March 8, 1991

Docket No. 50-348

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Mr. W. G. Hairston, III
Senior Vice President
Alabama Power Company
40 Inverness Center Parkway
Post Office Box 2641
Birmingham, Alabama 35201

Dear Mr. Hairston:

SUBJECT: ISSUANCE OF AMENDMENT NO. 87 TO FACILITY OPERATING LICENSE NO. NPF-2 REGARDING RTD BYPASS ELIMINATION AND STEAM GENERATOR TUBE PLUGGING - JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1, (TAC NO. 77966)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 87 to Facility Operating License NPF-2 for the Joseph M. Farley Nuclear Plant, Unit 1. The amendment consists of changes to the Technical Specifications in response to your submittal dated October 26, 1990, as supplemented January 14 and 31, and February 15, 1991.

The amendment changes the Technical Specifications to eliminate the resistance temperature detector bypass system and to allow an average of 15 percent steam generator tube plugging with a peak of 20 percent in any one steam generator. The amendment also includes an approximate 1.5 percent reduction in the reactor coolant system thermal design flow.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

Orignal signed by:

Stephen T. Hoffman, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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PDR ADOCK 05000348
P PDR

Enclosures:
1. Amendment No. 87 to NPF-2
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	LA:PD21:DRPE	PM:PD21:DRPE	D:PD21:DRPE	BC:DET:EMCB		
NAME	PAnderson	SHoffman:dt	EAdensam	CCheng		
DATE	3/5/91	3/6/91	3/8/91	3/6/91		

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Document Name: FARLEY AMENDMENT 77966

AMENDMENT NO. 87 TO FACILITY OPERATING LICENSE NO. NPF-2 - FARLEY, UNIT 1

Docket File
NRC PDR
Local PDR
PDII-1 Reading
S. Varga (14E4)
G. Lainas
E. Adensam
P. Anderson
S. Hoffman
OGC
D. Hagan (MNBB 3302)
E. Jordan (MNBB 3302)
G. Hill (4) (P1-137)
Wanda Jones (P-130A)
R. Jones 8-E-23
S. Newberry 8-H-3
C. Cheng 7-D-4
J. Calvo (11D3)
L. Cunningham (10-D-4)
ACRS (10)
GPA/PA
OC/LFMB
Farley File

cc: Farley Service List

Mr. W. G. Hairston, III
Alabama Power Company

Joseph M. Farley Nuclear Plant

cc:

Mr. R. P. McDonald
Executive Vice President
Nuclear Operations
Alabama Power Company
P. O. Box 1295
Birmingham, Alabama 35201

Mr. B. L. Moore
Manager, Licensing
Alabama Power Company
P. O. Box 1295
Birmingham, Alabama 35201

Mr. Louis B. Long, General Manager
Southern Company Services, Inc.
Houston County Commission
P. O. Box 2625
Birmingham, Alabama 35202

Mr. D. N. Morey
General Manager - Farley Nuclear Plant
P. O. Box 470
Ashford, Alabama 36312

Mr. J. D. Woodward
Vice-President - Nuclear
Farley Project
Alabama Power Company
P. O. Box 1295
Birmingham, Alabama 35201

Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 24 - Route 2
Columbia, Alabama 36319

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

Chairman
Houston County Commission
Dothan, Alabama 36301

Claude Earl Fox, M.D.
State Health Officer
State Department of Public Health
State Office Building
Montgomery, Alabama 36130

James H. Miller, III, Esq.
Balch and Bingham
P. O. Box 306
1710 Sixth Avenue North
Birmingham, Alabama 35201

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

ALABAMA POWER COMPANY

DOCKET NO. 50-348

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 87
License No. NPF-2

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

   A. The application for amendment by Alabama Power Company (the licensee), dated October 26, 1990, as supplemented January 14 and 31, and February 15, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, 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;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; 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 amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-2 is hereby amended to read as follows:

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(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 87 , are hereby incorporated into the license. Alabama Power Company shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Orignal signed by:

Elinor G. Adensam, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 8, 1991

OFC	:LA:PD21:DRPE	:PM:PD21:DRPE	: OGC	:D:PD21:DRPE	:	:	:
NAME	: PAnderson	: SHoffman:sw	:	: EAdensam	:	:	:
DATE	: 2/13/91	:2/13/91	: 2/26/91	:3/8/91	:	:	:

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Farley Amendment 77966

ATTACHMENT TO LICENSE AMENDMENT NO. 87

FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages as indicated. The revised areas are indicated by marginal lines.

