June 28, 1983

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. 22 to Facility Operating License NPF-9 and Amendment No. 3 to Facility Operating License NPF-17 - McGuire Nuclear Station, Units 1 and 2

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 22 to Facility Operating License NPF-9 and Amendment No. 3 to Facility Operating License NPF-17 for the McGuire Nuclear Station, Units 1 and 2. These amendments are in response to your letter dated November 23, 1982, and supplemented March 14, March 28, April 26, and April 27, 1983.

The amendments change the Technical Specifications to reduce the measurement uncertainty for total Reactor Coolant System flow rate.

A copy of the related safety evaluation report supporting Amendment No. 22 to Facility Operating License NPF-9 and Amendment No. 3 to Facility Operating License NPF-17 is enclosed.

Sincerely,

/S/

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

Enclosures:
1. Amendment No. 22 to NPF-9
2. Amendment No. 3 to NPF-17
3. Safety Evaluation

cc w/encl:
See next page

OFFICE	LA:DL:LB #4	DL:LB #4	DL:LB #4				
SURNAME	MDuncan/hmc	RBirkel	EAdensam				
DATE	6/16/83	6/16/83	6/28/83				

June 28, 1983

AMENDMENT NO. 22 TO FACILITY OPERATING LICENSE NPF-9 - McGUIRE NUCLEAR STATION, UNIT 1
AMENDMENT NO. 3 TO FACILITY OPERATING LICENSE NPF-17 - McGUIRE NUCLEAR STATION, UNIT 2

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OFFICE							
SURNAME							
DATE							

DUKE POWER COMPANY

DOCKET NO. 50-369

MCGUIRE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 22
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 23, 1983, and supplemented March 14, March 28, April 26, and April 27, 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:

OFFICE ▸							
SURNAME ▸							
DATE ▸							

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 22 , 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

/S/

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

Attachment:
Technical Specification
Changes

Date of Issuance: June 28, 1983

No legal objection (see correction p. 8 of SER) (Provided No Comments or Hearing requests Have been Received before issuance) 6/28/83

SER has wrong conclusion 6/27/83

Must be fixed before it is OK

No legal objection to form of amendment

OFFICE	LA:DL:LB #4	DL:LB #4	OELD	DL:LB #4	AD:L:DL		
SURNAME	MDuncan/hmc	RBirkel	LUTCHIN	EAdensam	TNovak		
DATE	6/6/83	6/6/83	6/24/83	6/13/83	6/1/83		

DUKE POWER COMPANY

DOCKET NO. 50-370

MCGUIRE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 3
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 23, 1982, and supplemented March 14, March 28, April 26, and April 27, 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:

OFFICE ▶							
SURNAME ▶							
DATE ▶							

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 3 , 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

/S/

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

Attachment:
Technical Specification
Changes

Date of Issuance: June 28, 1983

No legal objection to form of amendment

OFFICE	LA:DL:LB #4	DL:LB #4	OELD	DL:LB #4	AD:L:DL		
SURNAME	MDuncan/hmc	RBirkel	Cutchin	EAdensam	TNovak		
DATE	6/6/83	6/6/83	6/1?/83	6/16/83	6/1?/83		

ATTACHMENT TO LICENSE AMENDMENT NO. 22

FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

AND

TO LICENSE AMENDMENT NO. 3

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
3/4 2-8	3/4 2-7
3/4 2-9	3/4 2-10
3/4 2-11	3/4 2-12
B 3/4 2-4	B 3/4 2-3
B 3/4 2-5	
B 3/4 2-6	

OFFICE ▶							
SURNAME ▶							
DATE ▶							

SURVEILLANCE REQUIREMENTS (Continued)

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

e. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided for all core planes containing Bank "D" control rods and all unrodded core planes in a Radial Peaking Factor Limit Report per Specification 6.9.1.12,

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

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

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

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

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

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

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

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_1, R_2 shall be maintained within the region of allowable operation shown on Figure 3.2-3 for four loop operation:

Where:

a. $R_1 = \dfrac{F^N_{\Delta H}}{1.49\ [1.0 + 0.2\ (1.0 - P)]}$,

b. $R_2 = \dfrac{R_1}{[1\text{-}RBP(BU)]}$,

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

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

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

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and R_1, R_2 outside the region of acceptable operation shown on Figure 3.2-3:

a. Within 2 hours either:

1. Restore the combination of RCS total flow rate and R_1, R_2 to within the above limits, or

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

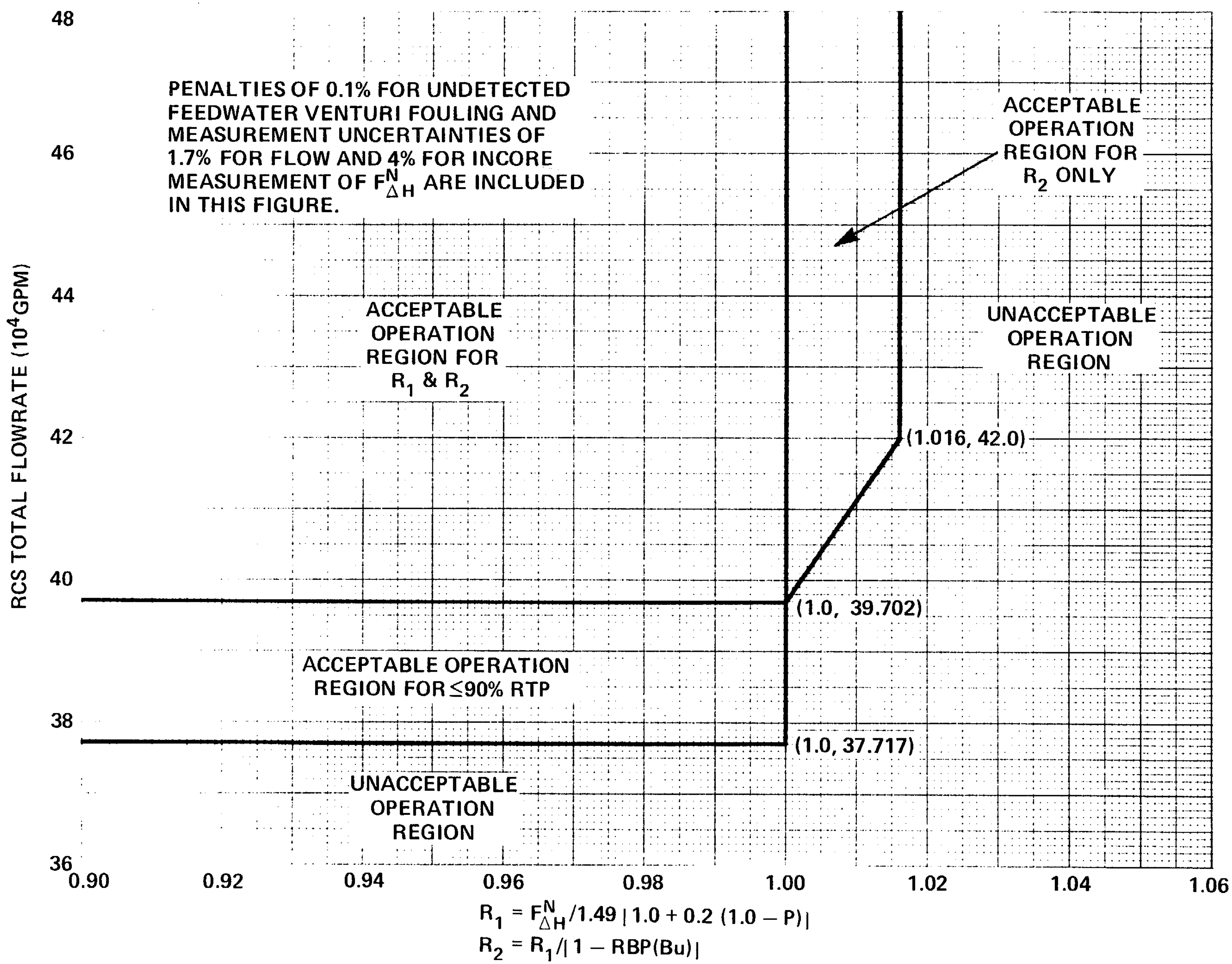

FIGURE 3.2-3 RCS TOTAL FLOWRATE VERSUS R_1 AND R_2 - FOUR LOOPS IN OPERATION

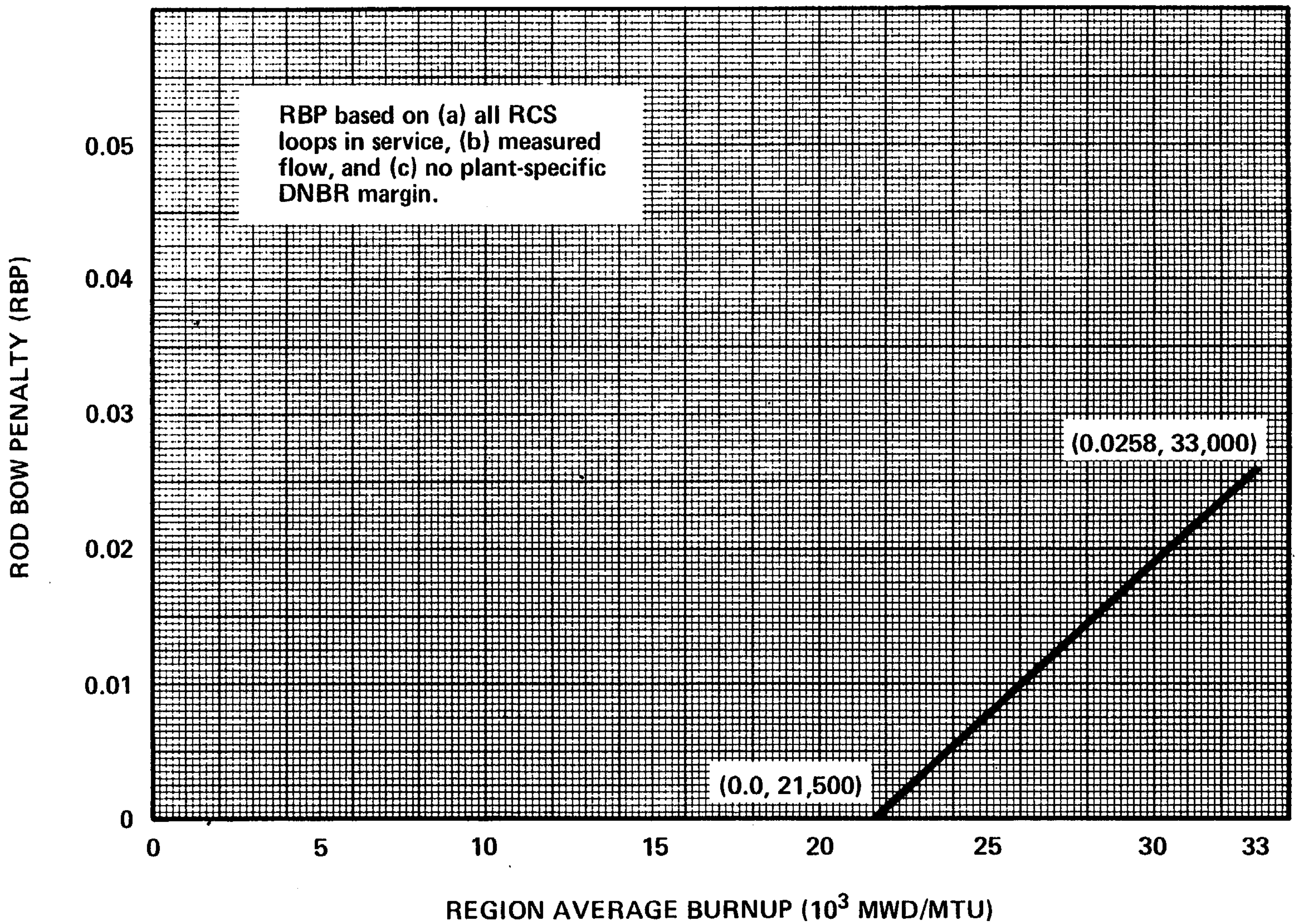

FIGURE 3.2-4 ROD BOW PENALTY AS A FUNCTION OF BURNUP

ACTION: (Continued)

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

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

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

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate determined by process computer readings or digital voltmeter measurement and R_1 and R_2 shall be within the region of acceptable operation of Figure 3.2-3:

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

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

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

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

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

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

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

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09:

  1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

     a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

     b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

  2. Within 2 hours either:

     a) Reduce the QUADRANT POWER TILT RATIO to within its limit, or

     b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

  3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours 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; and

  4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL power may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

*See Special Test Exception 3.10.2.

