February 17, 1988

Docket Nos.: 50-413
and 50-414

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. 40 to Facility Operating License NPF-35 and Amendment No. 33 to Facility Operating License NPF-52 - Catawba Nuclear Station, Units 1 and 2 (TACS 65524/65525)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 40 to Facility Operating License NPF-35 and Amendment No. 33 to Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications in response to your application dated July 22, 1987, as supplemented May 26, August 31, October 1, October 30, November 19 and December 14, 1987.

The amendments modify the Technical Specifications to accommodate the removal of the resistance temperature device (RTD) bypass manifold systems and the installation of in-line RTDs. These amendments are effective as of their date of issuance.

A copy of the related safety evaluation supporting Amendment No. 40 to Facility Operating License NPF-35 and Amendment No. 33 to Facility Operating License NPF-52 is enclosed.

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

Sincerely,

Kahtan N. Jabbour, Project Manager
Project Directorate II-3
Division of Reactor Projects I/II

Enclosures:
1. Amendment No. 40 to NPF-35
2. Amendment No. 33 to NPF-52
3. Safety Evaluation

cc w/enclosures: See next page

PDII-3/DRPI/II	PDII-3/DRPI/II	PDII-3/DRPI/II	PDII-3/DRPI/II
MRood/mac	SKirslis	KJabbour	Acting PD
01/22/88	01/22/88	01/22/88	02/17/88

Mr. H. B. Tucker
Duke Power Company

Catawba Nuclear Station

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

DUKE POWER COMPANY

NORTH CAROLINA ELECTRIC MEMBERSHIP CORPORATION

SALUDA RIVER ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-413

CATAWBA NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 40
License No. NPF-35

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

   A. The application for amendment to the Catawba Nuclear Station, Unit 1 (the facility) Facility Operating License No. NPF-35 filed by the Duke Power Company acting for itself, North Carolina Electric Membership Corporation and Saluda River Electric Cooperative, Inc., (licensees) dated July 22, 1987, and supplemented May 26, August 31, October 1, October 30, November 19, and December 14, 1987, 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-35 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 40, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into the license. Duke Power Company 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

Lawrence P. Crocker

Lawrence P. Crocker, Acting Director
Project Directorate II-3
Division of Reactor Projects I/II

Attachment:
Technical Specification Changes

Date of Issuance: February 17, 1988

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

DUKE POWER COMPANY

NORTH CAROLINA MUNICIPAL POWER AGENCY NO. 1

PIEDMONT MUNICIPAL POWER AGENCY

DOCKET NO. 50-414

CATAWBA NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 33
License No. NPF-52

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

   A. The application for amendment to the Catawba Nuclear Station, Unit 2 (the facility) Facility Operating License No. NPF-52 filed by the Duke Power Company acting for itself, North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency, (licensees) dated July 22, 1987, as supplemented May 26, August 31, October 1, October 30, November 19, and December 14, 1987, 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-52 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 33 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into the license. Duke Power Company 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

Lawrence P. Crocker , Acting Director
Project Directorate II-3
Division of Reactor Projects I/II

Attachment:
Technical Specification Changes

Date of Issuance: February 17, 1988

BC/RPB
L. J. Cunningham
2/17/88

PDII-3/DRPI/II MRood/mac 01/21/88	PDII-3/DRPI/II SKirslis 01/21/88	KNJ PDII-3/DRPI/II KJabbour 01/21/88	OGC-WFlint MYoung 02/9/88
PDII-3/DRPI/II Acting PD 02/17/88	AD/DRPII GLainas 01/ /88		

UHY no obj to check STATE & PCY before issuance

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 40 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into the license. Duke Power Company 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

, Acting Director
Project Directorate II-3
Division of Reactor Projects I/II

Attachment:
Technical Specification Changes

Date of Issuance: February 17, 1988

BC/RPB
L. J. Cunningham
2/17/88

PDII-3/DRPI/II	PDII-3/DRPI/II	KNJ PDII-3/DRPI/II	OGC-WFlint
MRood/mac	SKirslis	KJabbour	
01/21/88	01/21/88	01/21/88	02/9/88

PDII-3/DRPI/II
Acting PD
02/17/88

ATTACHMENT TO LICENSE AMENDMENT NO. 40

FACILITY OPERATING LICENSE NO. NPF-35

DOCKET NO. 50-413

AND

TO LICENSE AMENDMENT NO. 33

FACILITY OPERATING LICENSE NO. NPF-52

DOCKET NO. 50-414

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.