Remove Pages	Insert Pages
2-2	2-2
2-5	2-5
2-8	2-8
2-9	2-9
2-10	2-10
B 2-4	B 2-4
B 2-5	B 2-5
3/4 2-15	3/4 2-15
3/4 3-10	3/4 3-10
3/4 3-27	3/4 3-27
3/4 3-28	3/4 3-28

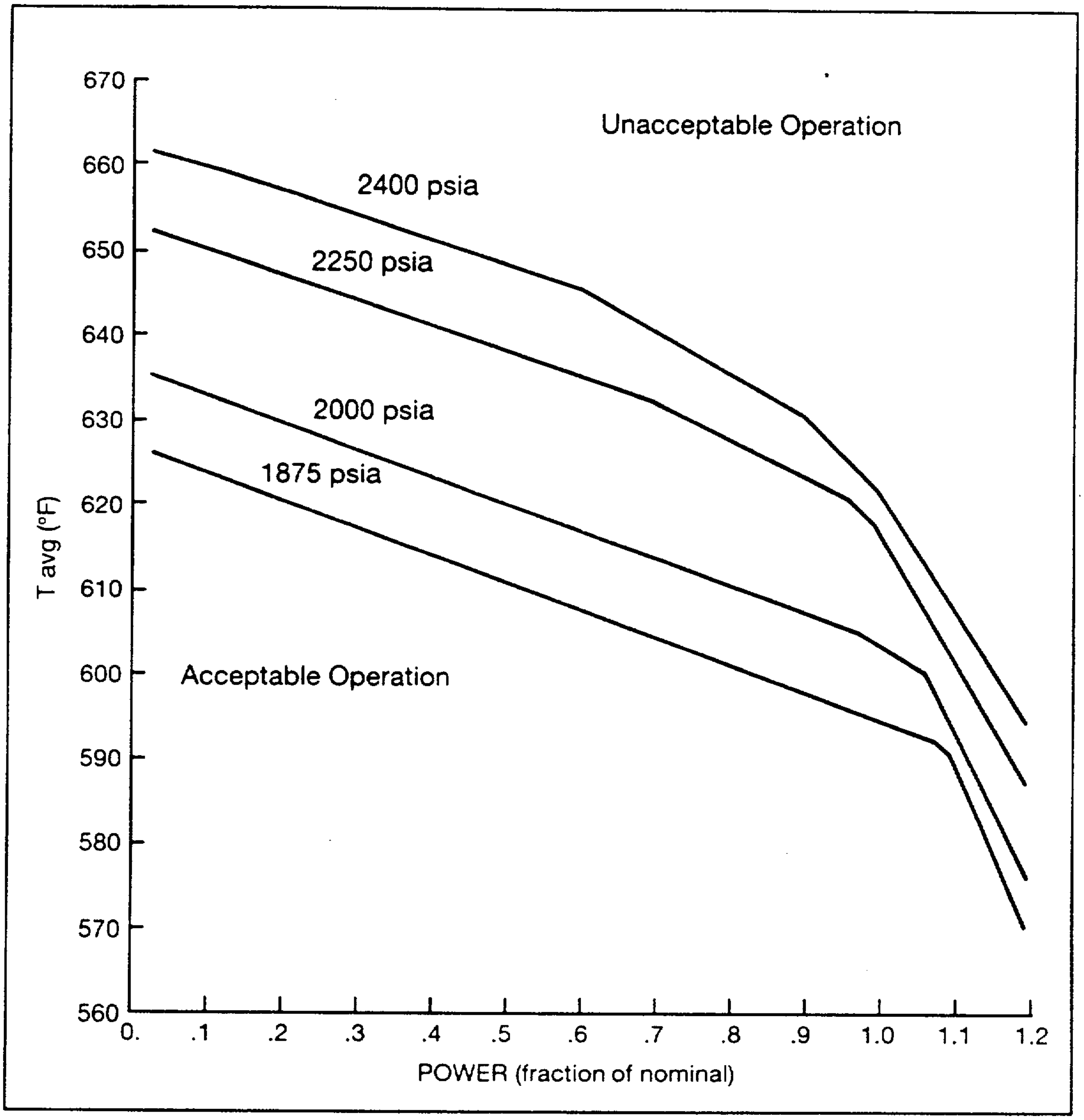

Figure 2.1-1 Reactor Core Safety Limit
Three Loops in Operation

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - ≤ 25% of RATED THERMAL POWER	Low Setpoint - ≤ 26% of RATED THERMAL POWER
	High Setpoint - ≤ 109% of RATED THERMAL POWER	High Setpoint - ≤ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 second	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 second
4. Power Range, Neutron Flux, High Negative Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 second	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 second
5. Intermediate Range, Neutron Flux	≤ 25% of RATED THERMAL POWER	≤ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	≤ 10^5 counts per second	≤ 1.3 X 10^5 counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 6
9. Pressurizer Pressure--Low	≥ 1865 psig	≥ 1855 psig
10. Pressurizer Pressure--High	≤ 2385 psig	≤ 2395 psig
11. Pressurizer Water Level--High	≤ 92% of instrument span	≤ 93% of instrument span
12. Loss of Flow	≥ 90% of design flow per loop*	≥ 88.5% of design flow per loop*

*Design flow is 87,200 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

Note 1: Overtemperature ΔT

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

where: ΔT = Measured ΔT by RTD instrumentation;

ΔT_o = Indicated ΔT at RATED THERMAL POWER;

T = Average temperature, °F;

$T' \leq$ 577.2°F (Maximum Reference T_{avg} at RATED THERMAL POWER);

P = Pressurizer pressure, psig;

P' = 2235 psig (Nominal RCS operating pressure);

$\frac{1 + \tau_1 s}{1 + \tau_2 s}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation;

