May 19, 1988

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. 84 TO FACILITY OPERATING LICENSE NPF-9 AND AMENDMENT NO. 65 TO FACILITY OPERATING LICENSE NPF-17 - MCGUIRE NUCLEAR STATION, UNITS 1 AND 2 (TACS 60178/60179)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 84 to Facility Operating License NPF-9 and Amendment No. 65 to Facility Operating License NPF-17 for the McGuire Nuclear Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications in response to your application dated October 29, 1985, and supplemented August 25, 1986, May 26, 1987, and January 19, 1988.

The amendments change the Technical Specifications (TS) to accommodate removal of the resistance temperature detector (RTD) bypass manifold systems and the installation of in-line RTDs. In accordance with a telephone conversation with Mr. Scott Gewehr of your staff on May 4, 1988, minor editorial changes were made to Notes (2) and (3) of TS Table 3.3-2. The amendments are effective as of their date of issuance.

A copy of the related safety evaluation supporting Amendment No. 84 to Facility Operating License NPF-9 and Amendment No. 65 to Facility Operating License NPF-17 is enclosed.

Notice of issuance of amendments will be included in the Commission's next bi-weekly Federal Register notice.

Sincerely,

Original signed by:

Darl Hood, Project Manager
Project Directorate II-3
Division of Reactor Projects I/II

8806060062 880519
PDR ADOCK 05000369
P PDR

Enclosures:
1. Amendment No. 84 to NPF-9
2. Amendment No. 65 to NPF-17
3. Safety Evaluation

cc w/enclosures:
See next page

DSH/m
OFFICIAL RECORD COPY

PD#II-3/DRP-I/II	PD#II-3/DRP-I/II	PD#II-3/DRP-I/II
MRood	DHood:sw	DMatthews
5/11/88	5/11/88	5/19/88

Mr. H. B. Tucker
Duke Power Company

McGuire Nuclear Station

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

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

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

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

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

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

S. S. Kilborn
Area Manager, Mid-South Area
ESSD Projects
Westinghouse Electric Corporation
MNC West Tower - Bay 239
P. O. Box 355
Pittsburgh, Pennsylvania 15230

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

Mr. Dayne H. Brown, Chief
Radiation Protection Branch
Division of Facility Services
Department of Human Resources
701 Barbour Drive
Raleigh, North Carolina 27603-2008

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. 84
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 (the licensee) dated October 29, 1985, as supplemented August 25, 1986, May 26, 1987, and January 19, 1988 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and 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 rules and 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 amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   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.

8806060071 880519
PDR ADOCK 05000369
P PDR

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. 84 , are hereby incorporated into the 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

Original signed by:

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects-I/II

Attachment:
Technical Specification
Changes

Date of Issuance: May 19, 1988

OFFICIAL RECORD COPY

LA:PDII-3	PM:PDII-3 DSH	OGC-WF	D:PDII-3
MRood DS/H/m	DHood:sw	J. Goldberg	DMatthews
5/11/88	5/11/88	5/17/88	5/19/88

no legal objection

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

DUKE POWER COMPANY

DOCKET NO. 50-370

McGUIRE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 65
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 (the licensee) dated October 29, 1985, as supplemented August 25, 1986, May 26, 1987, and January 19, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and 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 rules and 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 amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   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. 65, are hereby incorporated into the 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

Original signed by:

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects-I/II

Attachment:
Technical Specification
Changes

Date of Issuance: May 19, 1988

OFFICIAL RECORD COPY

LA:PDII-3	PM:PDII-3	OGC-WF	D:PDII-3
MRood	DHood:	J. Goldberg	DMatthews
5/11/88	5/11/88	5/17/88	5/19/88

no legal objection

ATTACHMENT TO LICENSE AMENDMENT NO. 84

FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

AND

TO LICENSE AMENDMENT NO. 65

FACILITY OPERATING LICENSE NO. NPF-17

DOCKET NO. 50-370

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

Amended Page	Overleaf Page
2-5	
2-8	
2-9	
2-11	
B 2-4a (new page)	
B 2-5	
3/4 3-1	3/4 3-2
3/4 3-9	

