February 4, 1984

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. 28 to Facility Operating License NPF-9 and Amendment No. 9 to Facility Operating License NPF-17 - McGuire Nuclear Station, Units 1 and 2

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 28 to Facility Operating License NPF-9 and Amendment No. 9 to Facility Operating License NPF-17 for the McGuire Nuclear Station, Units 1 and 2. These amendments are in response to your application dated November 18, 1983, and supplemented December 5 and 8, 1983.

The amendments change the Technical Specifications related to the minimum reactor coolant system flow rate for Unit 2.

A copy of the related safety evaluation report supporting Amendment No. 28 to Facility Operating License NPF-9 and Amendment No. 9 to Facility Operating License NPF-17 is enclosed.

Sincerely,

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

Enclosures:
1. Amendment No. 28 to NPF-9
2. Amendment No. 9 to NPF-17
3. Safety Evaluation

cc w/encl:
See next page

LA:DL:LB #4	DL:LB #4	DL:LB #4
MDuncan:eb	RBirkel	EAdensam
1/16/84	1/16/84	1/16/84

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. 28
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 18 and supplemented December 5 and 8, 1983, 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. 28 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, Chief
Licensing Branch No. 4
Division of Licensing

Attachment:
Technical Specification
Changes

Date of Issuance: February 4, 1984

Delete unnecessary 4 in 5GR

Do not issue until 2/4/84

DL:LB #4	DL:LB #4	OELD	DL:LB #4	AD:DL
MDuncan:eb	RBirkel		EAdensam	TNovak
1/16/84	1/16/84	1/23/84	1/27/84	1/30/84

DUKE POWER COMPANY

DOCKET NO. 50-370

McGUIRE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 9
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 18 and supplemented December 5 and 8, 1983, 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. 9 , 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, Chief
Licensing Branch No. 4
Division of Licensing

Attachment:
Technical Specification
Changes

Date of Issuance: February 4, 1984

concurrence subject to final check by NRR with SECY on no petitions filed by end of comment period

DL:LB #4	DL:LB #4	OELD	DL:LB #4	ADL:L
MDuncan:eb	RBirkel		EAdensam	TMNovak
1/16/84	1/16/84	1/23/84	1/25/84	1/30/84

ATTACHMENT TO LICENSE AMENDMENT NO. 28

FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

AND

TO LICENSE AMENDMENT NO. 9

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 a vertical line indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Amended Page	Overleaf Page
III	IV
V	VI
2-2	2-1
2-2a (new page)	
2-5	2-6
2-9	2-10
3/4 2-9	
3/4 2-9a (new page)	
3/4 2-10	
3/4 2-10a (new page)	
B 3/4 2-4	
B 3/4 2-5	

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SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figures 2.1-1 and 2.1-2 for four and three loop operation, respectively.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