Amended Page	Overleaf Page
2-4	2-3
2-7	
2-8	
2-10	2-9
B 2-5	B 2-6
3/4 3-1	
3/4 3-7	
3/4 3-29	3/4 3-30
3/4 3-35	
3/4 3-36	

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlocks Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value Column of Table 2.2-1, adjust the Setpoint consistent with the Trip Setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1 and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

Equation 2.2-1 $Z + R + S \leq TA$

Where:

Z = The value from Column Z of Table 2.2-1 for the affected channel,

R = The "as measured" value (in percent span) of rack error for the affected channel,

S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 2.2-1 for the affected channel, and

TA = The value from Column TA (Total Allowance) of Table 2.2-1 for the affected channel.

TABLE 2.2.-1
REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	0	≤109% of RTP*	≤111.1% of RTP*
b. Low Setpoint	8.3	4.56	0	≤25% of RTP*	≤27.1% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥ 2 seconds	≤6.3% of RTP* with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.4	0	≤25% of RTP*	≤31% of RTP*
6. Source Range, Neutron Flux	17.0	10	0	≤10^5 cps	≤1.4×10^5 cps
7. Overtemperature ΔT	7.2(8.9$^{\#}$)	4.47(5.41$^{\#}$)	2.03(2.65$^{\#}$)	See Note 1	See Note 2
8. Overpower ΔT	4.3(4.9$^{\#}$)	1.3(1.24$^{\#}$)	1.2(1.7$^{\#}$)	See Note 3	See Note 4
9. Pressurizer Pressure-Low	4.0	2.21	1.5	≥1945 psig	≥1938 psig***
10. Pressurizer Pressure-High	7.5	4.96	0.5	≤2385 psig	≤2399 psig
11. Pressurizer Water Level-High	5.0	2.18	1.5	≤92% of instrument span	≤93.8% of instrument span
12. Reactor Coolant Flow-Low	2.5	1.77(1.41$^{\#}$)	0.6	≥90% of loop design flow**	≥89.2% (88.8%$^{\#}$) of loop design flow**

*RTP = RATED THERMAL POWER
**Loop design flow = 96,900 gpm
***Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 2 seconds for lead and 1 second for lag. Channel calibration shall ensure that these time constants are adjusted to these values.
#Applicable upon deletion of RTD Bypass System.

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

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

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 lead-lag compensator for ΔT, $\tau_1 = 8(12^{\#})$ s, $\tau_2 = 3$ s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constant utilized in the lag compensator for ΔT, $\tau_3 = 0$;
ΔT_0	=	Indicated ΔT at RATED THERMAL POWER;
K_1	=	$1.411(1.38^{\#})$;
K_2	=	$0.02401/^{\circ}F$;
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_4 = 28(22^{\#})$ s, $\tau_5 = 4$ s;
T	=	Average temperature, $^{\circ}F$;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$;

TABLE 2.2-1 (Continued)
TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T'	$\leq$	590.8°F (Nominal T_{avg} allowed by Safety Analysis);
K_3	=	0.001189;
P	=	Pressurizer pressure, psig;
P'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, s^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron 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 -22.5% and -6.5%, $f_1(\Delta I) = 0$, 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 $q_t - q_b$ is more negative than -22.5%, 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 $q_t - q_b$ is more positive than -6.5%, the ΔT Trip Setpoint shall be automatically reduced by 1.641% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.4%(3.0%#).