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	N.A.	N.A.
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 25% of RATED THERMAL POWER High Setpoint - $\leq$ 109% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER High Setpoint - $\leq$ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
5. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
7. Overtemperature ΔT	See Note 1	See Note 3**
8. Overpower ΔT	See Note 2	See Note 4**
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*	$\geq$ 88.8% of design flow per loop*

*Design flow is 97,220 gpm per loop.
**Prior to removal of each unit's RTD bypass manifold, note 3a is applicable.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: OVERTEMPERATURE ΔT

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

Where:

ΔT = Measured ΔT by RTD Manifold Instrumentation,

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

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

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

τ_3 = Time constants utilized in the lag compensator for ΔT, $\tau_3 \leq$ 2 sec.*

ΔT_o = Indicated ΔT at RATED THERMAL POWER,

K_1 $\leq$ 1.200,

K_2 = 0.0222

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

τ_4, τ_5 = Time constants utilized in the lead-lag controller for T_{avg}, $\tau_4 \geq 28$ sec, $\tau_5 \leq 4$ sec.,

T = Average temperature, ^{o}F,

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

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 1: (Continued)

τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 \leq 2$ sec*
T'	=	$\leq$ 588.2°F Reference T_{avg} at RATED THERMAL POWER,
K_3	=	0.001095,
P	=	Pressurizer pressure, psig,
P'	=	2235 psig (Nominal RCS operating pressure),
S	=	Laplace transform operator, sec^{-1},

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

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

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

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

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

T	=	As defined in Note 1,
T''	=	$\leq$ 588.2°F Reference T_{avg} at RATED THERMAL POWER,
S	=	As defined in Note 1, and
$f_2(\Delta I)$	=	0 for all ΔI.

Note 3: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.6% of Rated Thermal Power.

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

Note 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 4.2% of Rated Thermal Power.

Overtemperature ΔT

The Overtemperature Delta T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower ΔT

The Overpower Delta T trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for overtemperature delta T protection, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, (2) rate of change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors, and (3) axial power distribution, to ensure that the allowable heat generation rate (kW/ft) is not exceeded. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP 9226, "Reactor Core Response to Excessive Secondary Steam Break."

BASES (With Bypass System Removed; RTDs in Thermowells)

Overtemperature ΔT

The Overtemperature Delta T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to thermal delays associated with the RTDs mounted in thermowells (about 5 seconds), and pressure is within the range between the Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower ΔT

The Overpower Delta T trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for overtemperature delta T protection, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, (2) rate of change of temperature for dynamic compensation for instrumentation delays associated with the loop temperature detectors, and (3) axial power distribution, to ensure that the allowable heat generation rate (kW/ft) is not exceeded. The Overpower ΔTtrip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP 9226, "Reactor Core Response to Excessive Secondary Steam Break."

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System Instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System Instrumentation channel and interlock shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific Reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

4.3.1.3 The response time of RTDs associated with the Reactor Trip System shall be demonstrated to be within their limits (see note 2 to Table 3.3-2) at least once per 18 months.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	10
2. Power Range, Neutron Flux - High Setpoint	4	2	3	1, 2	$2^{\#}$
Low Setpoint	4	2	3	$1^{\#\#\#}$, 2	$2^{\#}$
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	$2^{\#}$
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	$2^{\#}$
5. Intermediate Range, Neutron Flux	2	1	2	$1^{\#\#\#}$, 2	3
6. Source Range, Neutron Flux					
a. Startup	2	1	2	$2^{\#\#}$	4
b. Shutdown	2	1	2	3*, 4*, 5*	10
c. Shutdown	2	0	1	3, 4, and 5	5
7. Overtemperature ΔT					
Four Loop Operation	4	2	3	1, 2	$6^{\#}$
Three Loop Operation	(**)	(**)	(**)	(**)	(**)

