December 6, 1990

Docket No. 50-364

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Mr. W. G. Hairston, III
Senior Vice President
Alabama Power Company
40 Inverness Center Parkway
Post Office Box 1295
Birmingham, Alabama 35201

Dear Mr. Hairston:

SUBJECT: ISSUANCE OF AMENDMENT NO. 79 TO FACILITY OPERATING LICENSE NO. NPF-8 REGARDING STEAM GENERATOR TUBE PLUGGING - JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2, (TAC NO. 77236)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. to Facility Operating License NPF-8 for the Joseph M. Farley Nuclear Plant, Unit 2. The amendment consists of changes to the Technical Specifications in response to your submittal dated July 31, 1990.

The amendment changes the Technical Specifications 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 our related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

Original Signed By:

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

Enclosures:
1. Amendment No. 79 to NPF-8
2. Safety Evaluation

cc w/enclosures:
See next page

DFO1

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PDR ADOCK 05000364
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OFC	:LA:PD21:DRPR	:PM:PD21:DRPR	:BC:SRXB:DST	:BC:EMCB:DET	BC:PRPB	:D:PD21:DRPR	:	:	:	:
NAME	: PAnderson	: SHoffman:sw	: RJones	:C.Y.Cheng	:CCunningham	:EAdensam	:	:	:	:
DATE	: 11/28/90	: 12/5/90	: 12/4/90	:12/4/90	:12/5/90	:12/5/90	:			

OFFICIAL RECORD COPY

AMENDMENT NO. 79 TO FACILITY OPERATING LICENSE NO. NPF-8 - FARLEY, UNIT 2

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

ALABAMA POWER COMPANY

DOCKET NO. 50-364

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 79
License No. NPF-8

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

   A. The application for amendment by Alabama Power Company (the licensee), dated July 31, 1990, 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-8 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 79 , are hereby incorporated in the license. The licensee 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

Original 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: December 6, 1990

OFC	LA:PD21:DRPR	PM:PD21:DRPR	OGC	D:PD21:DRPR			
NAME	PAnderson	SHoffman:sw	MYoung	EAdensam			
DATE	12/4/90	12/4/90	12/6/90	12/6/90			

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ATTACHMENT TO LICENSE AMENDMENT NO. 79

FACILITY OPERATING LICENSE NO. NPF-8

DOCKET NO. 50-364

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. 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
3/4 2-15	3/4 2-15