τ_1 & τ_2 = Time constants utilized in the lead-lag controller for T_{avg} τ_1 = 30 secs, τ_2 = 4 secs;

$\frac{1 + \tau_4 s}{1 + \tau_5 s}$ = The function generated by the lead-lag controller for ΔT dynamic compensation;

τ_4 & τ_5 = Time constants utilized in the lead-lag controller for ΔT, $\tau_4 = \tau_5$ = 0 seconds;

$\frac{1}{1 + \tau_6 s}$ = Lag compensator on measured T_{avg};

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 sec;

s = Laplace transform operator, sec^{-1};

Operation with 3 loops

K_1 = 1.18;

K_2 = 0.0154;

K_3 = 0.000635;

Operation with 2 loops

K_1 = (values blank pending

K_2 = NRC approval of

K_3 = 2 loop operation)

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

and f_1 (ΔI) 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 -35 percent and +9 percent, f_1 (ΔI) = 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 -35 percent, the ΔT trip setpoint shall be automatically reduced by 1.37 percent of its value at RATED THERMAL POWER; and

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds +9 percent, the ΔT trip setpoint shall be automatically reduced by 1.75 percent of its value at RATED THERMAL POWER.

Note 2: Overpower ΔT

$$\Delta T \frac{(1 + \tau_4 s)}{(1 + \tau_5 s)} \leq \Delta T_o \left[K_4 - K_5 \left(\frac{\tau_3 s}{1 + \tau_3 s}\right) \left(\frac{1}{1 + \tau_6 s}\right) T - K_6 \left(T \left(\frac{1}{1 + \tau_6 s}\right) - T''\right) - f_2(\Delta I)\right]$$

where: ΔT = Measured ΔT by RTD instrumentation;

ΔT_o = Indicated ΔT at RATED THERMAL POWER;

T = Average temperature, °F;

T'' = Reference T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 577.2°F);

K_4 = 1.08;

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

K_6 = 0.00109/°F for $T > T''$ K_6 = 0 for $T \leq T''$;

$\frac{\tau_3 s}{1+\tau_3 s}$ = The function generated by the rate lag controller for T_{avg} dynamic compensation;

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

τ_3 = Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs;

$\frac{1 + \tau_4 s}{1 + \tau_5 s}$ = The function generated by the lead-lag controller for ΔT dynamic compensation;

τ_4 & τ_5 = Time constants utilized in the lead-lag controller for ΔT, $\tau_4 = \tau_5 = 0$ seconds;

$\frac{1}{1 + \tau_6 s}$ = Lag compensator on measured T_{avg};

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 sec;

s = Laplace transform operator, sec $^{-1}$;

f2(ΔI) = 0 for all ΔI.

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 3.4 percent.

Note 4: Pressure value to be determined during initial startup testing. Pressure value of $\leq$ 55 psia to be used prior to determination of revised value.

Note 5: Pressure value to be determined during initial startup testing.

Note 6: The channel's maximum trip point shall not exceed its computed trip point by more than 2.9 percent.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

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, thermowell, and RTD response time delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for transport, thermowell, and RTD response time delays from the core to RTD output indication. 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.

Operation with a reactor coolant loop out of service below the 3 loop P-8 setpoint does not require reactor protection system setpoint modification because the P-8 setpoint and associated trip will prevent DNB during 2 loop operation exclusive of the Overtemperature delta T setpoint. Two loop operation above the 3 loop P-8 setpoint is permissible after resetting the K1, K2, and K3 inputs to the Overtemperature delta T channels and raising the P-8 setpoint to its 2 loop value. In this mode of operation, the P-8 interlock and trip functions as a High Neutron Flux trip at the reduced power level.

Overpower ΔT

The Overpower delta T reactor trip provides assurance of fuel integrity (e.g., no fuel pellet melting) 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 includes corrections for axial power distribution, changes in density and heat capacity of water with temperature, and dynamic compensation for transport, thermowell, and RTD response time delays from the core to RTD output indication. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 10 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drop below 90% of nominal full loop flow. Above 36% (P-8) of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. This

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 3 Loops in Operation	2 Loops in Operation
Reactor Coolant System T_{avg}	$\leq$ 581.5°F	(**)
Pressurizer Pressure	$\geq$ 2220 psia*	(**)
Reactor Coolant System Total Flow Rate	$\geq$ 267,600 gpm***	(**)

* Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

** Values blank pending NRC approval of 2 loop operation.

*** Value includes a 2.3% flow uncertainty (0.1% feedwater venturi fouling bias included).

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	Not Applicable
2. Power Range, Neutron Flux	
a. High	$\leq$ 0.5 seconds*
b. Low	Not Applicable
3. Power Range, Neutron Flux, High Positive Rate	Not Applicable
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 0.5 seconds*
5. Intermediate Range, Neutron Flux	Not Applicable
6. Source Range, Neutron Flux	Not Applicable
7. Overtemperature ΔT	$\leq$ 6.0 seconds*
8. Overpower ΔT	Not Applicable
9. Pressurizer Pressure--Low	$\leq$ 2.0 seconds
10. Pressurizer Pressure--High	$\leq$ 2.0 seconds
11. Pressurizer Water Level--High	Not Applicable

