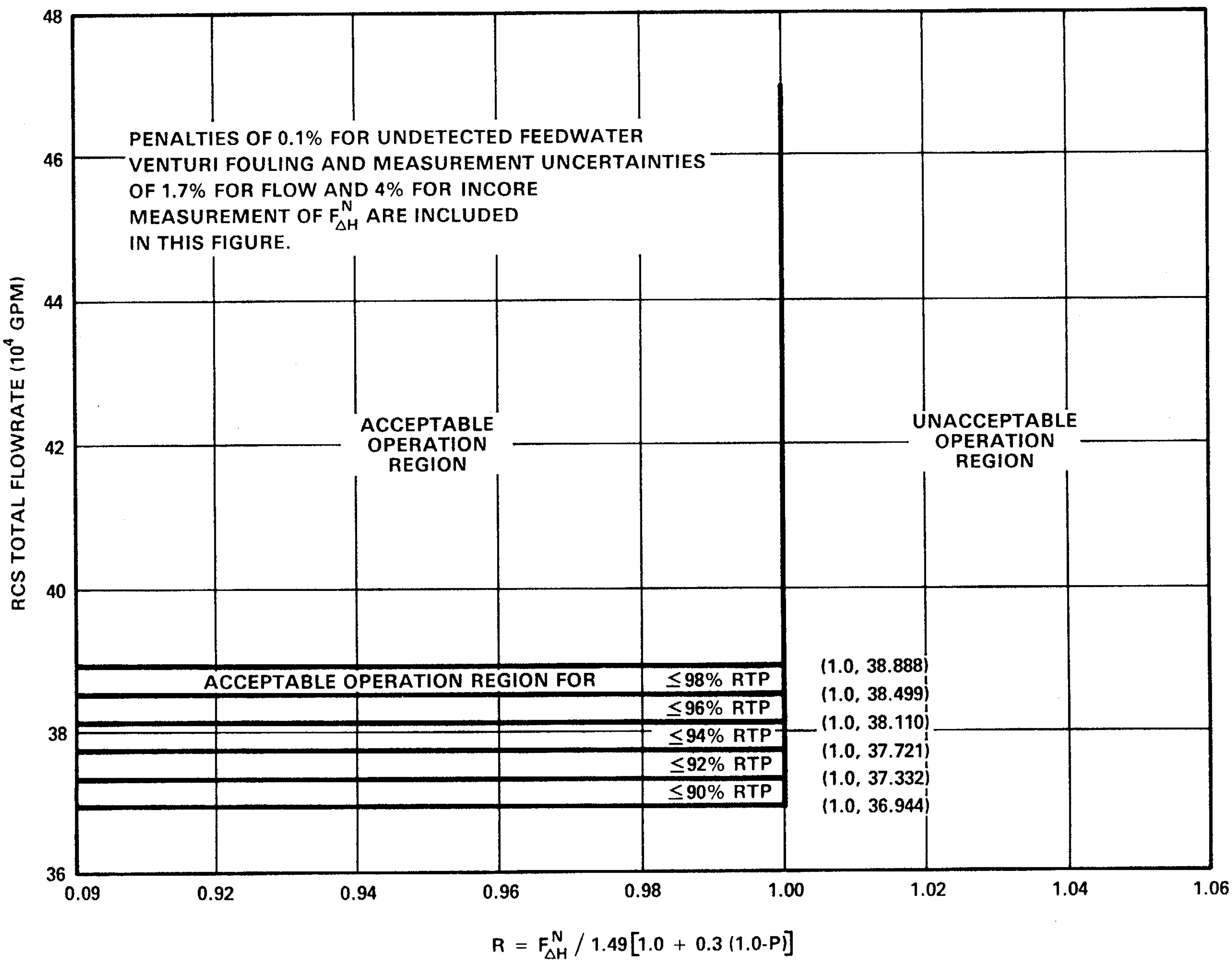

Figure 3.2-3b RCS FLOW RATE VERSUS R - FOUR LOOPS IN OPERATION (Unit 2)