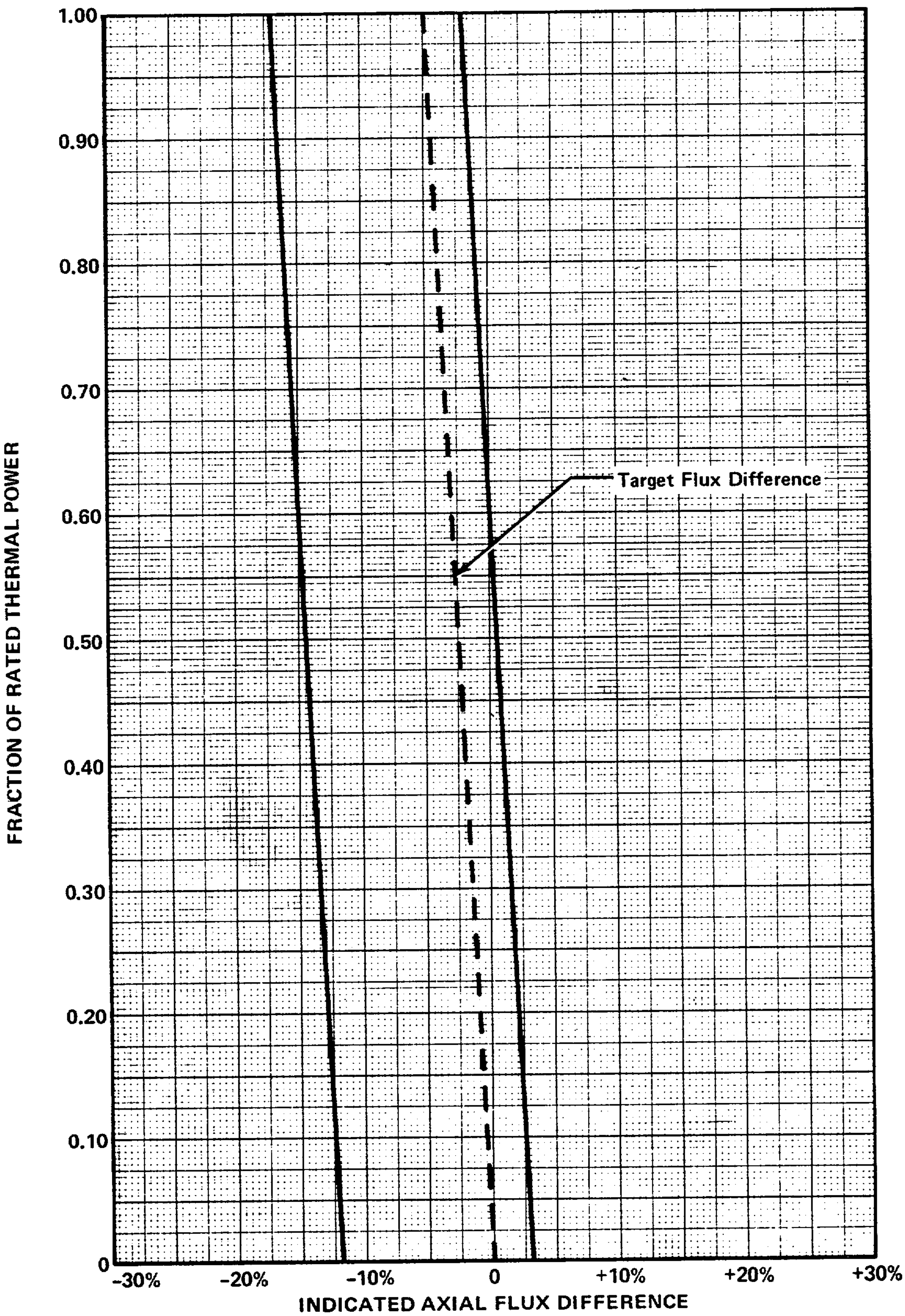

FIGURE B 3/4 2-1.
TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

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 design margin which total 9.1% when the analysis is performed with the approved interim methods. The penalties applied to $F^N_{\Delta H}$ to account for rod bow (Figure 3.2-4) as a function of burnup are consistent with those described in Mr. John F. Stolz's (NRC) letter to T. M. Anderson (Westinghouse) dated April 5, 1979, and W 8691, Rev. 1 (partial rod bow test data).

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.

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

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.

QUADRANT POWER TILT RATIO (Continued)

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 minimum DNBR of 1.30 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.

SAFETY EVALUATION REPORT
RELATED TO AMENDMENT NO. 22 TO FACILITY OPERATING LICENSE NPF-9
AND TO AMENDMENT NO. 3 TO FACILITY OPERATING LICENSE NPF-17
DUKE POWER COMPANY

INTRODUCTION

By letter dated November 23, 1982, (Ref. 1) Duke Power Company (the licensee) requested an amendment to the Technical Specifications, Appendix A of Operating License No. NPF-9 for the McGuire Nuclear Station Unit 1. The amendment would revise the uncertainty in the measurement of the total reactor coolant system (RCS) flow rate from 3.5% to 1.7% plus 0.1% as stated in LCO 3.2.3.d, and as reflected in the minimum specified measured flow rate for acceptable operation identified in Figure 3.2 of the plant technical specifications. Further, the surveillance requirements of Section 4.2.3.2 would be revised to indicate that the process computer readings or digital voltmeter measurements be used to confirm acceptable operation, and Section 4.2.3.5 would be revised to indicate that the RCS flow rate shall be determined by precision heat balance at least once per 18 months.

BACKGROUND AND DISCUSSION

A Technical Specification change requested by Duke Power Company by letter dated November 11, 1981, permitted operation of McGuire Unit 1 at RCS flow rates which are less than the design flow used in the safety analysis for power levels of 90% or less. At that time, based on preliminary flow measurements, it became apparent that they might not be able to achieve the thermal design flow limits defined in Technical Specification 3/4.2.3. Since the Westinghouse Standard Technical Specifications and the original McGuire Technical Specification require that the power level be reduced to 5% or less if the design flow used in the safety analyses is not satisfied, Duke requested that the Technical Specification minimum flow rate as a function of nuclear enthalpy rise factor be modified to permit operation at a reduced flow rate (95% of the original TS value) in conjunction with a reduced power level (90% of the licensed value). This change was approved and justified based on a staff study using W-3 correlation sensitivity factors showing that a 5% reduction in flow requires a 4.32% reduction in power. Since this sensitivity factor varies as a function of several parameters and since no safety analyses have been performed with initial flow lower than the original design flow, the licensee properly included some conservatism in its proposed power reduction as a function of flow rate resulting in the 90% power limit.