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

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. STEAM LINE ISOLATION		
a. Manual	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High-High	$\leq$ 16.2 psig	$\leq$ 18.2 psig
d. Steam Flow in Two Steam Lines--High, Coincident with T_{avg}--Low-Low	$\leq$ A function defined as follows: A Δp corresponding to 40% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load with $T_{avg} \geq 543°F$	$\leq$ A function defined as follows: A Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 111.5% of full steam flow at full load with $T_{avg} \geq 540°F$
e. Steam Line Pressure--Low	$\geq$ 585 psig	$\geq$ 575 psig
5. TURBINE TRIP AND FEED WATER ISOLATION		
a. Steam Generator Water Level--High-High	$\leq$ 75% of narrow range instrument span each steam generator	$\leq$ 76% of narrow range instrument span each steam generator

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
6. AUXILIARY FEEDWATER		
a. Automatic Actuation Logic	N.A.	N.A.
b. Steam Generator Water Level--Low-Low	≥ 17% of narrow range instrument span each steam generator	≥ 16% of narrow range instrument span each steam generator
c. Undervoltage - RCP	≥ 2680 volts	≥ 2640 volts
d. S.I.	See 1 above (all SI Setpoints)	
e. Trip of Main Feedwater Pumps	N.A.	N.A.
7. LOSS OF POWER		
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	≥ 3255 volts bus voltage*	≥ 3222 volts bus voltage* ≤ 3418 volts bus voltage*
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	≥ 3675 volts bus voltage*	≥ 3638 volts bus voltage* ≤ 3749 volts bus voltage*
8. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS		
a. Pressurizer Pressure, P-11	≤ 2000 psig	≤ 2010 psig
b. Low-Low T_{avg}, P-12 (Increasing) (Decreasing)	 544°F 543°F	 ≤ 547°F ≥ 540°F
c. Steam Generator Level, P-14	(See 5. above)	
d. Reactor Trip, P-4	N.A.	N.A.

* Refer to appropriate relay setting sheet calibration requirements.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 87 TO FACILITY OPERATING LICENSE NO. NPF-2

ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-348

1.0 INTRODUCTION

By letter dated October 26, 1990, as supplemented January 14 and 31, and February 15, 1991, Alabama Power Company (APCo or the licensee) submitted a request for changes to the Joseph M. Farley Nuclear Plant (Farley), Unit 1, Technical Specifications.

Farley, Unit 1, currently has a steam generator tube plugging (SGTP) limit of 10% based on the large break loss-of-coolant accident/emergency core cooling system (LOCA/ECCS) analysis shown on Technical Specification Figure 2.1-1. Based on APCo operating experience, it is expected that the number of steam generator tubes requiring corrective action in Unit 1 could exceed the current SGTP limit of 10%. Therefore, APCo has requested a change to the Technical Specifications to increase the SGTP limit from 10% to an average 15% SGTP with a peak limit of 20% SGTP in any one steam generator. Also included in the request is a reduction of approximately 1.5% in the reactor coolant system thermal design flow.

In support of the increased SGTP limit, the licensee submitted a report, WCAP-12694, "Alabama Power, Joseph M. Farley Unit No. 1, Increased Steam Generator Tube Plugging and Reduced Thermal Design Flow Licensing Report," dated August 1990. This report provides the licensee's review and evaluation of the Final Safety Analysis Report (FSAR), Chapter 15, accidents/transients to verify that the effects of increased tube plugging and reduced reactor coolant system (RCS) flow rate do not invalidate the current analyses of record and that all pertinent conclusions in the FSAR are still valid. The licensee also considered the effect of asymmetric RCS flow condition on accidents/transients. The following events were reanalyzed to justify the Technical Specification changes:

- o Large break LOCA/ECCS analysis
- o Small break LOCA
- o Major rupture of a main feedwater pipe
- o Uncontrolled rod cluster control assembly bank withdrawal from subcritical

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- Partial loss of forced reactor coolant flow
- Single reactor coolant pump locked rotor
- Steam generator tube rupture

The application for amendment also requested a revision of Technical Specifications Table 2.2-1, 3.2-1, 3.3-2, 3.3-4 and the Technical Specification Bases for overtemperature delta T/overpower delta T. The proposed amendment supports a plant modification to replace the existing resistance temperature detector (RTD) bypass manifold system with thermowell mounted, narrow range, fast response, dual element RTDs located directly in the RCS piping. The RTD bypass modification affects the FSAR Chapter 15 safety analysis because of revised response time characteristics and instrumentation uncertainties associated with the new thermowell mounted RTDs. The reactor protection system arithmetic average loop temperature (T-average) and loop differential temperature (delta-T) inputs and inputs to the plant control system are also modified.

The initial submittal on October 26, 1990, was later supplemented by submittals dated January 14 and 31, and February 15, 1991. These submittals provided revised analyses to incorporate additional penalties and uncertainties and minor revisions to Technical Specification pages. These supplemental submittals did not substantially alter the action noticed or change the staff's proposed initial determination of no significant hazards consideration as published in the Federal Register on December 26, 1990 (55 FR 53067).