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	N.A.
2. Power Range, Neutron Flux	$\leq$0.5 second (1)
3. Power Range, Neutron Flux, High Positive Rate	N.A.
4. Power Range, Neutron Flux, High Negative Rate	$\leq$0.5 second (1)
5. Intermediate Range, Neutron Flux	N.A.
6. Source Range, Neutron Flux	N.A.
7. Overtemperature ΔT	$\leq$10.0 seconds (1)(2)(3)
8. Overpower ΔT	$\leq$10.0 seconds (1)(2)(3)
9. Pressurizer Pressure--Low	$\leq$2.0 seconds
10. Pressurizer Pressure--High	$\leq$2.0 seconds
11. Pressurizer Water Level--High	N.A.

(1) Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.
(2) The $\leq$ 10.0 second response time includes a 6.5 second delay for the RTDs mounted in thermowells.
(3) The $\leq$ 10.0 second response time is applicable to each unit only after the RTD bypass manifold is removed; until then the value $\leq$ 8.0 sec.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 84 TO FACILITY OPERATING LICENSE NPF-9

AND AMENDMENT NO. 65 TO FACILITY OPERATING LICENSE NPF-17

DUKE POWER COMPANY

DOCKET NOS. 50-369 AND 50-370

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

INTRODUCTION

By letter dated October 29, 1985 and supplemented by letters dated August 25, 1986, May 26, 1987 and January 19, 1988, Duke Power Company (the licensee) requested amendments to Facility Operating License Nos. NPF-9 and NPF-17 for the McGuire Nuclear Station, Units 1 and 2. The proposed amendments would revise the Technical Specifications (TS) due to changes in the reactor trip system and engineered safety features response times to accommodate the removal of the Resistance Temperature Detector (RTD) bypass system and the installation of replacement RTDs in thermowells located directly in the hot leg and cold leg piping. This system will use narrow range fast response RTDs. This design modification is desired by the licensee because of problems with the existing RTD bypass system due to leakage from valve packing or mechanical joints. These problems reduce system reliability and result in high radiation doses during the performance of maintenance around the RTD bypass system.

The substance of the changes noticed in the Federal Register on September 10, 1986 and the proposed No Significant Hazards determination were not affected by the licensee's letters dated May 26, 1987 and January 19, 1988, which clarified certain aspects of the request.

EVALUATION

Present System Description

Currently, the hot and cold leg temperatures of each steam generator are measured by RTDs inserted into reactor coolant bypass loops. A bypass loop from upstream of the steam generator to downstream of the steam generator is used for the hot leg RTDs and a bypass loop from downstream of the reactor coolant pump to upstream of the pump is used for the cold leg RTDs. The RTDs are located in manifolds in the bypass loops and are directly inserted into the reactor coolant flow without thermowells. Each RTD manifold (one hot leg and one cold leg manifold per reactor coolant loop) contains two narrow-range RTDs: one for protection and control system inputs and one as a spare. Flow into each hot bypass is provided by three scoops located at 120° intervals around the hot leg pipe perimeter to take account of temperature variation across the pipe due to hot leg streaming. The action of the coolant pump provides well mixed coolant in the cold leg bypass manifold from a single tap into the cold leg.

8806060074 880519
PDR ADOCK 05000369
P PDR

Each loop's pair of RTDs (one in the hot leg and one in the cold leg) is used to provide inputs for protection system functions based on the average loop temperatures Tavg = $(T_{HOT} + T_{COLD})/2$ and the loop differential temperature, delta T $= T_{HOT} - T_{COLD}$. Protection functions based on these inputs are: overtemperature delta T and overpower delta T reactor trips with their associated (non-protection) rod stop and turbine runback actions, low Tavg main feedwater isolation, and low-low Tavg (P-12) steam dump block signals.

Each loop's pair of RTDs is also used to provide inputs for control system functions based on the average loop temperature and the loop differential temperature. Control functions based on these inputs are: turbine loading stop from auctioneered low Tavg; rod, steam dump and pressurizer level control from auctioneered high Tavg; and rod insertion limit alarms from auctioneered high delta T and Tavg.