Specification 4.2.3.2 requires that RCS flow rate and nuclear enthalpy rise hot channel factors be determined to be within the acceptable operating region of Figure 3.2 prior to operation above 75% of rated thermal power after each fuel loading and at least once per 31 effective full power days thereafter. Presently the RCS flow rate is determined by elbow tap flow measurements in each of the four primary coolant loops. The uncertainty in this method of determining the total RCS flow rate is presently specified as 3.5%. The licensee has proposed a change to the plant technical specifications whereby the RCS flow rate would be periodically determined by a heat balance across the steam generators. The RCS flow rate determined in this manner would be used to calibrate the readings of RCS flow rate as measured by the elbow tap flow measurements for subsequent confirmation of an acceptable operating region defined by LCO 3.2.3. As justification for this change the licensee provided an analysis to demonstrate that the uncertainty in measurement of the total RCS flow rate by the heat balance method does not exceed 1.7%.

EVALUATION

The licensees submittal of November 23, 1982 provided an analysis of the uncertainties in determining total RCS flow rate using a heat balance across the steam generator in each loop. A number of factors were considered which would contribute to the uncertainty in the determination of the total RCS flow rate. For each factor, its effect on RCS flow was stated. Using the square root of the sum of the squares (RSS) method, the individual factors were combined to determine the overall uncertainty in the RCS flow for each loop. A condition for the validity of the RSS method is that each of the factors is independent.

These conditions are also applied to the determination of the uncertainty in the total RCS flow rate. Since there are four loop flow rates which are summed to obtain the total RCS flow rate, the uncertainty of the individual flow rate is reduced by one over the square root of four to obtain the uncertainty in the total RCS flow rate. Therefore, particular attention was given in this review to the condition that each of the factors considered in the uncertainty analysis is independent. Since the overall uncertainty in total RCS flow rate is stated as 1.7%, an uncertainty value of 0.05% was used as a lower bound in assessing the significance of any individual factor considered in the uncertainty analyses. The 0.05% value for the bound of significance is arrived at as follows: If in the RSS analysis that produced the result of 1.7% an additional term with a value of 0.05% is considered, this would increase the result by an increment of only 0.001%, i.e., to the value of 1.701%. Therefore, it is concluded that a conservative basis exists for the use of 0.05% as a bound of significance, and that this would be true even if additional terms of this magnitude or less were neglected. As an example, feedwater pressure is a measured parameter which is used to determine both feedwater flow and feedwater enthalpy for each loop. Although the analysis treats the uncertainty in feedwater flow and enthalpy as being independent, both are dependent upon the uncertainty in the measurement of feedwater pressure. Since the uncertainties in feedwater pressure expressed in percent of RCS flow for both feedwater flow and enthalpy add up to less than 0.05%, it is concluded that this interdependency has a negligible effect on the loop RCS flow uncertainty.

A lack of independence may occur for factors which are commmon to all coolant loops. Since the uncertainty in total RCS flow is one half that of individual flow measurements, an uncertainty value of 0.025% was used as a bound in assessing the significance of any dependent factor which could impact total RCS flow. The basis for this conclusion is arrived at in a manner similar to that cited above except the value of the bound of significance on loop dependent parameters is one half that used for dependent parameters within a loop. As an example, feedwater flow is dependent upon the measurement of differential pressure across the feedwater flow nozzles in each loop. However, the same differential pressure measuring instrument is used to determine feedwater flow in each loop. Thus, the uncertainty in RCS flow rate is not independent between loops. However, since the uncertainty in RCS flow rate for each loop due to this measurement is less than 0.025%, it is concluded that this interdependency has a negligible effect on total RCS flow uncertainty.

For the secondary side of the heat balance the determination of three major parameters is required. They are feedwater flow, feedwater enthalpy and main steam enthalpy. The plant has feedwater flow nozzles which were installed in a section of feedwater pipe, and as a unit, flow calibrated in a laboratory. The licensee provided data (Ref. 2) to substantiate the uncertainty in the flow coefficient based on the accuracy of the flow calibration at Alden Research Laboratories. The staff concludes that this information provides adequate justification for the uncertainty associated with this factor of the uncertainty in feedwater flow. However, the staff did question the validity of the assumption that the characteristics of the flow nozzles do not change over the life of the plant. Of particular concern, was the potential that fouling of the feedwater flow nozzles could result in a bias which would result in an increase in calculated RCS flow for each loop as well as the total RCS flow rate. The licensee, therefore, revised this analysis (Ref. 3) to include consideration of an uncertainty of 0.1% due to feedwater flow nozzle fouling. The basis for this number is that monitoring and trending of various plant performance parameters are expected to reveal fouling of this magnitude. The staff does not have sufficient information to confirm that a bias of 0.1% can be detected by this means. However, the staff judges that the program being used by Duke Power Company for maintaining water chemistry has, from experience, been demonstrated to be excellent; and significant fouling, if it should occur at all, would not be expected to occur for many years. The staff expects that the licensee will maintain appropriate records at the plant site of its monitoring and trend analysis for this effect that can be audited by the NRC.

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The staff questioned the uncertainty in the measurement of the differential pressure across the feedwater flow nozzles since it was a very low number. The licensees provided specification data (Ref.4) to support the accuracy of this instrument. Based on this data and the fact that the RCS flow uncertainty due to this measurement is more than an order of magnitude below the lower bound of significance noted above, we find the accuracy value of the differential pressure measurement instrument used for feedwater flow to be acceptable.