2.0 EVALUATION

2.1 INCREASED TUBE PLUGGING LIMIT/REDUCED REACTOR COOLANT FLOW

2.1.1 LOCA Events

Large Break LOCA/ECCS

The limiting reactor coolant system large pipe break was found to be the double ended cold leg guillotine (DECLG) break based on the results of the LOCA sensitivity studies. Therefore, only the DECLG break is considered in the large break ECCS performance analysis to determine the effects of increased SGTP and reduced thermal design flow. Calculations were performed for the limiting Moody break discharge coefficient (C_D=0.4) under minimum safeguard conditions. The DECLG was analyzed with an NRC approved ECCS evaluation model.

The peak clad temperature (PCT) for the DECLG break was calculated to be 2069°F, which accounts for increased SGTP and reduced thermal design flow. A 4°F increase is added due to delayed isolation of the containment mini-purge valves, and 60°F for loose parts. This brings the resultant PCT to 2133°F for Farley, Unit 1. In addition, the impact of steam generator flow area reduction due to seismic effects has been considered and a PCT penalty of 50°F has been conservatively assessed. The resulting PCT for Farley, Unit 1, is 2183°F which is below the 10 CFR 50.46 limit of 2200°F.

The maximum local metal-water reaction is 5.76 percent, which is well below the embrittlement limit of 17 percent required by 10 CFR 50.46. The total core metal-water reaction is less than 0.3 percent when compared with the 1% criterion of 10 CFR 50.46. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The core temperature will continue to drop, and the ability to remove decay heat generated in the fuel for an extended period of time will be achieved.

The staff has concluded that the calculations for increased SGTP and reduced thermal design flow were performed for the worst case LOCA break, used an approved evaluation model which satisfies the requirements of Appendix K to 10 CFR Part 50, and met the requirements of 10 CFR 50.46. Thus, the staff finds the LOCA/ECCS evaluation acceptable.

Steam Generator Tube Collapse

In WCAP-12694, Westinghouse Electric Corporation (Westinghouse) has identified what appears to be a new issue for older model Westinghouse steam generators (such as the Farley, Unit 1, Model 51 steam generators) that is considered by the staff to be a separate issue from SGTP limits and this amendment. The issue concerns the potential for steam generator tube collapse during a safe shutdown earthquake (SSE) plus LOCA. Collapse of the steam generator tubing reduces the RCS flow area through the tubes. The reduction in flow area increases the resistance to the flow of steam from the core during a LOCA which in turn may potentially increase PCT.

This phenomenon has previously been examined in detail by Westinghouse for newer model steam generators (e.g., Model F at Callaway and Model D-3 at Watts Bar) and factored into the FSAR safety analyses for these plants. However, this phenomenon was not examined for Farley until preparation of WCAP-12659 which supported a Farley, Unit 2, license amendment issued on December 6, 1990, for the same increased steam generator tube plugging limits. Until the Farley, Unit 2, submittal, this phenomenon had not been previously reviewed by the staff.

The staff's concerns are the amount of potential flow area reduction and the potential tube integrity implications of collapsed tubes. Potential tube integrity implications arise from the fact that many plants are experiencing stress corrosion cracking of steam generator tubes. The staff is concerned that collapse of cracked tubes could lead to leakage of secondary system coolant into the primary system during a LOCA.

The staff's preliminary conclusion, however, is that the issue of tube collapse does not pose a significant enough safety concern to warrant immediate action. This conclusion is based on the fact that leak-before-break (LBB) analyses have been performed for most pressurized water reactors in accordance with General Design Criterion (GDC) 4 of Appendix A to 10 CFR Part 50. These analyses have shown that a large break LOCA (and, thus, consequent tube collapse) is an extremely low probability event for these plants. Therefore, the staff is examining, on a generic basis, this issue of tube collapse under SSE plus LOCA loads.

Details of the tube collapse assessment for Farley were presented to the staff at a meeting on November 7, 1990. The meeting handouts were documented by APCo's letter to the staff dated November 18, 1990. In addition, in a January 14, 1991, letter, the licensee submitted a scoping analysis stating that relevant LBB parameters for Farley, Unit 1, are enveloped by the generic analyses performed by Westinghouse in WCAP-9558, Revision 2, "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Through-Wall Crack," and accepted by the NRC in Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops." Based on the above analyses, the licensee concluded that the LBB methodology is applicable to the Farley, Unit 1, RCS primary loops and, thus, the probability of breaks in the RCS loop piping is sufficiently low that they need not be considered in the structural design basis. Excluding breaks in the RCS primary loops, the LOCA loads from the large branch line breaks were also assessed by the licensee and found to be of insufficient magnitude to induce tube collapse.

In summary, the staff finds that the subject amendment can be issued pending resolution of this issue. The issue of tube collapse is generic; and, based on the LBB considerations discussed above, the staff believes that this issue does not pose a significant safety concern requiring immediate resolution on Farley, Unit 1. The staff will continue to pursue resolution of the generic concerns independent of Farley, Unit 1. Therefore, the staff finds that Farley, Unit 1, can operate in accordance with this amendment prior to resolution of the generic issue without undue risk to the health and safety of the public. The staff will take appropriate action upon resolution of the generic issue if found to be warranted.