Modified System Description

In the proposed modified system, after removal of the bypass loops, the hot leg temperature inputs from each reactor coolant loop will be developed from three fast response narrow range RTDs mounted in thermowells located within the three existing RTD bypass manifold scoops. An outlet port will be provided at the end of each scoop and the thermowell will be positioned so that the RTD sensing element is located near the middle inlet hole of the scoop. The objective of this design is to ensure that the temperature sensed by the RTD is close to that of the previous scoop flow.

One RTD per loop will be mounted in a thermowell located at the existing penetration for the bypass loop into the cold leg downstream of the coolant pump. Additionally, a new penetration will be added to each cold leg for a spare thermowell-mounted, narrow range RTD. The RTDs are placed in thermowells to allow replacement without draindown. The thermowells, however, increase the response time.

Each hot leg temperature input for protection system functions will be developed by electronically averaging the signals from the three new fast response, narrow range RTDs. This averaged input will replace the single input from the currently installed hot leg RTD. Each cold leg input for protection system functions will be provided by the new fast response, narrow range RTD which replaces the currently installed cold leg RTD. In the event of a hot leg RTD failure, the electronics allow a bias developed from historical data for the failed RTD to be manually added via a potentiometer to the remaining two RTD signals in order to obtain an average value comparable to the three-RTD average prior to failure of one RTD. If a cold leg RTD fails, the spare cold leg RTD can be used instead. The failure of an RTD would be detected by the Tavg or delta T deviation alarm.

Inputs for the control system functions will be provided, through isolators, from the average loop temperatures and loop differential temperatures calculated by the protection system. This aspect of the design has not been changed; only the use of three hot leg RTDs instead of one per loop to provide an average hot leg temperature is different.

Effect of Modifications on Overall RTD Response Time

In the existing bypass system, the overall RTD response time of 10.0 seconds consists of 2.0 seconds for the RTD bypass piping and thermal lag, 0.5 second for the RTD response time, 6.0 seconds for the RTD filter time constant and 1.5 seconds for the electronics delay. In the licensee's January 19, 1988 submittal, the overall response time of the new thermowell RTD hot leg temperature measurement system is also given as 10.0 seconds, and consists of 6.5 seconds for the RTD-thermowell combination, a 2 second electronic filter time constant and 1.5 seconds for the electronic delay.

Recent testing at another plant after completion of a similar RTD bypass system removal modification has resulted in response times slightly greater than anticipated. Also, as noted in NUREG-0809 (Reference 1), extensive RTD testing has revealed degradation of RTD response time with aging. In accordance with the guidance in NUREG-0809, the licensee in its January 19, 1988 submittal revised Technical Specification (TS) 4.3.1.2 to provide for response time testing of all RTDs once per 18 months. The testing method specified is the Loop Current Step Response (LCSR) method, which is the approved in-situ method for measuring RTD response time.

Effect of Modifications on Temperature Measurement Uncertainty

With regard to the effect of the proposed plant modification on the uncertainty of the temperature measurements, the new method of measuring each hot leg temperature with three thermowell RTDs manufactured by the RdF Corporation, used in place of the RTD bypass system with three scoops, has been analyzed to be slightly less accurate. The measurement uncertainty of the RTDs manufactured by the RdF Corporation is slightly greater (by about 0.5°F) than that of the existing Rosemount RTDs. Also, the new thermowell measurement may have a small streaming error relative to the former scoop flow measurement because of the temperature gradient over the 5-inch scoop span. On the other hand, the modified system eliminates hot leg temperature uncertainties due to unbalanced scoop flows. Hot leg temperature uncertainties are further decreased because of the statistical advantage of using three RTDs rather than the single RTD used in the bypass method. Because of these compensating factors, the overall effect of the modifications on Tavg and delta T values is small, and the current values of nominal setpoints for the McGuire TS would remain valid for the modification.