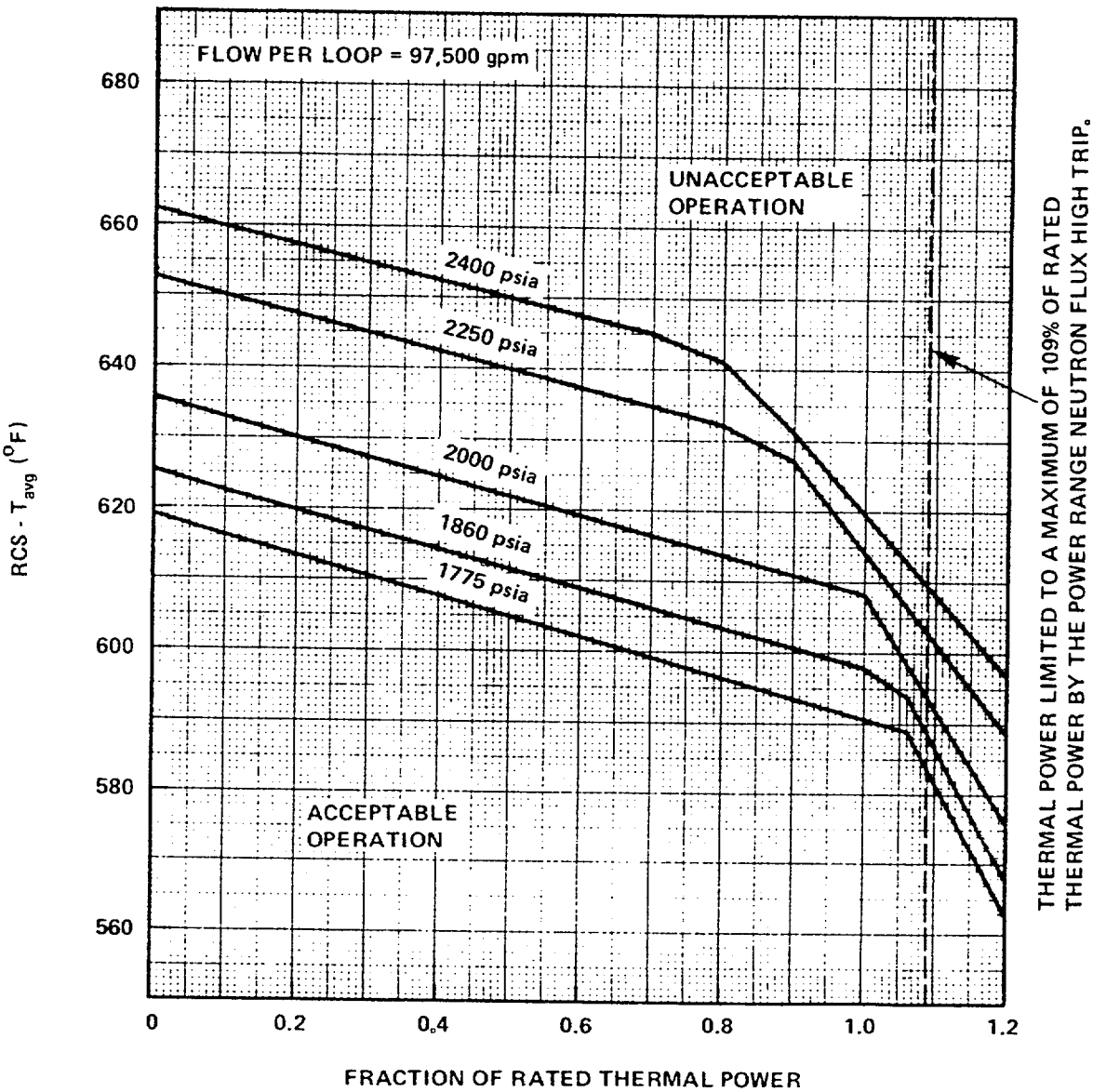

Figure 2.1-1A
REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION
UNIT 1

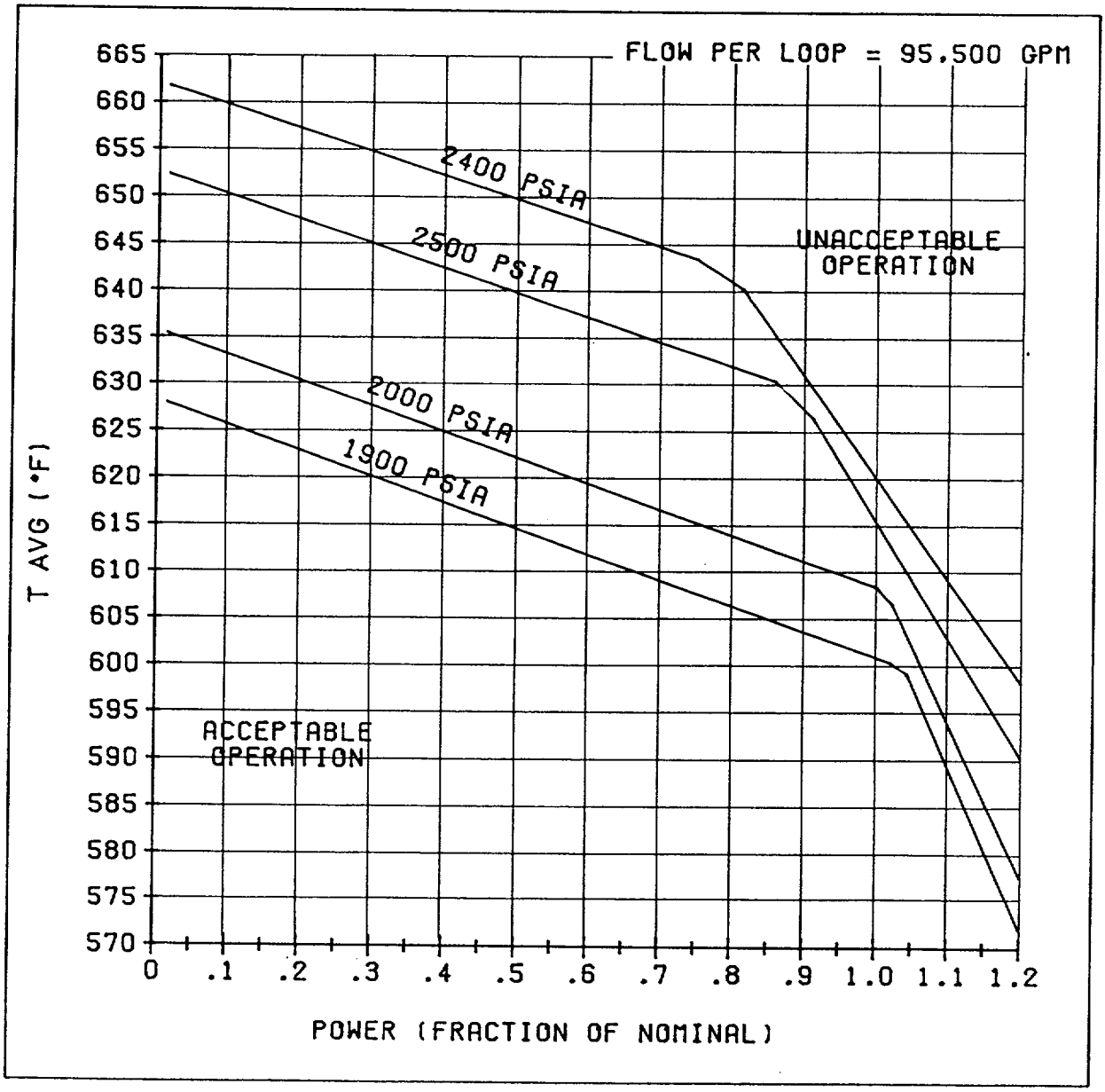

FIGURE 2.1-1b

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 - $\leq$ 25% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER
	High Setpoint - $\leq$ 109% 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 design flow per loop*

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

Amendment No. 28(Unit 1)
Amendment No. 9(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% of span at 100% of RATED THERMAL POWER.	≥ 11% of span from 0 to 30% of RATED THERMAL POWER, increasing to 53.9% 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 (Continued)

NOTE 1: (Continued)

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% and +9.5% for Unit 1 or between -36% and +8% for Unit 2, $f_1(\Delta I) = 0$, 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%, the ΔT Trip Setpoint shall be automatically reduced for Unit 1 by 0.863% or for Unit 2 by 1.173% of its value at RATED THERMAL POWER; and

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds +9.5% for Unit 1 or +8% for UNIT 2, the ΔT Trip Setpoint shall be automatically reduced for Unit 1 by 0.983% or for Unit 2 by 0.901% of its value at RATED THERMAL POWER.