Small Break LOCA

Small break LOCA analyses were performed to demonstrate that the NOTRUMP small break LOCA evaluation model (WCAP-10054-P-A) calculated lower PCTs than the WFLASH evaluation model (WCAP-11145-P-A). The Farley WLFASH small break LOCA analysis remains the analysis of record which calculates a PCT of about 1797°F.

The increase in SGTP and the reduction in thermal design flow will result in a small change in primary pressures and temperatures. It is concluded that these changes will have no adverse effect on the Farley, Unit 1, small break LOCA analysis margin to the PCT limit of 2200°F.

Steam Generator Tube Rupture

A sensitivity analysis was performed to determine the impact of the tube plugging increase and thermal design flow reduction on the steam generator tube rupture (SGTR) analysis. The results of the SGTR analysis indicate that the primary-to-secondary break flow and atmospheric steam release via the ruptured steam generator increased when compared to the results of the current Farley, Unit 1, SGTR analysis.

The increased mass releases were subsequently utilized by the licensee in a radiological analysis to determine the effect of the tube plugging increase and thermal design flow reduction on the offsite doses. The licensee used the Farley licensing basis methodology and current inputs. The results of the radiological analysis indicate that the site boundary thyroid and whole-body gamma doses are 3.3 and 0.14 rem, respectively. The low population zone thyroid and whole-body gamma doses are 1.4 and 0.05 rem, respectively.

These results show a slight increase in the offsite dose over those presented in the FSAR. The staff has reviewed the methodology and assumptions used by the licensee to analyze the radiological impact of a postulated steam generator tube rupture and finds this analysis appropriate. The dose increases are small, and the total dose remains well within a "small fraction" of the 10 CFR Part 100 exposure guidelines. Thus, we find the SGTR analysis acceptable.

2.1.2 Non-LOCA Evaluation

All non-LOCA transients were examined to determine the impact of the reduced thermal design flow. A penalty in the departure from nucleate boiling (DNB) margin is associated with the reduced flow. However, the existing DNB margin is sufficient to cover the DNB penalty due to reduced thermal design flow. The thermal design flow reduction is limited to approximately 1.5%. The licensee used the existing flow sensitivities data to demonstrate that non-DNB safety criteria will also continue to be met.

The licensee explicitly reanalyzed (1) major rupture of a main feedwater pipe and (2) uncontrolled rod cluster control assembly bank withdrawal from subcritical for the reduced thermal design flow. These events were reanalyzed using current and NRC accepted methodology and computer codes. Although the results of the analyses have changed, the conclusions presented in the FSAR remain valid for the new analyses.

Steam generator tube plugging asymmetries lead to flow asymmetries among the reactor coolant loops. The loop with the largest amount of tube plugging will have the lowest reactor coolant flow. The licensee explicitly reanalyzed the transients which are sensitive to flow asymmetries. The two transients analyzed were (1) partial loss of forced reactor coolant flow and (2) single reactor coolant pump locked rotor. The licensee used the NRC-approved methodology to account for the loop flow difference and a reduced thermal design flow.

The results of the partial loss of forced reactor coolant flow analysis show that the minimum DNB is bounded by the complete loss of forced reactor coolant flow analysis. Therefore, the increased tube plugging with reduced thermal design flow, as well as the asymmetrical steam generator tube plugging levels, does not alter the conclusions presented in the FSAR for the partial loss of forced reactor coolant flow event. The results of single reactor coolant pump locked rotor show that the conclusions of the FSAR with respect to the locked rotor event are met for the increased SGTP as well.

Thus, the staff finds that the non-LOCA events evaluation is acceptable.

2.2 RTD BYPASS MANIFOLD SYSTEM REPLACEMENT

2.2.1 Current System

The present reactor coolant temperature measurement system uses coolant scoops in the primary coolant to divert a portion of the reactor coolant into bypass manifold loops. The RTDs for T-hot and T-cold temperature measurement are located within the bypass manifolds and are inserted directly into the reactor coolant bypass flow without thermowells. Separate bypass loops are provided for each reactor coolant loop such that individual T-hot and T-cold loop temperature signals can be developed for use in the reactor protection and control systems. A bypass loop from the hot leg side of each steam generator to the intermediate leg is used for the T-hot RTDs. Another bypass loop from the cold leg side of the reactor coolant pump to the intermediate leg is used for the T-cold RTDs. Both T-hot and T-cold manifolds empty through a common header to the intermediate leg between the steam generator and reactor coolant pump. Flow for each T-hot bypass loop is provided by three scoop tubes located at 120 degree intervals around the hot leg. Because of the mixing effects of the reactor coolant pump only one scoop connection is required for bypass flow to the T-cold bypass manifold.

The bypass manifold system was developed to resolve concerns with temperature streaming (temperature gradients) within the hot leg primary coolant. The temperature streaming is caused by incomplete mixing of the coolant leaving various regions of the reactor core at different temperatures. The bypass manifold system compensates for the temperature streaming by allowing the primary coolant to mix within the bypass manifold. The bypass system also limits high velocity coolant flow to the RTDs and allows RTD replacement without the need to drain the RCS.