There will be no change in the present RTD temperature deviation alarms which include both a Tavg and a delta T deviation alarm. This alarm system compares the Tavg or delta T signals to a pre-set threshold value. This value is nominally set to + or - 2°F and is adjusted during startup and subsequent operation such that it is just beyond the range of normal operating variations.

RTD Drift

Literature studies (References 1 to 4) have indicated some tendency for long-term drift in RTD readings. The licensee will compensate for drift by calibrating the RTDs at each refueling. The calibration method is the Westinghouse-recommended RTD cross-calibration method at heatups after each refueling. This procedure requires multiple measurements at three or four different temperatures. To date, Westinghouse has evaluated the data from over 400 RTDs using this technique, and several repeat tests performed one to three years apart have not shown any indication of drift in only one direction. The results of the tests indicate that the RTDs drift less than was assumed for uncertainty calculations for the protection system. The procedure sensitivity is sufficient to discern a random drift of less than 1.0°F by one or several RTDs. If a drift is noticed, either the calibration of the resistance to voltage converter for the affected RTD would be adjusted to account for the shift, or, if the drift is appreciable, the RTD would be declared inoperable and would be replaced.

Comparison of Delta T Readings Before and After Modifications

Since both the old and the new methods of coolant temperature measurements have an inherent streaming inaccuracy, accounted for in the staff's safety analyses, it is not appropriate to compare the new method to the old method and declare any differences as errors. It is possible, however, to compare the normalized full power delta T measured before and after the modifications. It is expected that the delta T readings will be very similar once any secondary side measurement errors, such as feedwater flow, have been factored into the power calculation. If there were any significant differences between the two delta T readings, it would indicate that a problem existed with one of the measurement methods. The licensee will perform a comparison of the temperature indications after the modification with measurements prior to the modifications. The NRC will be notified of the results of this comparison including an explanation of any variations larger than expected.

Effect of Modifications on RCS Flow Measurement Uncertainty

The RCS flow measurement uncertainty after the RTD system modifications was analyzed by the licensee using the methodology in letter NS-EPR-2577 dated March 31, 1982 from E. P. Rahe, Jr. of Westinghouse to C. H. Berlinger of NRC. The methodology is based upon use of a calometric procedure to determine RCS flow. This analysis used data from the plant-specific instrumentation of the McGuire plant.

As mentioned above, RTD system modification will result in a slightly increased uncertainty in individual RTD readings and in the individual hot leg temperature determination for each loop. However, in using the calometric procedure to determine RCS flow, the temperatures in all four loops are considered. The RCS temperature uncertainties are reduced because data from the cross-calibration of the RTDs in all four loops during heatups before power operation are used. Because of this statistical advantage, the RCS flow measurement uncertainty remained the same as the current value of 1.7% (not including a 0.1% penalty for feedwater fouling allowance). The staff reviewed this analysis and finds that the flow measurement uncertainty will not be increased by the RTD system modifications and remains acceptable.

Effect of Modifications on Non-LOCA Accident Analyses

Non-LOCA accident analyses which rely on overtemperature and overpower delta T (OTDT and OPDT) reactor trips can potentially be affected by RTD modifications, primarily through their effect on RTD response time. These events include (1) Uncontrolled Rod Cluster Control Assembly (RCCA) withdrawal, (2) Uncontrolled Boron Dilution at Power, and (3) Steamline Rupture at Power.

Since the overall RTD response time in the modified system (10.0 seconds) will remain the same as in the present bypass system, there is no impact on the FSAR Chapter 15 non-LOCA accident analyses, and the conclusions presented in the FSAR and our SER remain valid.