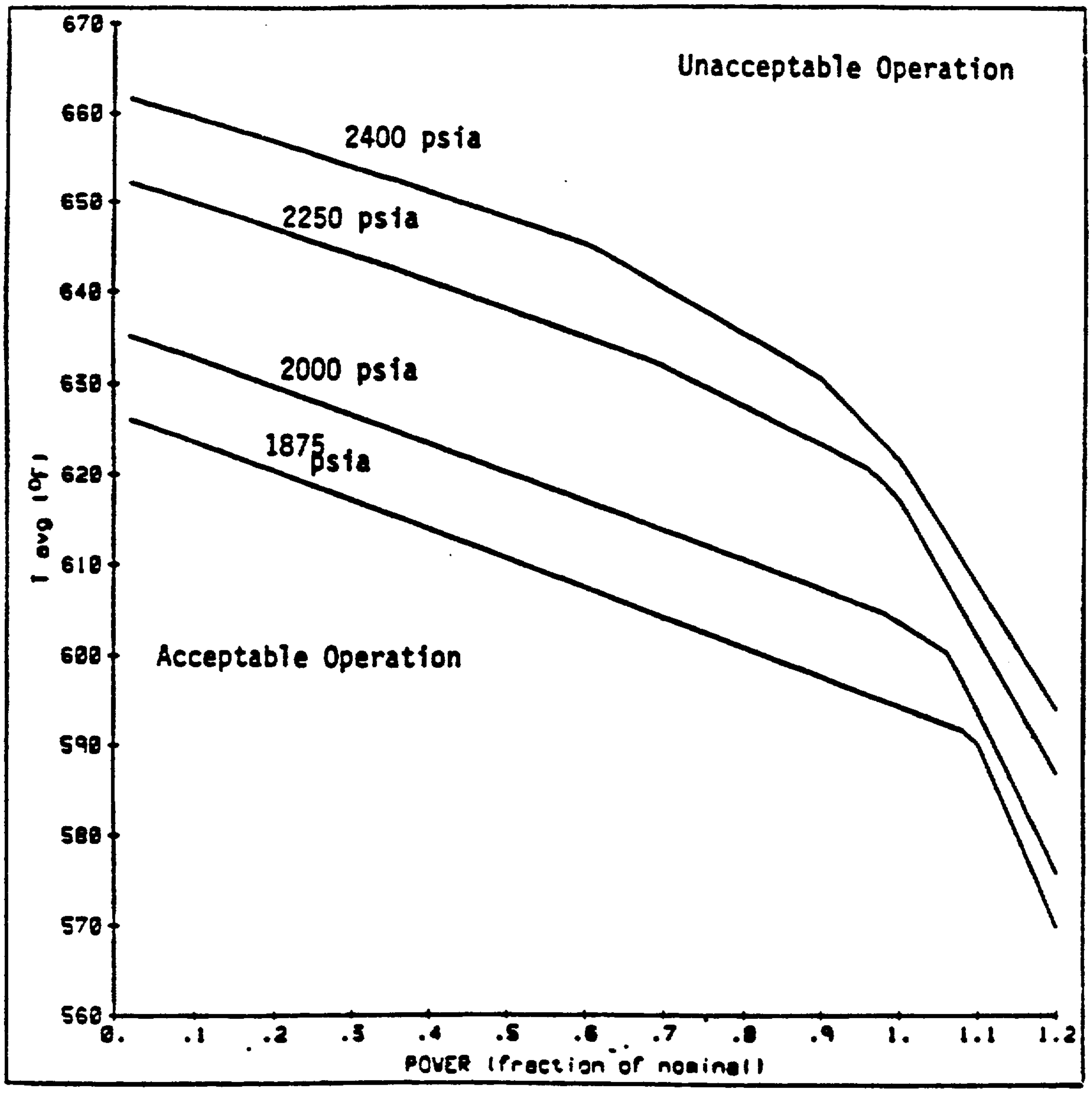

Figure 2.1-1 Reactor Core Safety Limit

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 - $\leq$ 25% of RATED THERMAL POWER High Setpoint - $\leq$ 109% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER High Setpoint - $\leq$ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 second	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 second
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 second	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 second
5. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 3
9. Pressurizer Pressure--Low	$\geq$ 1865 psig	$\geq$ 1855 psig
10. Pressurizer Pressure--High	$\leq$ 2385 psig	$\leq$ 2395 psig
11. Pressurizer Water Level--High	$\leq$ 92% of instrument span	$\leq$ 93% of instrument span
12. Loss of Flow	$\geq$ 90% of design flow per loop*	$\geq$ 89% 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 $\Delta T \leq \Delta T_o \left[K_1 - K_2 \frac{1 + \tau_1 S}{1 + \tau_2 S} (T - T') + K_3 (P - P') - f_1 (\Delta I)\right]$

where: 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.

S = Laplace transform operator, sec^{-1}.

Operation with 3 loops	Operation with 2 loops
K_1 = 1.18	K_1 = (values blank pending
K_2 = 0.0154	K_2 = NRC approval of
K_3 = 0.000635	K_3 = 2 loop operation)

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:

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS
NOTATION continued

(i) for $q_t - q_b$ between -35 percent and +9 percent, $f_1(\Delta I) = 0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds -35 percent, the ΔT trip setpoint shall be automatically reduced by 1.37 percent of its value at RATED THERMAL POWER.

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

where: 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 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 3 Loops in Operation	2 Loops in Operation
Reactor Coolant System T_{avg}	$\leq$ 581.2°F	**
Pressurizer Pressure	$\geq$ 2220 psia*	**
Reactor Coolant System Total Flow Rate	$\geq$ 261,600 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 79 TO FACILITY OPERATING LICENSE NO. NPF-8

ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-364

1.0 INTRODUCTION

By letter dated July 31, 1990, Alabama Power Company (APCo or the licensee) submitted a request for changes to the Joseph M. Farley Nuclear Plant, (Farley) Unit 2, Technical Specifications.

Farley, Unit 2, 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 as 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 2 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-12659, "Alabama Power, Joseph M. Farley Unit No. 2, Increased Steam Generator Tube Plugging and Reduced Thermal Design Flow Licensing Report," dated July 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:

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

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

2.0 EVALUATION

2.1 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. This brings the resultant PCT to 2073°F for Farley, Unit 2.

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 NRC 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-12659, Westinghouse Electric Corporation (Westinghouse) has identified what appears to be a new issue for older model Westinghouse steam generators (such as the Farley, Unit 2, 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 to support the subject license amendment and has 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 that November 18, 1990 letter, the licensee submitted a scoping analysis stating that relevant LBB parameters for Farley, Unit 2, 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." The licensee is currently performing detailed LBB analyses for Farley, Unit 2, which they have committed to provide to the staff by January 31, 1991. However, based on the above, the licensee concludes that the LBB methodology is applicable to the Farley, Unit 2 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 2. The staff will continue to pursue resolution of the generic concerns independent of Farley, Unit 2. Therefore, the staff finds that Farley, Unit 2, 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) calculates lower PCTs than the WFLASH evaluation model (WCAP-11145-P-A). The Farley WFLASH 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 2, 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 analysis (SGTR). The results of the SGTR analyses indicate that the primary-to-secondary break flow and atmospheric steam release via the ruptured steam generator increased as compared to the results of the current Farley, Unit 2, 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.

2.2 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 differences 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. As a result, 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.

2.3 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 from the current licensed value of 265,500 gpm to a new licensed value of 261,600 gpm. Calculations of reactor trip system instrumentation trip setpoints are revised based on the reduced core flow rate. 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 NRC 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 50.46 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.

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 34363) on August 22, 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 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: December 6, 1990

Principal Contributors: K. Desai
E. Murphy