The two remaining measured parameters for determining feedwater flow are feedwater temperature and pressure. Feedwater temperature is used to determine both feedwater flow and enthalpy. Since the uncertainty on loop RCS flow due to temperature exceeds the lower bounds of significance, the licensee's revised analysis (Ref.3) accounts for the fact that temperature uncertainty dependence exists in loop RCS flow uncertainty. The uncertainty values for feedwater flow and enthalpy due to temperature are summed to account for this dependence on RCS loop flow. The same readout instrument is used to measure feedwater temperature in each loop. This interdependence has not been included in the analysis, however since the readout instrument uncertainty does not significantly exceed the lower bounds of significance, this interdependence may be neglected. The dependence of feedwater pressure as a single measurement which affects the uncertainty in feedwater flow and enthalpy was addressed in the example related to lower bounds of significance noted previously. With regard to the magnitude of the uncertainty in the measurement of feedwater pressure and temperature, the staff concludes that reasonable values are used.

On April 25, 1983, the staff met with the licensee to discuss the uncertainty analysis and to review data taken from a heat balance to determine RCS flow. At this time it was noted that a portion of the feedwater to each steam generator enters the upper steam generator nozzle. This flow does not pass through the main feedwater nozzles and is measured separately. In that this flow is about 2 to 3 percent of the total feed to the steam generator it is significant to the heat balance and uncertainty analysis. The licensee's revised analysis (Ref. 3) includes the uncertainty in RCS loop flow based on a 5% uncertainty in measuring this auxiliary flow rate. Also, steam generator blowdown flows were included with the same accuracy of measurement. The staff finds these values reasonable.

The enthalpy of steam at the outlet of the steam generator is determined based on the measurement of pressure. Since the uncertainty in RCS flow due to the steam pressure measurement is three orders of magnitude less than the lower bounds of significance, its uncertainty is not of consequence. Consideration was given to moisture carry-over to the main steam lines as it would have an effect on the calculated thermal output of the steam generator. Based on a comparison of the value of the uncertainty in moisture carry-over to the design value of moisture carry-over for these steam generators, the staff concludes that a conservative treatment of this factor was used in this analysis.

The primary side heat balance includes consideration of thermal input to the RCS from the reactor coolant pumps. This value is calculated by the process computer based on measurements of reactor coolant pump voltage and current which are provided as hard wired inputs to the computer. The uncertainty analysis includes a single value of the component error and RCS flow uncertainty for reactor coolant pump thermal power. This uncertainty in RCS flow is approximately at the lower bounds of significance and is considered a reasonable value.

The heat losses from primary and secondary piping are estimated values. The uncertainty in this factor is conservatively taken at one half of its value. This effect on RCS flow uncertainty is approximately at the lower bounds of significance and is considered a reasonable value. The effect of charging and letdown to and from the primary coolant loops was included in the revised uncertainty analysis (Ref. 3). The uncertainty in measuring the flows and charging temperature appear to be reasonable values. Since their effect on RCS flow uncertainty is approximately at the lower bounds of significance, they are considered reasonable values.

The final consideration in the primary side is the determination of the hot and cold leg enthalpy. The values are based on measurments of hot and cold leg temperature and pressurizer pressure.

The temperatures of the hot and cold legs are determined by direct resistance readings of the RTD's located in the bypass manifolds that receive a sample of the reactor coolant flow. While the pressure of the reactor coolant sample at the hot and cold leg manifolds should be approximately equal, the staff estimates that the pressure at the manifolds could be lower than the pressurizer pressure by approximately 20 psi. Since this single pressure measurement is used to calculate the RCS flow in each loop, the error in total RCS flow rate would be of the same magnitude. The staff estimate of this error is about 6 times the lower bounds of significance for interdependent factors affecting total RCS flow. As a consequence the licensee has stated that he will correct the pressurizer pressure measurement for static and dynamic head when determining hot and cold leg enthalpy at the RTD manifolds.