NOTE 2: OVERPOWER ΔT

$$\Delta T \left(\frac{1}{1+\tau_1 S}\right) \leq \Delta T_o \{K_4 - K_5 \left(\frac{\tau_5 S}{1+\tau_5 S}\right) \left(\frac{1}{1+\tau_4 S}\right) T - K_6[T\left(\frac{1}{1+\tau_4 S}\right) - T"] - f_2(\Delta I)\}$$

Where:

ΔT = As defined in Note 1,

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

τ_1 = As defined in Note 1,

ΔT_o = As defined in Note 1,

K_4 $\leq$ 1.0908,

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

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

τ_5 = Time constant utilized in the rate-lag controller for T_{avg}, τ_5 = 5 sec,

$\frac{1}{1+\tau_4 S}$ = As defined in Note 1,

τ_4 = As defined in Note 1,

K_6 = 0.00126/°F for $T > T"$ and $K_6 = 0$ for $T \leq T"$,

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.

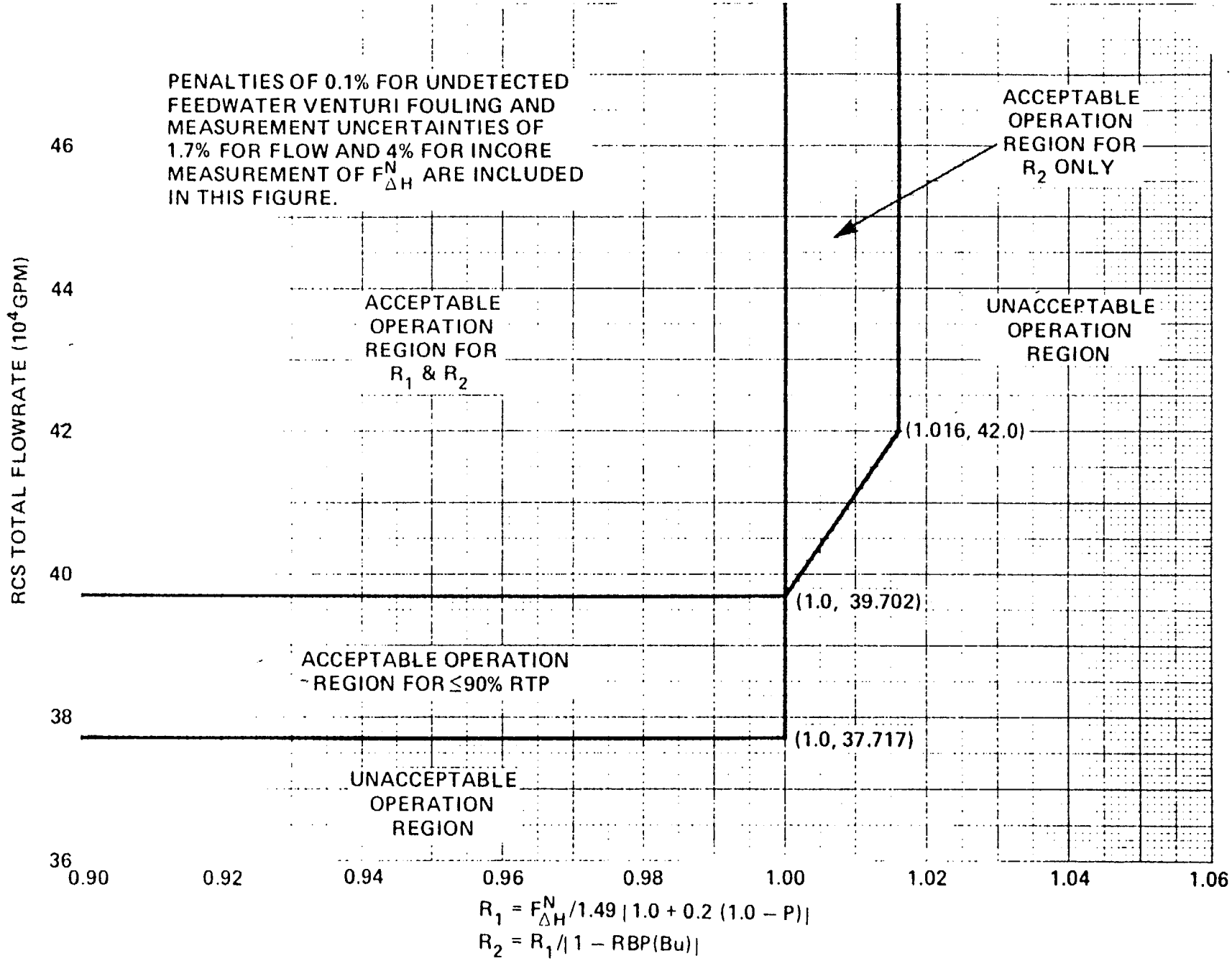

FIGURE 3.2-3A UNIT 1 RCS TOTAL FLOWRATE VERSUS R_1 AND R_2 - FOUR LOOPS IN OPERATION

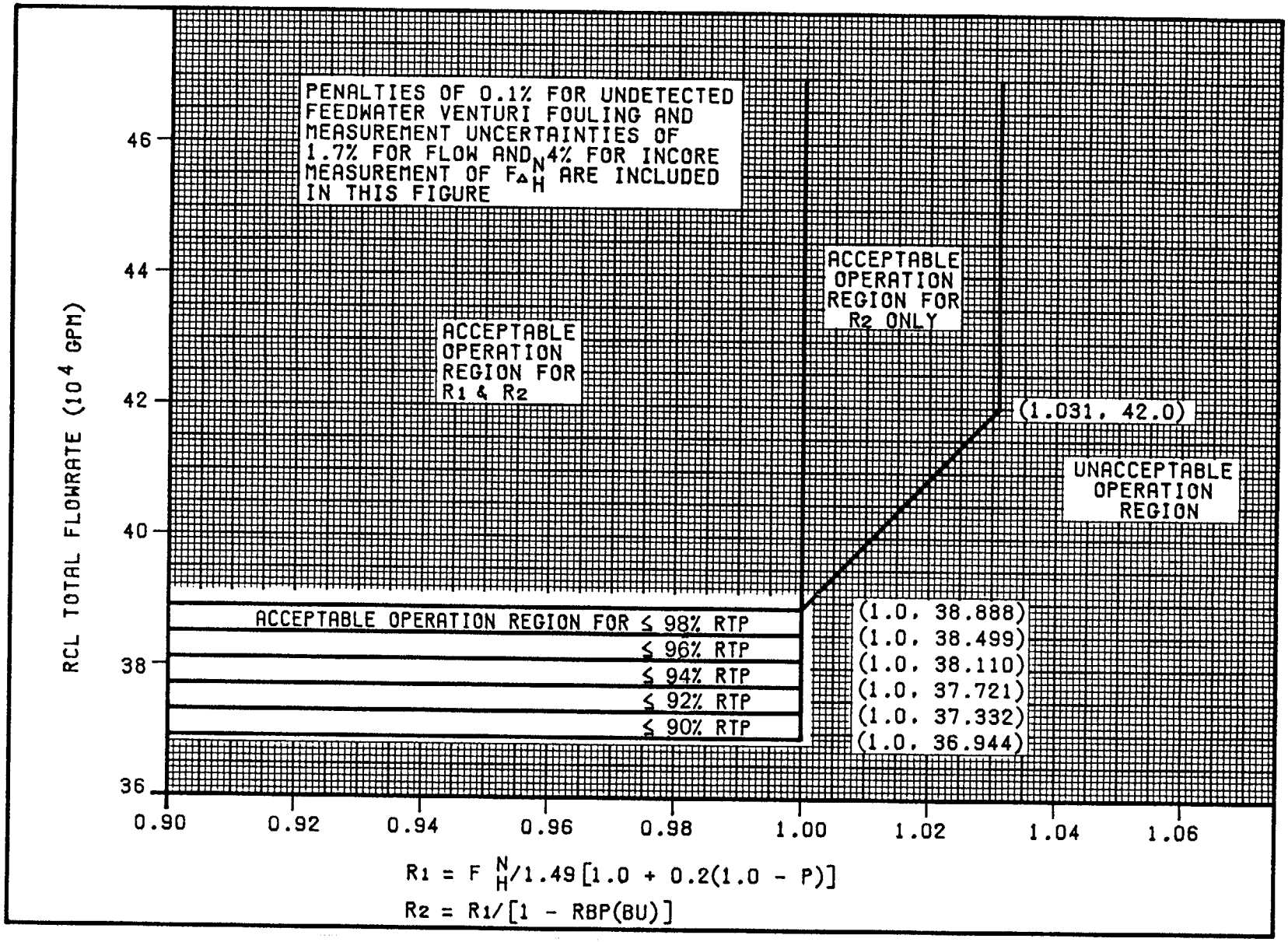

FIGURE 3.2-3b UNIT 2 RCS TOTAL FLOW RATE VERSUS R_1 AND R_2 - FOUR LOOPS IN OPERATION

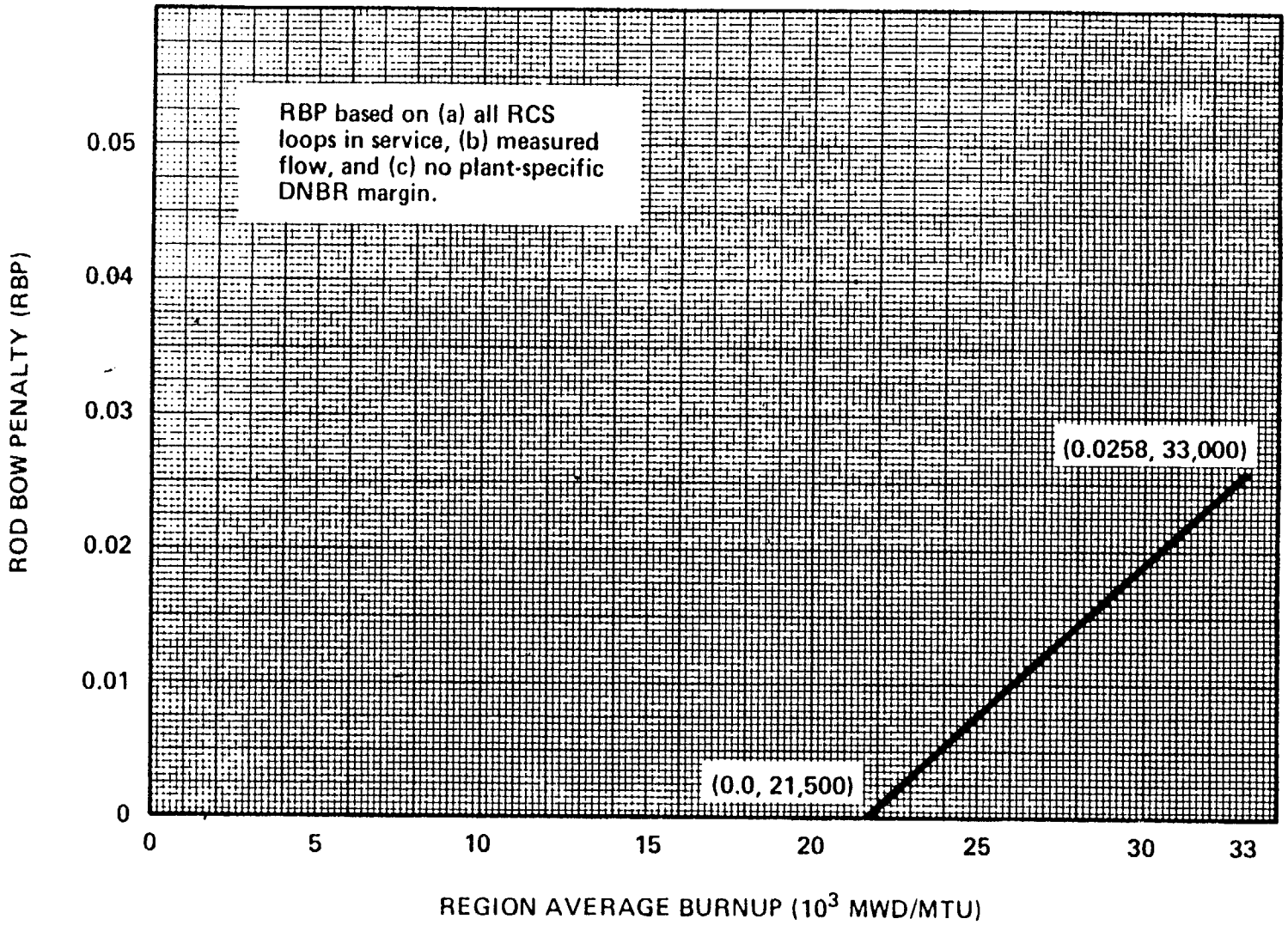

FIGURE 3.2-4A UNIT 1 ROD BOW PENALTY AS A FUNCTION OF BURNUP

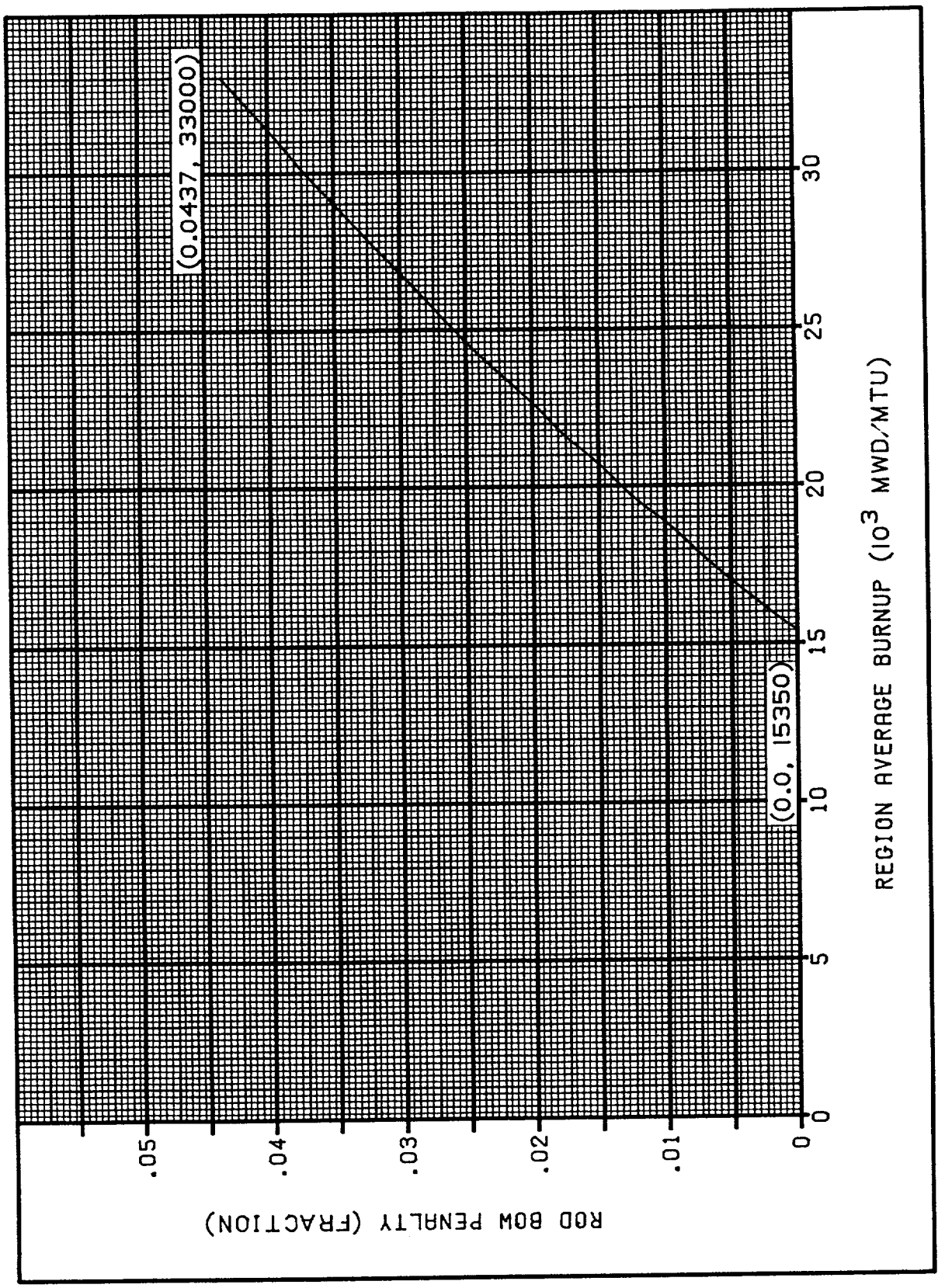

FIGURE 3.2-4b UNIT 2 ROD BOW PENALTY AS A FUNCTION OF BURNUP

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

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 Figures 3.2-3 and 3.2-4, RCS flow rate and $F^N_{\Delta H}$ 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) 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_1 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 Units 1 and 2, the margin used to partially offset rod bow penalties is 9.1 percent. 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%

For McGuire Unit 2, the margin used to partially offset rod bow penalties is 5.9 percent with the remaining 3.2 percent used to trade off against measured flow being as much as 2 percent lower than thermal design flow plus uncertainties. The penalties applied to $F^N_{\Delta H}$ to account for rod bow (Figures 3.2-4 Unit 1 and Unit 2) 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.