TABLE 2.2-1 (Continued)
TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

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

Where:

ΔT	=	As defined in Note 1,
$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	As defined in Note 1,
τ_1, τ_2	=	As defined in Note 1,
$\frac{1}{1+\tau_3 S}$	=	As defined in Note 1,
τ_3	=	As defined in Note 1,
ΔT_o	=	As defined in Note 1,
K_4	=	1.0704,
K_5	=	0.02/°F for increasing average temperature and 0 for decreasing average temperature,
$\frac{\tau_7 S}{1+\tau_7 S}$	=	The function generated by the rate-lag controller for T_{avg} dynamic compensation,
τ_7	=	Time constant utilized in the rate-lag controller for T_{avg}, τ_7 = 10 s,
$\frac{1}{1+\tau_6 S}$	=	As defined in Note 1,
τ_6	=	As defined in Note 1,

TABLE 2.2-1 (Continued)
TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6 = 0.001707/°F for T > 590.8°F and K_6 = 0 for T $\leq$ 590.8°F,

T = As defined in Note 1,

T" = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 590.8°F),

S = As defined in Note 1, and

$f_2(\Delta I)$ = 0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.6%(2.8%#).

#Applicable upon deletion of RTD Bypass System.

Intermediate and Source Range, Neutron Flux

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

Overtemperature ΔT

The Overtemperature ΔT 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 (1) (with the RTD Bypass System installed) piping transit delays from the core to the temperature detectors (about 4 seconds), or (2) (with the RTD Bypass System removed) 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 2.2-1. 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 ΔT 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 ΔT trip, 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, and (2) rate of change of temperature for dynamic compensation for either piping delays from the core to the loop temperature detectors (with the RTD Bypass System installed), or instrumentation delay associated with the loop temperature detectors (with the RTD Bypass System removed), 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 Releases."

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, is automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power, the Pressurizer High Water Level trip is automatically blocked by P-7 (a level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, is automatically reinstated by P-7.

Reactor Coolant Flow

The Low Reactor Coolant Flow trips provide core protection and prevents DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90% of nominal full loop flow. Above P-8 (a power level of approximately 48% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. Conversely, on decreasing power between P-8 and P-7 an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

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 and the automatic trip logic 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. The response time of RTDs associated with the Reactor Trip System shall be demonstrated to be within their limits at least once per 18 months.

*This surveillance need not be performed for the primary RTD response time testing portion of items 7 and 8 from Table 3.3-2 for Unit 2 until prior to entering STARTUP following the Unit 2 first refueling.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

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

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

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
4. Steam Line Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure-High-High	12.7	0.71	1.5	$\leq$ 3 psig	$\leq$ 3.2 psig
d. Steam Line Pressure - Low	4.6	1.31	1.5	$\geq$ 725 psig	$\geq$ 694 psig*
e. Steam Line Pressure-Negative Rate - High	8.0	0.5	0	$\leq$ 100 psi	$\leq$ 122.8 psi**
5. Feedwater Isolation					
a. Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Steam Generator Water Level-High-High (P-14)					
1. Unit 1	5.4	2.18	1.5	$\leq$ 82.4% of narrow range instrument span	$\leq$ 84.2% of narrow range instrument span
2. Unit 2	9.7	2.18	1.5	$\leq$ 78.1% of narrow range instrument span	$\leq$ 79.9% of nar[illegible] range instrument span
c. T_{avg}-Low	4.0(6.0[#])	1.12(0.71[#])	1.2(0.8[#])	$\geq$ 564°F	$\geq$ 562°F(561°F[#])
d. Doghouse Water Level-High	1.0	0	0.5	11 inches above 577' floor level	12 inches above 577' floor level
e. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
6. Turbine Trip					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Steam Generator Water Level-High-High (P-14)					
1. Unit 1	5.4	2.18	1.5	$\leq$ 82.4% of narrow range instrument span	$\leq$ 84.2% of narrow range instrument span
2. Unit 2	9.7	2.18	1.5	$\leq$78.1% of narrow range instrument span	$\leq$79.9% of narrow range instrument span
d. Trip of All Main Feedwater Pumps	N.A.	N.A.	N.A.	N.A.	N.A.
e. Reactor Trip (P-4)	N.A.	N.A.	N.A.	N.A.	N.A.
f. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				
7. Containment Pressure Control System					
a. Start Permissive	N.A.	N.A.	N.A.	$\leq$ 0.4 psid	$\leq$ 0.45 psid
b. Termination	N.A.	N.A.	N.A.	$\geq$ 0.3 psid	$\geq$ 0.25 psid
8. Auxiliary Feedwater					
a. Manual Initiation	N.A	N.A.	N.A.	N.A	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
18. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	1955 psig	$\geq$1944 psig
b. Pressurizer Pressure, not P-11	N.A.	N.A.	N.A.	1955 psig	$\leq$1966 psig
c. Low-Low T_{avg}, P-12	N.A.	N.A.	N.A.	$\geq$553°F	$\geq$551°F(550°F#)
d. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.
e. Steam Generator Level, P-14	See Item 5. above for all Steam Generator Water Level Trip Setpoints and Allowable Values.				