Effect of Modifications on LOCA Accident Analysis

The replacement of the RTD bypass system will impact the uncertainties associated with RCS temperature and flow measurement. The effect of these uncertainties on the LOCA evaluation has been considered. The magnitudes of the uncertainties in the RCS inlet and outlet temperatures, thermal design flow rate and the steam generator performance data used in the LOCA analyses are such that the conclusions of the existing analyses are not affected. Past sensitivity studies concluded that the inlet temperature effect on peak clad temperature is dependent on break size. As a result of these studies, the LOCA analyses are performed at a nominal value of the inlet temperature without consideration of small uncertainties. The RCS flow rate and steam generator secondary side temperature and pressure are also determined using the loop average temperature (Tavg) output. These nominal values used as inputs to the analyses are not affected by the RTD modifications. We find that the replacement of the bypass system by the in-line thermowell RTDs will not affect the LOCA analyses input, and hence the results of the analyses remain unaffected. Therefore, the plant design changes due to the RTD bypass replacement are acceptable from a LOCA analysis standpoint.

Effect of Modifications on Plant Instrumentation and Controls

The staff has evaluated the effect of the proposed modification upon the plant's instrumentation and control system based upon Sections 7.2 and 7.3 of the Standard Review Plan (SRP). Those sections state that the objectives of the review are to confirm that the reactor trip and engineered safety features actuation system satisfy the requirements of the acceptance criteria and guidelines applicable to the protection system and will perform their safety function during all plant conditions for which they are required. Since the staff's review indicates that the modified system does not functionally change the reactor trip and engineered safety features actuation systems (except three hot leg RTDs are utilized instead of just one), the staff's existing evaluation conclusions for these systems, as documented in Section 7 of the SER for McGuire Units 1 and 2 (NUREG-0442), remain valid. Based on this and the licensee's statement that the new hardware for the RTD bypass elimination has been qualified to WCAP-8587, "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment," the staff finds the plant modifications to eliminate the RTD bypass manifold and to install fast response RTDs directly in the reactor coolant system hot and cold legs to be acceptable.

Changes to McGuire Technical Specifications

As a result of the proposed plant modifications to remove the existing RTD bypass manifolds and replace them by in-line RTDs, the following changes to the McGuire TS have been requested by the licensee:

Change 1 - In Table 2.2-1 under Functional Units 7 and 8, add "**" to the entries under Allowable Value to reference a new footnote (see Change 2).

Change 2 - On page 2-5 add a new footnote "** Prior to removal of each unit's RTD bypass manifolds, Note 3a is applicable."

Change 3 - In Table 2.2-1 under Functional Unit 8, revise the entry under Allowable Value from "Note 3" to "Note 4."

Change 4- In Table 2.2-1 under Functional Unit 12, revise the entry under Allowable Value from "89%" to "88.8%."

Change 5 - On page 2-8 revise the value for T_3 from " ≤ 6 sec." to " ≤ 2 sec."

Change 6 - On page 2-9 revise the value for T_6 from " ≤ 6 sec." to " ≤ 2 sec."

Change 7 - On page 2-11 revise the allowable value in Note 3 from "2%" to "3.6% of Rated Thermal Power."

Change 8 - On page 2-11 add the following new footnote: "Note 3a: The channel's maximum trip setpoint shall not exceed its computed trip setpoint by more than 2%."

Change 9 - On page 2-11 add the following new footnote: "Note 4: The channel's maximum trip setpoint shall not exceed its computed trip setpoint by more than 4.2% of Rated Thermal Power."

Change 10 - Add new page B 2-4a which is equivalent to old page B 2-5 with the phrase "(WITH RTD BYPASS SYSTEM INSTALLED)" added to the title "BASES."

Change 11 - On page B 2-5 add the phrase "(WITH BYPASS SYSTEM REMOVED; RTDs IN THERMOWELLS)" to the title "BASES."

Change 12 - On page B 2-5 under "Overtemperature ΔT," delete the words "piping transit delays from the core to the temperature detectors (about 4 seconds)" and substitute "thermal delays associated with the RTDs mounted in thermowells (about 5 seconds)" in the first sentence.

Change 13 - On page B 2-5 under "Overpower ΔT", delete the words, "for piping delays from the core to the" and substitute "for instrumentation delays associated with the" in the second sentence.