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

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 Figures 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.

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 limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

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 control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.0.

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

SAFETY EVALUATION REPORT

RELATED TO AMENDMENT NO. 28 TO FACILITY OPERATING LICENSE NPF-9

AND TO AMENDMENT NO. 9 TO FACILITY OPERATING LICENSE NPF-17

DUKE POWER COMPANY

INTRODUCTION

By letter dated November 18, (Ref. 1) 1983, as supplemented by letters dated December 5 (Ref. 2) and 8, (Ref. 3) 1983, the Duke Power Company (licensee) requested amendments to Appendix A of Operating Licenses NPF-9 and NPF-17. The proposed change involves Technical Specifications related to minimum reactor coolant system (RCS) flow for Unit 2. In addition, changes to the limits for safety system settings were proposed to accommodate the new RCS flow requirements. Unit 2 operation is restricted to 90% rated thermal power because the measured RCS flow does not meet the existing Technical Specification flow requirement for operation above 90%. Briefly, the proposed Technical Specification changes, primarily concerned with Unit 2, are (1) reduction by 2% of the reactor coolant system flow rate required for operation of Unit 2 at 100% power; (2) revision of the safety system setting limits to accommodate the RCS flow reduction; and (3) provision for a 2% reduction in power for each 1% reduction in the measured RCS flow below the flow requirement for 100% power.

EVALUATION

With regard to the proposed Technical Specification amendment to allow plant operation of Unit 2 at 100% power with a reduced reactor coolant system flow of 98% of the existing rated flow, the licensee has performed a safety analysis to determine the impact of the flow reduction. Since the RCS flow reduction will result in lower DNBR during the normal operation and anticipated transients, the licensee has performed a sensitivity study on the rate of change of DNBR with respect to flow reduction, as described in a December 5, 1983 letter (Ref. 2). The sensitivity study is performed with the approved THINC IV code for various state points representative of normal operating condition and transients including the DNBR limiting loss of flow event. A DNBR sensitivity factor associated with each statepoint is calculated for both the typical cell and the thimble cell. The result of the analysis shows a maximum sensitivity factor of 1.6 obtained from the typical cell. This maximum sensitivity factor is used to assess the impact of flow reduction on DNBR even though the thimble cell is DNBR limiting. Using the sensitivity factor of 1.6, the DNBR reduction would be 3.2% for a 2% flow reduction. This DNBR reduction is comparable to the staff independent calculation using the sensitivity factor from a Battelle Pacific Northwest Laboratories (PNL) study (Ref. 4).

The licensee has identified a 9.1% thermal margin due to the conservatism used in the design analysis. A breakdown of the source of this thermal margin is as follows: 1.6% from the use of the design limit DNBR of 1.3 versus the allowable limit of 1.28, 2.9% from the use of the grid spacing factor (Ks) of 0.046 versus the allowable 0.059, 1.2% from the use of the thermal diffusion coefficient of 0.038 versus the allowable 0.051, 1.7% from the use of a DNBR multiplier of 0.86 versus the allowable 0.88, and 1.7% from the 17X17 rod bundle pitch reduction. This total margin of 9.1% is lower than the margin identified previously in the McGuire SER (Ref. 5) and is, therefore, conservative. The licensee has used 6.9% of the thermal margin to compensate for the DNBR penalty due to fuel rod bowing. The remaining 3.2% margin can, therefore, be used to compensate for the 3.2% DNBR penalty resulting from 2% RCS flow reduction. Therefore, the proposed Technical Specification change to operate at 98% of the current rated RCS flow will not result in the minimum DNBR falling below the specified acceptable fuel design limit. This conclusion is applicable to DNBR limiting transients described in Chapter 15 of the Final Safety Analysis Report (FSAR).

Non-DNB-limited transients were also evaluated and the reults were determined to be within their respective limits. These transients were:

- Control Rod Withdrawal From A Subcritical Condition
- Boron Dilution
- Loss of Load
- Loss of Normal Feedwater/Station Blackout
- Steamline Break
- Rupture of a Main Feedwater Line
- Locked Rotor
- Control Rod Ejection
- LOCA Analysis

Operation under this aspect of the proposed amendments would not involve a significant increase in the consequences of accidents previously evaluated.

With regard to the change to the Unit 2 Technical Specification to allow operation below 100% but above 95% of the rated RCS flow, the licensee proposes a 2 to 1 power/flow tradeoff, i.e., the reactor power will be reduced by 2% for every 1% reduction in measured flow above 95% of the full flow. A similar amendment has previously been approved for McGuire Unit 1 where operation of 95% flow in conjunction with 90% power level is acceptable.

The licensee has provided a sensitivity study showing that for every 1% flow reduction, a 0.088% power reduction is required in order to maintain the same thermal margin to DNB.

The staff independent calculation using the PNL sensitivity study results has shown comparable results. Therefore, the proposed Technical Specification change to allow operation at reduced RCS flow with 2 to 1 power/flow tradeoff is conservative and acceptable.