TABLE 3.3-4 (Continued)

TABLE NOTATIONS

*Time constants utilized in the lead-lag controller for Steam Line Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds. Channel calibration shall ensure that these time constants are adjusted to these values.

**The time constant utilized in the rate-lag controller for Steam Line Pressure-Negative Rate-High is greater than or equal to 50 seconds. Channel calibration shall ensure that this time constant is adjusted to this value.

#Applicable upon deletion of RTD Bypass System.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 40 TO FACILITY OPERATING LICENSE NPF-35

AND AMENDMENT NO. 33 TO FACILITY OPERATING LICENSE NPF-52

DUKE POWER COMPANY, ET AL.

DOCKET NOS. 50-413 AND 50-414

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

INTRODUCTION

By letter dated July 22, 1987, and supplemented by letters dated May 26, August 31, October 1, October 30, November 19 and December 14, 1987, Duke Power Company, et al., (the licensee) requested amendments to Facility Operating License Nos. NPF-35 and NPF-52 for the Catawba Nuclear Station, Units 1 and 2. The proposed amendments would revise the Technical Specifications due to changes in the reactor trip system and engineered safety features response times to accommodate the removal of the Resistance Temperature Device (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 resistance temperature detectors (RTDs). This design modification is to overcome major drawbacks of the RTD bypass system which lacked reliability (leakage from valve packing or mechanical joints) and resulted 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 December 2, 1987 and the proposed No Significant Hazards determination was not affected by the licensee's letter dated December 14, 1987, which clarified certain aspects of the request.

EVALUATION

Currently, the hot and cold leg temperatures 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 and are directly inserted into the reactor coolant bypass loop 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 leg bypass loop 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 using a single tap into the cold leg.

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; rod insertion limit alarms from auctioneered high delta T and Tavg.

In the proposed modified system, 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 (except for Loop B where two of the three thermowells will be mounted in the scoops, but the third thermowell, because of structural interference, will be located 8.5 inches downstream of the existing scoop in an independent boss). An outlet port is provided at the end of each scoop and the thermowell is 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. 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 the 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.

The RTD modifications affect plant accident analysis by changing the RTD response time and hot leg temperature measurement uncertainty. In the licensee's July 22, 1987 submittal, the overall response time of the new thermowell RTD hot leg temperature measurement system is given as 7.0 seconds, made up of 5.5 seconds for the RTD thermowell combination and 1.5 seconds for the electronic delays. The increase over the 4.0 second response time for the bypass system was principally due to slow conduction through the thermowell. Because of the increased channel response time, there are no longer delays from the time when fluid conditions in the reactor coolant system (RCS) require an overtemperature delta T or overpower delta T reactor trip until the trip actually takes place. However, as reported in the licensee's submittal of July 22, 1987, the original safety analyses for the bypass RTD system conservatively assumed a response time of 8.0 seconds and this response time was found to be acceptable.