The output from the bypass loop RTDs provides the signals necessary to calculate T-average and delta-T. The T-average and delta-T signals are then input to the reactor protection system. The input of T-average and delta-T signals to the plant control system are derived from a separate set of bypass loop RTDs and T-average and delta-T calculations.

However, as referenced by the licensee, the bypass manifold system created its own set of operational problems. Examples presented by the licensee included plant shutdowns due to primary leakage through valves or flanges, and by interruption of bypass flow due to valve stem failure. Additionally, the licensee stated that the bypass piping contributes to increased radiation exposure throughout the loop compartments when maintenance must be performed in these areas.

2.2.2 Proposed System

In contrast to the bypass manifold system, the modified system hot leg temperature measurement for each loop will be obtained using three fast response, narrow range, dual element RTDs mounted in thermowells. Where possible, the hot leg RTDs will be mounted in thermowells within the existing bypass manifold scoop penetrations. Each bypass scoop will be modified such that reactor coolant will flow in through the existing holes of the bypass scoop past the RTD/thermowell assembly and out through a new hole machined in the bypass scoop. If structural components interfere with the placement of a thermowell in an existing scoop, then the scoop will be capped and an alternate penetration will be made to accommodate the RTD thermowell. This modified RTD arrangement will perform the same sampling/temperature averaging function as the original bypass manifold system.

The cold leg temperature measurements will be obtained by one fast response, narrow range, dual element RTD located at the discharge of the reactor coolant pump. This RTD will be mounted in a thermowell within the existing cold leg bypass manifold penetration. Because of the mixing action of the reactor coolant pump, temperature gradients in the cold leg are eliminated and, as a result, only one RTD is necessary for cold leg temperature measurement. As in the hot leg, the bypass manifold penetration will be modified to accept the RTD thermowell. Additionally, the bypass manifold return line will be capped at the nozzle on the intermediate leg.

The licensee will replace the bypass manifold direct-immersion RTDs with Weed Instrument Co., Inc., dual element RTDs mounted in thermowells. The spare element of each RTD will be terminated at the 7300 process system electronics rack input terminals in the control room. This arrangement is intended to allow on-line accessibility to the RTD spare elements in the event of an RTD element failure.

The licensee states that the new thermowell mounted RTDs have a response time equal to or faster than the maximum allowed time for the old bypass piping transport, thermal lag and direct immersion RTD (about 4 seconds). The

4-second response time of the Weed RTD is a conservative value that is supported by industry experience. The RTD manufacturer will perform response time testing of each RTD and thermowell prior to installation to ensure the RTD/thermowell response time is bounded by the values referenced in Technical Specification Table 2.1-1. The licensee will also verify the response time of the new RTDs after installation in the plant. The additional electronic delays of the new thermowell mounted RTD system are such that the response time of the modified RTD system will continue to meet the requirements (6 seconds) currently referenced in the Technical Specification Table 2.1-1.

These modifications will not affect the single existing wide range RTDs installed in each hot and cold leg of the reactor coolant system. These RTDs will continue to provide hot and cold leg temperature information for reactor startup, shutdown, or post-accident monitoring.

To accomplish the hot leg temperature averaging function previously done by the bypass manifold system, the modified hot leg RTD temperature signals (three per loop) will be electronically averaged in the reactor protection system. The averaged T-hot signal will then be used with the T-cold signal to calculate reactor coolant system loop delta-T and T-average values for use in the reactor protection and control systems. The averaging function will be accomplished by additions to existing 7300 reactor protection equipment.

The present bypass system uses separate dedicated RTDs for the control and protection systems. However, the modified system thermowell mounted RTDs are used for both protection and control. This Class IE to Non-Class IE interface requires the use of isolation devices for the control system T-average and delta-T signals derived from the reactor protection system. The licensee has stated that the isolation devices utilized in the bypass manifold modification are 7300 (NLP-3) devices and were previously reviewed under WCAP-8892-A. The T-average and delta-T signals used in the control grade logic are input into a median signal selector (MSS) in lieu of the high auctioneered T-average or delta-T signal used by the present plant control system. The MSS selects the signal that is between the highest and lowest values of the three T-average and delta-T loop inputs. By selecting the median value, the MSS provides the plant control system with a valid T-average and delta-T value. The MSS also preserves the functional independence between control and protection systems that now share common sensors within the RPS by preventing spurious control system responses caused by a single signal failure.

To ensure proper operation of the MSS, the existing manual switches that defeat a T-average or delta-T signal from a single loop will be eliminated. Also, the conversion to thermowell mounted RTDs will result in the elimination of the control grade RTDs and their associated control board indicators. The protection system channels will now provide inputs to the control system through isolators and the MSS. The existing control board alarms, indicators and T-average and delta-T deviation alarms will continue to provide the means to detect RTD failures.