In connection with the reactor operation at reduced RCS flow, the Technical Specifications regarding the reactor core safety limit and the thermal overtemperature ΔT trip setpoint are revised to ensure reactor operation within the DNB design bases and the hot leg boiling limit. The revision is done using the Westinghouse standard methodology described in WCAP-8745 (Ref. 6). The licensee has determined that the current setpoint equation for the overtemperature ΔT trip is adequate with a slight modification in the F_1 (Δ I) function. This modification is shown in Table 2.2-1 of the proposed Technical Specification revision. Since WCAP-8745 contains the standard methodology used for overtemperature trip design bases, we conclude that the revised core safety limit and overtemperature ΔT trip setpoint are acceptable.

In summary, we have reviewed the safety analysis performed by the licensee to justify the proposed Technical Specification change for operation at reduced RCS flow. The review included sensitivity studies on the impact on the flow reduction on DNB thermal margin, the DNBR limiting transients, and control rod withdrawal and ejection transients. We have found that there is enough thermal margin to compensate for the DNBR penalty resulting from 2% flow reduction, and that the proposed 2 to 1 power/flow tradeoff for operation above 95% flow is also acceptable. We have also concluded that the non-DNB-limited transient analyses in Chapter 15 of the FSAR remain valid.

ENVIRONMENTAL CONSIDERATION

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (49 FR 529) on January 4, 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 consideration 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 regulation 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.

REFERENCES:

1. Letter from H. B. Tucker (Duke Power Company) to H. R. Denton (NRC), "McGuire Nuclear Station, Docket Nos. 50-369 and 50-370, November 18, 1983.

2. Letter from H. B. Tucker (Duke Power Co.) to H. R. Denton (NRC), "McGuire Nuclear Station, Docket No. 50-370," December 5, 1983.

3. Letter from H. B. Tucker (Duke Power Co.) to H. R. Denton (NRC), "McGuire Nuclear Station, Docket Nos. 50-369 and 50-370," December 8, 1983.

4. G. M. Hesson and J. M. Cuta, "Analysis of the Sensitivity of Calculated MDNBR to Eight Selected DNB Parameters, "FATE-70-101, Battelle Pacific Northwest Laboratories, March 1979.

5. NUREG-0422, "Safety Evaluation Report Related to Operation of McGuire Nuclear Station, Units 1 and 2," March 1, 1978.

6. WCAP-8745, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," March 1977.

Principal Contributors: Ralph Birkel, Licensing Branch No. 4, DL
Y. Hsii, Core Performance Branch, DSI
E. Throm, Reactor Systems Branch, DSI

Dated: February 4, 1984

UNITED STATES NUCLEAR REGULATORY COMMISSION

DUKE POWER COMPANY

DOCKET NOS. 50-369 AND 50-370

NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES AND PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION AND OPPORTUNITY FOR HEARING

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of amendments to Facility Operating License Nos. NPF-9 and NPF-17, issued to Duke Power Company (the licensee), for operation of the McGuire Nuclear Station, Units 1 and 2, located in Mecklenburg County, North Carolina.

The amendments would (1) reduce by 2% the reactor coolant system (RCS) flow rate required for operation of McGuire Unit 2 at 100% power and revise the limits for safety systems setting to accommodate the RCS flow reduction and (2) provide for a 1% reduction in power for each 1% reduction in the measured RCS flow below the flow requirement for 100% power for McGuire Unit 2.

The operation of McGuire Unit 2 at 90% power, as a part of the unit power ascension program, has identified a low reactor coolant flow condition that, pursuant to the existing technical specification requirement, prevents the unit from operating above 90% power. The first part of the amendment which reduces reactor coolant system flow would not affect the probability of accidents previously evaluated nor create the possibility of a new or different kind of accident; however, lower RCS flow can have some effect on the consequences of accidents previously evaluated. The effects of lower RCS flow have been evaluated for the accidents discussed in the Final Safety Analysis Report (FSAR), Chapter 15. This evaluation shows that adequate thermal margin to Departure from Nucleate Boiling Ratio (DNBR) would be maintained (i.e. DNBR greater than 1.30).

Non-DNB-limited transients were also evaluated and the results were determined to be within their respective limits. Therefore, operation under this aspect of the proposed amendments would not involve a significant increase in the consequences of accidents previously evaluated. Similarly, because the evaluation showed that the original analysis results are valid for the DNB-limited transients, the safety margins inherent in the DNBR limit of 1.30 (based on the W-3 correlation) are unaffected. Also, the non-DNB-limited transients remain within their respective limits. Therefore, this aspect of the proposed amendments does not involve a significant reduction in a safety margin. This is in response to the licensee's application for amendments dated November 18, 1983.

Before issuance of the proposed license amendments, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations.

The Commission has made a proposed determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that 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 or a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The Commission has provided examples of amendments likely to involve no significant hazards considerations (48 FR 14870). One example of actions likely to involve no significant hazards considerations is an amendment which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce a safety margin, but where the results of the change are

clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan. Because the evaluation previously discussed shows that the DNB limit of 1.30 is met (Re: Standard Review Plan Section 4.4, Acceptance Criterion 1) and other design-basis transients would remain within their respective limits, the above example can be applied to this situation.

The second part of the amendment involves a requirement to reduce power by 1% for each 1% reduction in RCS flow below the minimum flow required for 100% power. Thermal-hydraulic sensitivity studies have shown that this power/flow tradeoff is conservative with respect to DNB margin. Therefore, this aspect of the amendment would not: (1) involve a significant increase in the probability or consequences of an accident, (2) create the possibility of a new or different kind of an accident, and (3) involve a significant reduction in a safety margin.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination. The Commission will not normally make a final determination unless it receives a request for a hearing.

Comments should be addressed to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D. C. 20555, ATTN: Docketing and Service Branch.

By February 3, 1984 the licensee may file a request for a hearing with respect to issuance of the amendments to the subject facility operating licenses and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written petition for leave to intervene. Request for a hearing and petitions for leave to intervene

shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of hearing or an appropriate order.