In the supplemental submittal of November 19, 1987, the licensee changed the RTD response time from 7.0 to 8.0 seconds. The 8.0 second response time provided one second of added margin in the analyses.

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 November 19, 1987 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.

Since the safety analyses referenced in the licensee's July 22, 1987 submittal found that the 8.0 second response time was acceptable, no additional analyses are required to justify the proposed revision to 8.0 seconds.

With regard to the effect of the 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 more accurate. The new RTD thermowell with measurement at one point may have a small streaming error relative to the former scoop flow measurement because of a temperature gradient over the 5-inch scoop span. However, this gradient has been calculated to have a small effect. Also, since possible temperature uncertainties from imbalanced scoop flows are eliminated, the overall result is more reliable. In addition, since the new method uses three RTDs for each hot leg temperature measurement, it is statistically a more accurate temperature measurement than the former method which used only one RTD for each hot leg temperature measurement. Therefore, the current values of nominal setpoints for the Catawba Technical Specifications are still valid.

There has been 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.

The method to be used by the licensee for calibrating the RTDs at each refueling prior to startup 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.

Since both the old and new methods of coolant temperature measurement 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 modification. 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 dramatic 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 modification. The NRC will be notified of the results of this comparison including any explanation of variations larger than expected.

Non-LOCA accident analyses are affected by the plant modifications primarily through their effect of increasing RTD response time. Only those events which rely on the Overtemperature and Overpower delta T (OTDT and OPDT) reactor trips are impacted. The accidents in FSAR Sections 15.1 to 15.6 were examined and the following non-LOCA accidents affected by the longer response time were reanalyzed: (1) the Uncontrolled Rod Cluster Control Assembly (RCCA) Withdrawal; (2) uncontrolled boron dilution at power; and (3) the Steamline Rupture at Power. The applicant stated that the LOFTRAN computer code was used for the analysis of these events.

The first accident, Uncontrolled RCCA Withdrawal, is described in Section 15.4.2 of the FSAR. For this event, the High Neutron Flux and Overtemperature delta T reactor trips are assumed to provide protection against DNB. This event was analyzed with the increased time constants and lead/lag changes. Plots of DNBR versus time were provided which showed that the DNBR criterion was met for this accident.

For the Boron Dilution at Power event, manual operation, as described in Section 15.4.6 of the FSAR, the time from initiation of the event to reactor trip is determined from the Uncontrolled RCCS Withdrawal at Power analysis. The licensee stated that based upon the results of the Uncontrolled RCCS Withdrawal at Power analysis, the conclusions presented in the FSAR for the Boron

Dilution at Power event, manual operation, remain valid, i.e., there is greater than 15 minutes from the time of an alarm until the total loss of shutdown margin occurs.

For the Steamline Rupture at Power event the analysis included the increased response time and lead/lag changes. The analysis showed that the design basis as described in WCAP-9226-Rev. 1, "Reactor Core Response to Excessive Secondary Steam Release", January 1978, has been met.

The effect of the increase in RTD response time on the FSAR Chapter 15 Loss of Load/Turbine Trip event is analyzed for both beginning and end of life conditions in Section 15.2.3 of the FSAR. No credit for reactor trip on turbine trip is assumed in the safety analyses. Therefore, reactor trips on high pressurizer pressure, overtemperature delta T, and low-low steam generator water level reactor trips provide the necessary protection for this event during the starting mode. For the Loss of Load/Turbine Trip analyses presented in the FSAR, increased RTD response times were assumed for the Catawba positive moderator temperature coefficient (MTC) safety evaluation. For all four cases analyzed, reactor trips occurred on either a high pressurizer pressure or low-low steam generator water level signal. An overtemperature delta T signal was never generated prior to reactor trip. Therefore, the analyses currently presented in the Catawba FSAR, based on the positive MTC safety evaluation, have adequately addressed the increased RTD response time resulting from the RTD bypass elimination.