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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	$\leq$ 0.5 second*
3. Power Range, Neutron Flux, High Positive Rate	N.A.
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 0.5 second*
5. Intermediate Range, Neutron Flux	N.A.
6. Source Range, Neutron Flux	N.A.
7. Overtemperature ΔT	$\leq$ 6.0 (Unit 1), 8.0 (Unit 2) seconds*
8. Overpower ΔT	$\leq$ 6.0 (Unit 1), 8.2 (Unit 2) seconds*
9. Pressurizer Pressure--Low	$\leq$ 2.0 seconds
10. Pressurizer Pressure--High	$\leq$ 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 (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
4. Steam Line Isolation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High-High	≤ 2.9 psig	≤ 3.0 psig
d. Negative Steam Line Pressure Rate - High	≤ -100 psi/sec	≤ -120 psi/sec
e. Steam Line Pressure - Low	≥ 585 psig	≥ 565 psig
5. Turbine Trip and Feedwater Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Steam Generator Water level--High-High (P-14)	≤ 82% of narrow range instrument span each steam generator	≤ 83% of narrow range instrument span each steam generator
6. Containment Pressure Control System		
a. Start Permissive	≤ 0.25 psid	≤ 0.25 psid
b. Termination	≤ 0.25 psid	≤ 0.25 psid

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% (Unit 1), 40.0% (Unit 2) of span at 100% of RATED THERMAL POWER.	≥ 11% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ 53.9% (Unit 1), 39.0% (Unit 2) 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% (Unit 1), 40.0% (Unit 2) of span at 100% of RATED THERMAL POWER.	≥ 11% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ 53.9% (Unit 1), 39.0% (Unit 2) 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.

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 $F_Q(Z)$

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 $F_Q(Z)$

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 8022 and 8256 gallons,

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

d. A nitrogen cover-pressure of between 430 and 484 psig, 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 REACTIVITY CONTROL SYSTEMS

BASES

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 1.3% delta k/k 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 1.3% 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.

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.

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 (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)

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 indicated AFD to relatively small target band and power swings (AFD target band of ±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.

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:

a. Control rods in a single group move together with no individual rod insertion differing by more than $\pm$ 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 3.2-3, RCS flow rate and power may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the 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 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.

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

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

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 3.2-3. 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.

The hot channel factor $F^M_Q(z)$ 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.

6.9.1.9 RADIAL PEAKING FACTOR LIMIT REPORT

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 Regulation, 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.

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

SAFETY EVALUATION REPORT

RELATED TO AMENDMENT NO. 42 TO FACILITY OPERATING LICENSE NPF-9

AND TO AMENDMENT NO. 23 TO FACILITY OPERATING LICENSE NPF-17

DUKE POWER COMPANY

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

I. INTRODUCTION

By letter dated November 16, 1984, Duke Power Company requested changes to Technical Specifications to reflect the transition to the use of optimized fuel assemblies (OFA). One of the requested changes was addressed by Amendment Nos. 39 and 20 to McGuire Nuclear Station, Units 1 and 2, Facility Operating Licenses NPF-9 and NPF-17, respectively. This evaluation addresses the remaining changes.

II. EVALUATION

This submittal is closely related to previous submittals by Duke Power Company for the Unit 1 first reload and for the generic transition to OFA loadings for Units 1 and 2 (enclosures to references 2 and 3).

The Unit 2, Cycle 2 reload is very similar to the Unit 1, Cycle 2 reload and change to OFA fuel, and almost everything reviewed and approved for it and the associated Technical Specification changes as described by Amendment Nos. 32 and 13 to McGuire Nuclear Station, Units 1 and 2, Facility Operating Licenses NPF-9 and NPF-17, respectively, is directly applicable to Unit 2. Unit 2 is being reloaded with 60 new OFA fuel assemblies as was Unit 1. The core parameters related to transient analyses, for the most part, will remain within the range covered by the approved generic transition of OFA analyses as did Unit 1. Where these parameters are changed, the transient events have been reexamined. There are a number of changes to Unit 2 relating to analysis methodology changes and operational parameter changes. These were covered in the Unit 1 and generic reviews. The following changes for Unit 2 were evaluated and found acceptable in the previous Unit 1 Amendment No. 32 and need not be further evaluated here.