Change 14 - On page 3/4 3-9, change the footnote identified as "*" to footnote "(1)."

Change 15 - In Table 3.3-2 for Functional Units 2, 4, 7 and 8, under Response Time, reference footnote "(1)" in lieu of footnote "*."

Change 16 - In Table 3.3-2 change the response time for Functional Units 7 and 8, Overtemperature ΔT and Overpower ΔT, from "8.0" to "10.0."

Change 17 - In Table 3.3-2 for Functional Units 7 and 8 under Response Time, reference a new footnote "(2) The $\leq$10.0 second response time includes a 6.5 second delay for the RTDs mounted in thermowells" which is added to the page.

Change 18 - In Table 3.3-2 for Functional Units 7 and 8 under Response Time, reference a new footnote "(3) The $\leq$10.0 second response time is applicable to each unit only after the RTD bypass manifold is removed; until then the value $\leq$8.0 sec."

Change 19 - On page 3/4 3-1, add a new Surveillance Requirement: "4.3.1.3 The response time of RTDs associated with the reactor trip system shall be demonstrated to be within their limits (see Note 2 to Table 3.3-2) at least once per 18 months."

Changes 1, 2, 3, 8, 10, 11, 12, 13, 14, 15, 17, and 18 above are editorial changes necessary to encompass the removal of the RTD bypass manifold and the situation where removal of the bypass has been completed on only one of the two units. On the basis that these changes add clarity and conciseness to the technical specifications, we find them acceptable.

Changes 4, 7, and 9 above are new values based on revised instrumentation uncertainties resulting from the bypass manifold elimination. These new values were calculated using essentially the Westinghouse setpoint methodology as previously approved by the staff for generic use (see NUREG-0717, SER for Virgil C. Summer Nuclear Station) as documented in the licensee's letter dated May 26, 1987. The staff finds these changes acceptable.

Changes 5, 6, and 16 above are new values based on revised individual component response times resulting from the bypass manifold elimination. Since the new individual response times produce a total response time for the two reactor trips which were previously approved by the staff (see the staff's SER related to Amendment 42 to Facility Operating License NPF-9 and Amendment 23 to Facility Operating License NPF-17), we find these changes acceptable.

Change 19 provides a means to detect RTD drift and make appropriate adjustments before allowable limits are exceeded. This change is therefore acceptable.

Mechanical Safety Evaluation

The staff has reviewed the fabrication and inspection methods described in the licensee's letter dated October 29, 1985 for the replacement of the RTD bypass system with the new RTD thermowell system. This change requires modifications to the hot leg scoops, the crossover leg bypass return nozzle, the cold leg piping and the cold leg bypass manifold connection. The new thermowells, caps and penetrations will be fabricated in accordance with the ASME Code, Section III. The welding will be by approved procedures and inspected by penetrant testing per the ASME Code Section XI. In accordance with Article IWA-4000 of Section XI, a hydrostatic test of the new pressure boundary welds will be performed.

The staff finds that the mechanical aspects of the proposed RTD thermowell system, fabricated, examined and tested as described above, are acceptable.

Radiological Safety Evaluation

The licensee has estimated the occupational radiation exposure for the RTD bypass modification in the submittals of October 29, 1985 and August 25, 1986. The estimate is based on anticipated stay times for each major subtask and estimated dose rates. The estimates per loop and per unit are given in the table below.

	Subtask	Manhour Estimate	Dose Estimate (Person-Rem)
(1)	Preparation for RTD bypass modification	33	1.09
(2)	Shielding Installation/ Removal	64	9.6
(3)	Remove/Replace pipes, hangers, electrical interferences, etc...	417	11.1
(4)	Modify the RTDs	120	12.0
	Total per loop	634	33.79
	Total per unit (4 loops)	2536 man-hours	135.16 person-rem

Specific measures to keep doses as low as is reasonably achievable (ALARA) will include preplanning of mechanical operations, use of temporary shielding and special tooling, familiarization of workers with the work area, and close supervision of the work in process by health physics technicians and ALARA personnel. The licensee will adhere to administrative limits for occupational exposure to individual workers at McGuire which are lower than the NRC limits in 10 CFR 20.101 (for example the licensee's administrative whole body dose limit is 1.0 Rem per quarter, which is less than the 10 CFR 20.101(b) dose limit of 1.25 Rem per quarter). Therefore, the licensee's administrative limits for individual workers, as applied to the RTD modifications, are acceptable.