The impact of the increased RTD response time on the FSAR Chapter 15 non-LOCA accident analyses has been evaluated. For the events impacted, it was demonstrated that the conclusions presented in the FSAR remain valid.

The elimination of the RTD bypass system impacts the uncertainties associated with RCS temperature and flow measurement. The effect of these uncertainties on the LOCA evaluation was 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 previous analyses will not be 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 bypass elimination. It is concluded that the elimination of the RTD bypass piping 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 elimination are acceptable from a LOCA analysis standpoint without requiring any reanalysis.

The RCS flow measurement uncertainty after the RTD bypass removal modifications was analyzed using the methodology in WCAP-11169 Rev. 1, "RCS Flow Uncertainty for Shearon Harris Unit 1," October 1986. This analysis used the plant-specific instrumentation for the Catawba Plant. The results of the analysis indicated that the flow measurement uncertainty was reduced from the current value of ±2.1% (not including a 0.1% penalty for feedwater venturi fouling allowance) to a new value of ±1.7% (including the cold leg elbow taps and excluding feedwater venturi fouling). Much of this reduced uncertainty is from the statistical advantage of using three RTDs for the hot leg temperature measurement in the new method rather than the one in the former method. The licensee has chosen not to request any plant specification changes to take advantage of the reduced flow uncertainty. Since the 2.1% allowance is conservative, its retention is acceptable.

The staff's review and evaluation of the plant's instrumentation and controls is based upon Sections 7.2 and 7.3 of the 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 original evaluation conclusions for these systems, as documented in Section 7 of the SER for Catawba Units 1 and 2 (NUREG-0954), 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.

As a result of the plant modifications and new instrumentation associated with the removal of the existing RTD bypass manifold and replacement by fast response RTDs, the following changes to the plant's Technical Specifications were proposed:

Change 1 - Include new additional entries for the Total Allowance, Z and Sensor Error for Functional Unit 7, Overtemperature delta T, in Table 2.2-1 of "(8.9#)," "(5.41#)," and "(2.65#)" respectively.

Change 2 - Include new, additional entries for the Total Allowance, Z and Sensor Error for Functional Unit 8, Overpower delta T, in Table 2.2-1 of "(4.9#)," "(1.24#)," and "(1.7#)" respectively.

Change 3 - Include new additional entries for Z and Allowable Value for Functional Unit 12, Reactor Coolant Flow-low, in Table 2.2-1 of "(1.41#)" and "(88.8%#)" respectively.

Change 4 - On page 2-4 add a new footnote "# Applicable upon deletion of the RTD Bypass System."

Change 5 - Include new additional entries for t_1, K_1 and t_4 in NOTE 1 to Table 2.2-1 of "(12#)," "(1.38#)" and "(22#)" respectively.

Change 6 - Include a new, additional entry, "(3.0%#)," for the allowable value for overtemperature delta T contained in NOTE 2 to Table 2.2-1.

Change 7 - Include a new, additional entry, "(2.8%#)," for the allowable value for overpower delta T contained in NOTE 4 to Table 2.2-1.

Change 8 - On page 2-10 add a new footnote "# Applicable upon deletion of RTD Bypass System."

Change 9 - On page B 2-5 under "Overtemperature delta T," add "(1) (with the RTD Bypass System installed)" to the first sentence between "to" and "piping." Also add "or (2) (with the RTD Bypass System removed) thermal delays associated with the RTDs mounted in thermowells (about 5 seconds)," before "and pressure" in the first sentence.

Change 10 - On page B 2-5 under "Overpower delta T," add "either" to the second sentence between "for" and "piping." Also add "(with the RTD Bypass System installed), or instrumentation delay associated with the loop temperature detectors (with the RTD Bypass System removed)," between "detectors" and "to" in the second sentence.

Change 11 - On page 3/4 3-1 add a sentence stating that: "The response time of RTDs associated with the Reactor Trip System shall be demonstrated to be within their limits at least once per 18 months."