1. Change to OFA fuel; fuel mechanical design, nuclear design, thermal-hydraulic design

2. Change in axial power distribution control from constant axial offset control (CAOC) to relaxed axial offset control (RAOC) or base load operation

3. Change from standard thermal-hydraulic design methodology to improved thermal-design procedure using WRB-1

4. Change to allow a positive Moderator Temperature Coefficient over part of the operating power range

5. Change of shutdown margin in Modes 1, 2, 3 and 4 from 1.6 to 1.3 percent delta k

6. Change of $F_{\Delta H}$ power dependent modifier from 0.2 to 0.3

7. Removal of rod bow related requirement for $F_{\Delta H}$.

As with the Unit 1 reload, core nuclear parameters for Unit 2 reload fall within bounds used in analyses for the generic OFA submittal analyses, and new transient and accident analyses are not required because of these parameters. As with Unit 1, the dropped rod events were reanalyzed for Unit 2, as required by the approved methodology, and were satisfactory.

Differences from Unit 1 Review

The Unit 2 submittal (and review) differs from the Unit 1 submittal primarily in two areas, (1) a new loss of coolant accident (LOCA) analysis, and (2) a reevaluation of transients and accidents because of a core flow reduction relative to the generic OFA and Unit 1 reload analyses.

LOCA

LOCA for Unit 1 was reanalyzed using analysis applicable for transition and full OFA cores for McGuire 1 and 2 as discussed in the generic transition OFA report. However, the Unit 2 analysis used BART (WCAP-9561) for core reflood heat transfer calculations. BART is approved for use on non-UHI plants but had not been approved for a UHI plant such as McGuire Unit 2. This methodology and the analysis for Unit 2 have been reviewed and are acceptable. This analysis for Unit 2 met LOCA criteria using a power peaking factor, F_Q, of 2.26, and this value has been incorporated in the Unit 2 Technical Specifications.

Reduced Core Flow

The generic transition OFA submittal assumed a Thermal Design Flow (TDF) of 386,000 gpm. For Unit 2, Cycle 2 the TDF will be 382,000 gpm. This is a one percent reduction in core flow from the approved analysis. As a result of this reduction, all relevant transients and accident analyses from the generic report were reexamined and when necessary reanalyzed and departure from nucleate boiling (DNB) and non-DNB limits evaluated; and the protection system setpoints and time constants were reviewed and recalculated and changed where necessary.

The reexamination verified that the core DNB limits are unchanged from the generic OFA report and Unit 1 reload values, and the DNB basis is met for all the relevant transients. The Technical Specification limits relating to DNB remain unchanged but the vessel exit boiling limits become more restrictive.

Each event in which non-DNB limits are of interest was also reexamined.

The control rod withdrawal at zero power, loss of load, steamline break and locked rotor events were reexamined to verify that fuel and clad temperature and system pressure changes (which were all small) would remain within limits. For the steamline break this was determined via conclusions that the return to power was less severe. The loss of feedwater/station blackout, rupture of main feedwater line, and limiting control rod ejection events were reanalyzed with reduced flow and found to fall within limits. The primary events for overtemperature and overpower ΔT trip protection, control rod withdrawal at power and small steamline breaks, were reanalyzed, using new setpoints and time constants and met DNB limits.

For the loss of feedwater/station blackout and rupture of main feedwater line events the steam generator level low-low setpoint used a revised value in the reanalysis, and these values are in the new Technical Specifications.

The new LOCA analysis used the reduced flow value.

Our review of this reexamination has concluded that a suitable examination of the effects of the decreased flow has been carried out and, with the related review of the Technical Specifications, appropriate core limits will be maintained.

Technical Specifications

A number of Technical Specification changes are proposed for the Unit 2, Cycle 2 reload operation. Most of these changes are the same as (or have only minor variations from) those for Unit 1, Cycle 2. This applies to both specification changes and corresponding bases changes. The Unit 2 changes were presented both in Attachment 1 and Attachment 2A to the November 16, 1984, submittal. Attachment 1 also contained a few Technical Specification changes for both Units 1 and 2 that are primarily administrative changes. We have reviewed the proposed changes, line by line, and find them acceptable. The following list of changes (from Attachment 1) does not include further discussion where the change has already been discussed in the previous Unit 1 Amendment No. 32.