As required by 10 CFR §2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to interevene or who has been admitted as a party may amend the petition without requesting leave of the Board up to fifteen (15) days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than fifteen (15) days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought

to be litigated in the matter, and the bases for each contention set forth with reasonable specificity. Contentions shall be limited to matters within the scope of the amendment under consideration. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant

hazards consideration. The final determination will consider all public and State comments received. Should the Commission take this action, it will publish a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C., by the above date. Where petitions are filed during the last ten (10) days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at (800) 325-6000 (in Missouri (800) 342-6700). The Western Union operator should be given Datagram Identification Number 3737 and the following message addressed to Elinor G. Adensam: petitioner's name and telephone number; date petition was mailed; plant name; and publication date and page number of this FEDERAL REGISTER notice. A copy of the petition should also be sent to the Executive Legal Director, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and to Mr. Albert Carr, Duke Power Company, P.O. Box 33189, 422 South Church Street, Charlotte, North Carolina 28242, attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board designated to rule on the petition and/or request, that the petitioner has made a substantial showing of good cause for the

granting of a late petition and/or request. That determination will be based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendments which is available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28242.

Dated at Bethesda, Maryland, this 27th day of December 1983

FOR THE NUCLEAR REGULATORY COMMISSION

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

LA:DL:LB #4	DL:LB #4	OELD	DL:LB #4
MDuncan/hmc	RBirkel	R. Hawson	EAdensam
12/2/83	12/2/83	12/9/83	12/16/83

INITIAL

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

AND NOTICING ACTION

Docket No. 50-369/370 Facility: McGuire Nuclear Station, Units 1 & 2

Licensee: Duke Power Company Date of application: November 18, 1983

Request for: The proposed amendments would change the McGuire 1/2 Technical Specification requirements related to minimum reactor coolant system flow rate for Unit 2.

(See attached notice or press release for more details.)

Initial Determination:

(X) Proposed determination - amendment request involves no significant hazards considerations (NSHC).

( ) Final determination - amendment request involves significant hazards considerations (SHC).

Basis for Determination

(X) Licensee's NSHC discussion has been reviewed and is accepted. See attached amendment request.

( ) Basis for this determination is presented in the attached notice.

( ) Other (state):

(Attach additional sheets as needed.)

Initial Noticing Action: (Attach appropriate notice or input for monthly FRN)

1. ( ) Monthly FRN. Notice of opportunity for hearing (30 days) and request for comments on proposed NSHC determination - monthly FRN input is attached (Attachment 8).

2. ( ) Individual FRN (30 days). Same notice matter as above. Time does not allow waiting for next monthly FRN (Attachments 9a and 9b).

(THIS FORM SHOULD BE TYPED EXCEPT FOR UNUSUAL, URGENT CIRCUMSTANCES.)

3. ( ) Local media notice. Valid exigent circumstances exist (evaluated below). Local media notice requesting public comments on proposed NSHC determination is attached (Attachment 10).

4. ( ) No notice. A valid emergency situation exists (evaluated below) and there is no time for public notice on proposed NSHC determination. (No attachment.)

5. (X) Individual FRN (30-days). Licensee's claim of exigent or emergency circumstances is invalid (evaluated below). Notice of opportunity for hearing (30 days) and request for comments on proposed NSHC determination is attached (Attachments 9a and 9b). Letter of explanation to licensee is also attached.

6. ( ) Individual FRN (30-days). The amendment request involves SHC. Notice of opportunity for prior hearing is attached (Attachment 5). Letter to licensee also attached.

7. ( ) Individual Short FRN. Valid emergency circumstances exist (evaluated below). There is no time for the usual 30-day FRN. (Attachment 16).

Evaluation of exigent or emergency circumstances (if applicable):

See attached Sheet.

(attach additional sheets as needed)

Approvals:

		Date
1.	R.A. BIRKEL (Project Manager)	12/2/83
2.	E.G. ADENSAM (Branch Chief)	12/2/83
3.	Richard J. Clark (OELD)	12/9/83

Additional approval (for noticing actions types 3, 4, 5, 6 and 7):

4.	D M Crutchfield (Assistant Director)	12/23/83

Additional approval (for noticing action types 4 and 5):

5.	A. Schwencer (Director, Division of Licensing)	12/23/83

Attachment: as indicated

cc: Original - Docket File (with note "Docket File only")
Project Manager
Licensing Assistant
Branch Files

Evaluation of exigent or emergency circumstances

The matter of unanticipated low reactor coolant system flow was discovered two years ago, in late 1981, during the power ascension program for McGuire Unit 1. Amendment No. 9 to NPF-9 was issued on November 23, 1981, to revise the technical specification reactor coolant flow rate and nuclear enthalpy rise hot channel factor. One year later, November 1982, the licensee requested a second technical specification change, again to enable Unit 1 to operate at 100% rated thermal power. With the granting of Amendment No. 22 to NPF-9, on June 28, 1983, Unit 1 was permitted to operate at 100% power. This same amendment applied to Unit 2 because both units utilize a common set of Technical Specifications.

During our evaluation of the McGuire thermal design flow inadequacies it became obvious that the ability of the McGuire units to maintain thermal design flow has little (perhaps 1-2%) or no margin. This is supported by the licensee's most recent request of November 18, 1983, for a third technical specification change related to reactor flow. The measured reactor coolant system flow in Unit 2 is further reduced and requires further revision to the McGuire technical specifications. The technical merits of the most recent request are being evaluated by the staff.

Recognizing the history of this matter over the past two years, we conclude that the licensee should reasonably have anticipated the events leading to what Duke claims to be a derated condition. We have discussed this matter with OELD and have their concurrence in this denial.

OELD agrees that the basis given for denial of emergency treatment is reasonable.