Change 12 - Include a new, additional entry, "(8#)," for the response times for Functional Unit 7, Overtemperature delta T, and Functional Unit 8, Overpower delta T, in Table 3.3-2.

Change 13 - On page 3/4 3-7 add a new footnote "# Applicable upon deletion of RTD Bypass System.

Change 14 - Include new, additional entries in Table 3.3-4 for the Total Allowance, Z, Sensor Error and Allowable Value for Functional Unit 5.c, Tavg-Low, of "(6.0#)," "(0.71#)," "(0.8#)" and "(561°F#)" respectively.

Change 15 - Include a new, additional entry in Table 3.3-4 for the Allowable Value for Functional Unit 18.c, Low-Low Tavg, P-12, of "(550°F#)."

Change 16 - On page 3/4 3-36 add a new footnote "# Applicable upon deletion of RTD Bypass System."

Changes 1, 2, 3, 5, 6, 7, 14, and 15 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 Catawba and for generic use (NUREG-0717, SER for Virgil C. Summer Nuclear Station) as documented in the licensee's letter dated July 22, 1987. The staff finds these changes acceptable. Change 11 provides an additional surveillance to verify that the RTDs associated with the Reactor Trip System remain within their limits. On the basis that this change would ensure RTD operability, the staff finds it acceptable.

Changes 4, 8, 9, 10, 13, and 16 are editorial changes necessary to accommodate the removal of the RTD bypass manifold and the situation where removal of the bypass manifold has been completed on only one of the two units. On the basis that these changes add clarity and conciseness to the plant's technical specifications, the staff finds them acceptable.

Change 12 is acceptable because an RTD response time of 8.0 seconds has been found to be acceptable in previous safety analyses.

The staff has reviewed the fabrication and inspection methods described in WCAP-11308, Rev. 2, RTD Bypass Elimination Report for Catawba Units 1 and 2, September 1987 for the replacement of the RTD bypass system with the new RTD thermowell system. This change requires modifications to the hot leg piping, 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, Class 1. 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 carried out.

The staff finds that the mechanical safety of the proposed RTD thermowells system fabricated, examined and tested as described above is acceptable.

The licensee has estimated the occupational radiation exposure for the RTD bypass modification in the submittal of October 30, 1987. The estimate is based on anticipated stay times for each major subtask and estimated dose rates. The annual estimates per unit are given in the table below.

	Subtask	Manhour Estimate	Dose Estimate (Person-Rem)
(1)	Preparation for RTD bypass modification	33	1.0
(2)	Shielding Installation/ Removal	64	9.6
(3)	Remove/Replace pipes, hangers, electrical interferences, etc...	417.5	10.4
(4)	Modify the RTDs	100	12.0
	Total per loop	614.5	33.0
	Total per unit (4 loops)	2458 man-hours	132.0 person-rem

The dose avoided through reduced maintenance and operational requirements is the order of 50 to 100 person-rem per unit per year. Comparing this to the total one-time dose of 132 person-rem for the RTD replacement operation, a net savings of several thousand person-rem over plant life can be projected.

An estimate of the curies of beta and gamma radioactivity contained on the RTD components to be removed (piping, insulation, hangers, rupture restraints, valves and instrumentation) is 5.26 curies per unit. The expected total volume of this radwaste is 574 cubic feet.

Based on the above and on the licensee's radiation protection and ALARA programs previously evaluated and found to be acceptable in Chapter 12 of the SER, the staff concludes that the RTD bypass removal is acceptable from the radiological viewpoint.

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. 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.

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.

CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (52 FR 45885) on December 2, 1987. The Commission consulted with the state of South Carolina. No public comments were received, and the state of South 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.

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Dated: February 17, 1988

DATED: February 17, 1988

AMENDMENT NO. 40 TO FACILITY OPERATING LICENSE NPF-35 - Catawba Nuclear Station, Unit 1
AMENDMENT NO. 33 TO FACILITY OPERATING LICENSE NPF-52 - Catawba Nuclear Station, Unit 2

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