After replacement of the RTD bypass manifold system, present occupational exposures associated with valve and manifold maintenance, in-service inspection and snubber inspection would be avoided. Over the life of the plant, the licensee estimates this dose would be approximately 1250 person-rem per unit. The net occupational exposure savings would therefore be approximately 1115 person-rem per unit.

No significant liquid or gaseous radioactive wastes are expected to be generated as a result of the RTD replacement implementation activities. Therefore no increases in liquid or airborne effluents (and related offsite doses to the public or maximum individuals) are expected as a result of the modifications. Some solid radwaste will be generated, and the licensee has specifically identified the radioactive materials slated for disposal (typically valves, hangers, and possible decontamination materials). This waste will be shipped to an appropriate land burial site, or scrapped if decontamination is feasible.

The solid radwaste volume will be about 13.6 cubic meters, containing an estimated 8.4 curies of radioactivity. This is only about 8% of the average annual volume of radwaste shipped from the McGuire station, and less than 2% of the average volume of radioactive waste shipped per PWR in recent years (729 cubic meters per PWR per year, 1980 - 1984). The licensee has identified dose rates and contamination levels which fall into the typical ranges of such wastes. The types, volumes and activities of these wastes as characterized are well within the parameters of normal operations evaluated for radiological impact in the FES and SER for McGuire 1 & 2.

On the basis of the above considerations, and the licensee's radiation protection programs previously found to be acceptable in the SER, the staff concludes that the radiological and ALARA aspects of the proposed RTD replacement are acceptable, and that the proposed modification will result in an overall reduction in occupational exposure.

ENVIRONMENTAL CONSIDERATION

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

CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (51 FR 32266) on September 10, 1986. The Commission consulted with the state of North Carolina. No public comments were received, and the state of North Carolina did not have any comments.

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

REFERENCES

(1) NUREG-0809, Safety Evaluation Report, Review of Resistance Temperature Detector Time Response Characteristics, August 1981.

(2) NUREG-CR-4928, Degradation of Nuclear Plant Temperature Sensors, June 1987.

(3) K. R. Carr, An Evaluation of Industrial Platinum Resistance Thermometer Temperature - Its Measurement and Control in Science and Industry, ISA publication, Vol. 4, Part 2, 1972, pages 971-982.

(4) B. W. Mangum, The Stability of Small Industrial Platinum Resistance Thermometers, Journal of Research of the NBS, Vol. 89, No. 4, July-August 1984, Pages 305-350.

Principal Contributors: S. Kirslis, PDII-3/DRPI/II
D. Hood, PDII-3/DRPI/II
F. Burrows, DEST/SRXB
H. Balukjian, DEST/SRXB

Dated: May 19, 1988

50-369

DATED: May 19, 1988

AMENDMENT NO. 84 TO FACILITY OPERATING LICENSE NPF-9 - McGuire Nuclear Station, Unit 1
AMENDMENT NO. 65 TO FACILITY OPERATING LICENSE NPF-17 - McGuire Nuclear Station, Unit 2

DISTRIBUTION:
Docket File
NRC PDR
Local PDR
PD#II-3 R/F
McGuire R/F
S. Varga
G. Lainas
D. Matthews
M. Rood
D. Hood
OGC-WF
J. Partlow
E. Jordan
W. Jones
T. Barnhart (8)
ACRS (10)
GPA/PA
ARM/LFMB
E. Butcher
D. Hagan
W. Hodges, SRXB
S. Newberry, ICSB
J. Craig, PSB