The staff noted the fact that the single measurement of pressurizer pressure is a dependent variable when assessing the uncertainty on hot and cold leg enthalpy. The revised analysis treats uncertainties in hot and cold leg enthalpy due to pressure uncertainties as interdependent. Further since the original analysis considered the uncertainty in RCS flow as a single factor due to pressure, the staff investigated the sensitivity of flow uncertainty due to a change in hot and cold leg enthalpy and found that one is about 4 to 5 times more sensitive to a small change in pressure than the other. In the revised analysis the sensitivity of RCS flow uncertainty per psi change in pressure was stated to be equal for both hot and cold leg enthalpy. When questioned on this matter the licensee stated that the value given reflects the change in flow uncertainty when both hot and cold leg enthalpy vary by the same amount. However, the analysis assigns this value to both hot and cold leg enthalpy. Thus, one would conclude that twice the required uncertainty was used to determine pressure effects on hot and cold leg enthalpy for loop RCS flow. However, the same pressure measurement is used for all loops and a dependence exists which was not included in the analysis. Since the total RCS flow uncertainty is one half of loop flow uncertainty, the conservatism of a factor of 2 in loop flow uncertainty is eliminated when the pressure dependence on total RCS flow uncertainty is accounted for. Finally the revised analysis stated a value for the pressure measurement uncertainty which was about 50% greater than the original analysis. When questioned on this point the licensee stated that this calculation was based on a full range pressure channel rather than the narrower span pressurizer pressure measurement channels used by the protection system. The licensee noted that the original accuracy for pressure measurement should be used. Further since there are four of these channels, a minimum of three should always be available. The revised analysis uses the average of three channels, with the uncertainty in this average being reduced by a factor of one over the square root of three. While the staff questions the validity of this approach, due to the potential for biased errors, the effect borders on the lower bounds of significance.

The temperatures of the hot and cold legs are determined by resistance measurements of two RTD's installed in each manifold. The staff questioned the basis for the uncertainty in the resistance measurement as provided by the licensee (Ref. 4) which was expressed in terms of the measurement span of instruments not used in this measurement. In the revised analysis (Ref.3), the licensee provided the basis for the uncertainty in resistance measurements based on the specifications of the equipment used to make this measurement. The instrument accuracy is stated as ± 0.075 ohms. Although data provided by the licensee on a previous heat balance indicates that resistance readings were only recorded to the nearest 0.1 ohms, the licensee has stated that subsequent reading will be recorded to the nearest 0.01 ohm. The staff concludes that the added precision in resistance readings is appropriate in light of the instruments specified accuracy and readout capability. The same readout instrument is used to measure the resistance of all RTD's. This dependence was not accounted for in the analysis, however the difference in RCS flow uncertainty for hot and cold leg temperature measurements is about at the lower bounds of significance. Thus the staff finds that this interdependence may be neglected.

The sample of the primary coolant for the hot leg RTD manifold is obtained by three probes mounted in the 29 inch diameter hot leg pipe. The probes sample the reactor coolant to a depth of seven inches to obtain a sample which is representative of the average hot leg temperature. Test data from other Westinghouse plants, where temperature gradients were measured across the hot leg, were used with conservative values of sample probe flows as a basis to establish a temperature uncertainty due to this measurement configuration. Based on the discussion of the considerations included on temperature streaming and its effect on hot leg temperature measurement, the staff concludes that a reasonable uncertainty for temperature streaming effects has been included in the analysis. It should be noted that this effect is the dominant consideration in the uncertainty analysis and accounts for about two thirds of the uncertainty in total RCS flow rate.

Measurements in the primary and secondary loops required for the heat balance are performed by taking 12 readings of each parameter over a period of about one hour. For feedwater flow, 36 readings are taken. The calculation of RCS flow is based on the average value of each measured parameter over the one hour period. On examination of the data from a previous heat balance calculation, the staff noted that feedwater flow is one variable for which variations as large as a 10 percent change in measured feedwater flow occurred. As a means to assess the uncertainty in RCS flow due to these variations, the licensee calculated the standard deviation in each set of the 36 readings of feedwater flow. From this, the average of all sets was obtained and divided by the square root of 36 to give a standard deviation of averages. The uncertainty due to data scatter was then assessed, at a 95% confidence level, to be twice the standard deviation of averages. Similarly the uncertainty in RCS flow was determined for other measured parameters. Since the RSS method was used to combine uncertainties, the effect on loop RCS flow uncertainty was calculated by the staff to add only 0.07% uncertainty to the loop flow uncertainty of 3.02%. The staff concludes that the treatment of fluctuations in process parameters in this manner do not significantly impact the final result.

On the basis of the determination of loop RCS flow by the heat balance calculations, the elbow tap flow measurements for the primary coolant system are calibrated to provide subsequent measurements of RCS flow as required by plant technical specifications. Additional uncertainty is factored into the total RCS flow rate obtained by the heat balance calculation to account for uncertainties associated with the elbow tap flow measurements. Since these instruments provide signals to the protection system, uncertainties in these instruments have previously been analyzed by Westinghouse in establishing the basis for protection system set points. However, in the assessment of RCS flow uncertainty, certain factors have been excluded based on conditions under which the plant is operating when the normalization process takes place, in contrast to conditions existing when considered in the set point methodology analysis. In addition credit is taken for a set of components as a group which are calibrated more frequently than assumed in set point analysis. The net effect is that each elbow flow measurement channel has an uncertainty of about 1%. Assuming that two of the three elbow flow measurement channels will always be available, the uncertainty with two measurements is taken as one over the square root of two of that for one channel. Finally it was assumed that each loop measurement is independent. This results in a further reduction in total RCS flow uncertainty by a factor of one over the square root of four. Where each of the various factors which contributes to the uncertainty of the elbow flow measurement channels are truly independent and random, this is a valid process for evaluating uncertainties. However, where bias errors may be introduced due to dependent conditions, this approach is non-conservative. Examples would be process conditions of temperature and pressure, environmental conditions, the same test equipment used to calibrate components common to all channels, and the like. Because a strong case has not been made that these factors are truly independent, the staff has considered that a value of about 0.7%, being about halfway between the accuracy value of one channel and that of eight independent channels, would be a more appropriate basis for consideration. However, this value combined with a total heat balance uncertainty of about 1.5% using RSS is still within the proposed technical specification limit of 1.7% total RCS flow uncertainty.