An RTD failure in the cold leg can be handled by using the spare cold leg RTD element provided within each loop. A failure of a hot leg RTD can be managed in two ways. The first method disconnects the failed element and reconnects the spare element of the same RTD. The second method requires plant personnel to manually defeat the failed signal and rescale the electronics to average the remaining two hot leg RTD inputs. A bias value is then added to the T-hot average signal to compensate for the failed RTD and maintain a value comparable with the previous three RTD average. The bias value is developed per procedure/Technical Specification requirements using data recorded at 100 percent power and during normal protection system surveillances.

The licensee stated that following the initial thermowell RTD cross calibration, the calibration reference will consist of the average of the RTD temperatures. The staff is concerned that the use of an average RTD value as a reference during cross calibration instead of a calibrated reference may lead to a net drift of the average temperature value indicated by the RTDs over time, should the installed RTDs drift systematically. The licensee indicated that RTD drift is random and with a total uncertainty less than ± 1.2 degrees specified in the submittal. Based on the above, the licensee felt that the cross calibration methodology utilized by the plant is acceptable. The staff concurred with the licensee's justification but will continue to evaluate this issue on a generic basis.

For LOCA events, the elimination of the RTD bypass system impacts the uncertainties associated with RCS temperature and flow measurement. The magnitude of the uncertainties are such that RCS inlet and outlet temperatures, thermal design flow rate and the steam generator performance data used in the LOCA analyses will be affected slightly. The evaluation of the slight increase in the T-average uncertainty has resulted in an estimated increase of 3°F for the large break LOCA PCT and a 2°F increase for the small break LOCA PCT. There is sufficient margin to 2200°F for both LOCA analyses to offset the estimated increase due to RTD bypass elimination.

For non-LOCA transients, only those transients which assume overtemperature-delta-T protection are potentially affected by changes in the RTD response time. As indicated in the Technical Specification Table 2.1-1, the overall response time remains unchanged from that assumed in previous safety analyses. Consequently, the conclusion of the safety analyses for these transients remains valid. The effects of the increase in T-average uncertainty by 0.3°F for the transients have been evaluated for all non-LOCA transients. The zero power transients are not affected by the change. The DNB related transients have been shown to be acceptable by using existing DNB margin. The FSAR safety analyses conclusions are unchanged and all applicable non-LOCA safety analysis acceptance criteria continue to be met.

2.3 Technical Specification Changes

The licensee proposed changes to the Technical Specifications which involve approval to increase the equivalent tube plugging limit from the current licensed value of 10% uniform plugging to a new licensed value of 15% average with a 20% peak in any one steam generator. The specific plugging limit is removed from the Technical Specifications, consistent with the Westinghouse Standard Technical Specifications. Also included is a decrease of approximately 1.5% in reactor coolant system total flow rate. Calculations of reactor trip system instrumentation trip setpoints are revised based on the reduced core flow rate. Replacement of the RTD bypass system results in revised Technical Specification allowable values and response times associated with the reactor protection system. The staff finds these Technical Specification changes acceptable based on the evaluations contained in Sections 2.1 and 2.2 above.

3.0 SUMMARY

The staff has reviewed the licensee's revised LOCA analysis and evaluation of the impact of the proposed changes on the non-LOCA safety analyses and finds that the proposed increase in steam generator plugging limit and the decrease in thermal design flow to be acceptable because (1) the requirements of 10 CFR 50.54 and Appendix K to 10 CFR Part 50 continue to be met and (2) the conclusions of the FSAR Chapter 15 safety analyses remain valid.

Based on our review, the staff concludes that the modified RTD system is not functionally different from the current system except for the use of three RTDs instead of one in each hot leg. The reactor protection or engineered safety features actuation systems will operate as before. The additional electronics for averaging the three T-hot RTD signals are to be qualified to the same level as the existing 7300 electronics. The isolation devices are also standard 7300 series equipment and were previously reviewed under WCAP-8892A. The RTD qualification will satisfy the requirement of 10 CFR 50.49.

To support the modifications required to eliminate the RTD bypass manifold system, the licensee proposed changes to the Technical Specifications. The revisions are a result of differences in the instrument and system uncertainties between the thermowell mounted RTD system and the bypass manifold temperature measurement arrangement. Evaluations performed by the licensee indicate that the uncertainty values are acceptable. The review by the staff supports this conclusion.

The licensee performed a detailed evaluation to determine the impact of the RTD bypass elimination on transients and accident analyses. The staff concludes that the FSAR safety analyses conclusions are unchanged and all applicable acceptance criteria continue to be met.

Based on the above, the staff finds the proposed plant modification to replace the RTD bypass manifold system with thermowell mounted, fast response, narrow range RTDs located directly in the RCS piping to be acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes the surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released off site, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration, and there has been no public comment on such finding. Accordingly, this amendment meets 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 this amendment.

5.0 CONCLUSION

The Commission made a proposed determination that this amendment involves no significant hazards consideration which was published in the FEDERAL REGISTER (55 FR 53067) on December 26, 1990, and consulted with the State of Alabama. No public comments or requests for hearing were received, and the State of Alabama did not have any comments.

The staff has concluded, based on the consideration discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Principal Contributors: K. Desai
E. Murphy
C. Doutt
S. Hoffman