Technical Specification Changes

Section 2.1; Figure 2.1-1b: The safety limits for DNB for Unit 2 have not further changed beyond the changes resulting from the use of OFA fuel and corresponding changes in analyses methodology discussed in the Unit 1 SER, but the boiling limits are more restrictive because of the change in core flow. The bases for Section 2.1 have changed to reflect the OFA related thermal-hydraulic methodology changes. These changes are acceptable.

Section 2.2; Table 2.2-1: The change to a lower core flow, to the altered steam generator water level low-low setpoint, and to the overpower and over temperature T setpoints and time constants for Unit 2 are given in this table. These changes are approved as a result of the review of the analyses of the effects of the flow and setpoint changes on transients and accidents and the changes from using OFA fuel and the related methodology (as in Unit 1). The procedures and methodology for overpower, overtemperature T trip setpoint changes (Reference 4) are standard as used for all cycles of Westinghouse designed reactors approved by the staff and are acceptable.

Section 3/4.1.1.1 and Bases (and Bases for 3/4.1.2): The change for the shutdown margin in Modes 1, 2, 3 and 4 is from 1.6 to 1.3 percent Δk as in Unit 1.

Section 3/4.1.1.3: This change is to allow a positive moderator temperature coefficient as in Unit 1.

Section 3/4.2.1 and Bases: The change is from CAOC to RAOC and Base Load Operation as in Unit 1 (including the supplemental review for Unit 1 Base Load).

Section 3/4.2.2: The change in Unit 2 to an F_Q of 2.26 is approved as a result of the approval of the LOCA analysis using this value. The change to F_Q surveillance (from F_{xy} surveillance) is as in Unit 1.

Section 3/4.2.3: The change in the $F_{\Delta H}$ power factor from 0.2 to 0.3 and the elimination of the rod bow factor are as in Unit 1.

Table 3.3-2 and 3.3-4: The changes in time constants and setpoints are the same as in Section 2.2, Table 2.2-1.

Section 3/4.3.3.2: This change in Unit 2 and for consistency, in Unit 1, reflects the elimination of F_{xy} surveillance.

Section 3.5.1.1: The cold leg injection accumulator volume and pressure values are changed to those used in the LOCA analyses and are acceptable.

Section 6.9.1.9: This is a reporting requirement for W (z) values for RAOC as in Unit 1.

III. ENVIRONMENTAL CONSIDERATION

These amendments involve a change in use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

IV. CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (49 FR 50802) on December 31, 1984, and consulted with the state of North Carolina. No public comments were received, and the state of North Carolina did not have any comments.

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

V. REFERENCES

1. Letter to H. R. Denton (NRC) from H. B. Tucker (Duke Power) "McGuire Nuclear Station, Docket Nos. 50-369 and 50-370, McGuire 2/Cycle 2 OFA Reload", November 16, 1984.

2. Letter to E. G. Adensam (NRC) from H. B. Tucker (Duke Power) "McGuire Nuclear Station, Docket Nos. 50-369 and 50-370, McGuire 1/Cycle 2 OFA Reload", December 12, 1983.

3. Letter to E. G. Adensam (NRC) from H. B. Tucker (Duke Power) "McGuire Nuclear Station, Docket Nos. 50-369 and 50-370", November 14, 1983.

4. S. L. Ellenberger, et al., "Design Basis for the Thermal Overpower T and Thermal Overtemperature ΔT Trip Functions", WCAP-8745, March 1977.

Principal Contributors: Jon B. Hopkins, Licensing Branch No. 4, DL
Howard J. Richings, Core Performance Branch, DSI

Dated: March 22, 1985

Ma_n 22, 1985

AMENDMENT NO. 42 TO FACILITY OPERATING LICENSE NPF-9 - McGUIRE NUCLEAR STATION, UNIT 1
AMENDMENT NO. 23 TO FACILITY OPERATING LICENSE NPF-17 - McGUIRE NUCLEAR STATION, UNIT 2

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