In addition, for all anticipated operational occurrences (AOO), the minimum DNBR has been determined to be no less than 1.4 compared to the DNBR limit of 1.3 for the W-3 correlation. Sensitivity studies have shown that the effect of RC flow on DNBR is about 1.5% DNBR per 1% of RC flow reduction. Therefore, the reduction in DNBR margin due to reduced flow will be less than 3%, which is not sufficient to cause the DNBR limit of 1.3 to be violated even if the initial flow assumed for analyzed AOOs is reduced by 2%. Likewise, a 2% reduction in initial flow assumed for accident transients would not result in violation of design or licensing limits. Therefore, we conclude that the proposed McGuire Technical Specification change reducing flow measurement uncertainity from 3.5% to 1.8% will have no significant safety concern.

ENVIRONMENTAL CONSIDERATION:

We have determined that the amendment does not authorized a change in effluent types or total amounts. Furthermore, we have determined that this amendment will permit operation at the full power level reviewed and evaluated in our Final Environmental Statement (NUREG-0063) dated April 1976. Therefore, we have determined that this amendment will not result in any significant environmental impact different from that previously evaluated. Having made this determination, we have further concluded that the amendment involves 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 this amendment.

FINAL NO SIGNIFICANT HAZARDS CONSIDERATION (SHC) DETERMINATION

The Commission made a proposed determination that the amendment involves no SHC which was published in the Federal Register (48 FR 2717) on June 13, 1983, and consulted with the State of North Carolina. No public comments were received and the State of North Carolina did not have any comments. Based on the Commission's final review and the absence of State and Public comments, the Commission has made a final determination that the amendment involves no significant hazards consideration.

CONCLUSION

The staff has reviewed the reduction in flow uncertainty now proposed by the licensee and has concluded that changing the flow uncertainty to 1.7% plus 0.1% as stated in LCO 3.2.3.d from the previous value of 3.5% has been adequately justified on the basis of physical factors (all volatile chemistry treatment of reactor coolant and accuracy of instrument components) and statistical methods used in the computation of overall measurement uncertainty.

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The most dominant factor in the uncertainty in the measurement of RCS flow by the heat balance method is the uncertainty associated with the hot leg temperature streaming error. Likewise any error in its assumed value would have a larger impact on the results of this analysis. Based on the information presented, the staff concludes that an acceptable case has been made to address this uncertainty.

The conditions under which the RSS method of combining uncertainties is valid have been closely examined. In that the total RCS flow rate uncertainty is assessed to be one half of that of the loop flow uncertainty, this is a very significant consideration. In some cases where dependency exists between factors affecting either or both RCS loop flow and total RCS flow, the effects have been judged to be non-significant. The licensee revised his analysis to address dependent factors where we considered it appropriate. Therefore, we conclude that interdependencies which impact the validity of the RSS method have been either appropriately considered in the analysis or, based on staff judgement, do not impact the proposed technical specification limiting uncertainty.

The revised analysis included the effects of fouling of the feedwater flow nozzles. The licensees basis for including 0.1 percent uncertainty in total RCS flow rate for this effect was based on the ability to detect changes of this magnitude by a program that trends changes in plant measurements. Since the staff cannot confirm the capability of the trending program to reveal changes of this magnitude, it expects that the licensee will maintain appropriate records at the plant site of its monitoring and trend analysis for this effect that can be audited by the NRC. The staff concludes, however, that the use of the additional 0.1% uncertainty due to potential feedwater venturi fouling is acceptable until more conclusive data are available.

In conclusion the staff finds the proposed changes to the plant technical specifications based on a 1.7% uncertainty in total RCS flow rate determined by the primary side elbow taps which are calibrated periodically by precision heat balance to be acceptable and based on the considerations discussed above, that: (1) this amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously and does not involve a significant reduction in a margin of safety. On this basis the staff concludes that this amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

REFERENCES

(1) Duke Power Company Letter from Hal B. Tucker to Harold R. Denton, Director, NRR, NRC dated November 23, 1982.

(2) Duke Power Company Letter from Hal B. Tucker to Harold R. Denton Director, NRR, NRC dated March 14, 1983.

(3) Duke Power Company Letter from Hal B. Tucker to Harold R. Denton Director, NRR, NRC dated April 27, 1983.

(4) Duke Power Company Letter from Hal B. Tucker to Harold R. Denton Director, NRR, NRC dated March 28, 1983.

Date: June 28, 1983

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