October 26, 1987

Docket Nos. 50-348
and 50-364

DISTRIBUTION
See attached sheet

Mr. R. P. McDonald
Senior Vice President
Alabama Power Company
Post Office Box 2641
Birmingham, Alabama 35291-0400

Dear Mr. McDonald:

SUBJECT: ISSUANCE OF AMENDMENT NO. 73 TO FACILITY OPERATING LICENSE NO. NPF-2 AND AMENDMENT NO. 65 TO FACILITY OPERATING LICENSE NO. NPF-8 - JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2, REGARDING AN INCREASE IN STEAM GENERATOR TUBE PLUGGING LIMIT AND ASSOCIATED F_Q CHANGE (TAC NOS. 62283 AND 62284)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 73 to Facility Operating License No. NPF-2 and Amendment No. 65 to NPF-8 for the Joseph M. Farley Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your submittal dated August 25, 1986, superceded June 2, 1987, and supplemented September 16, and 23, 1987.

The amendments change the Technical Specifications to increase the steam generator tube plugging from 5 percent to 10 percent and to increase the heat flux hot channel factor coefficient, F_Q, slightly.

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

Sincerely,

Edward A. Reeves, Sr. Project Manager
Project Directorate II-1
Division of Reactor Projects I/II

Enclosures:
1. Amendment No. 73 to NPF-2
2. Amendment No. 65 to NPF-8
3. Safety Evaluation

cc: w/enclosures
See next page

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EReeves/sh
10/13/87

D:PD21:DRPR
EAdensam
10/14/87

Mr. R. P. McDonald
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Joseph M. Farley Nuclear Plant

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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. 73
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   A. The application for amendment by Alabama Power Company (the licensee), dated August 25, 1986, superceded June 2, 1987, and supplemented September 16 and 23, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act),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 set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   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:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 73, 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 and shall be implemented within 30 days of receipt of the amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

EGA for

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 26, 1987

LA:PD21:DRPR	PM:PD21:DRPR	OGC-B	D:PD21:DRPR
PAnderson	EReeves/dsf		EAdensam
10/13/87	10/13/87	10/16/87	10/26/87

ATTACHMENT TO LICENSE AMENDMENT NO. 73

TO 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 pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
2-2	2-2
3/4 2-4	3/4 2-4
B3/4 2-1	B3/4 2-1

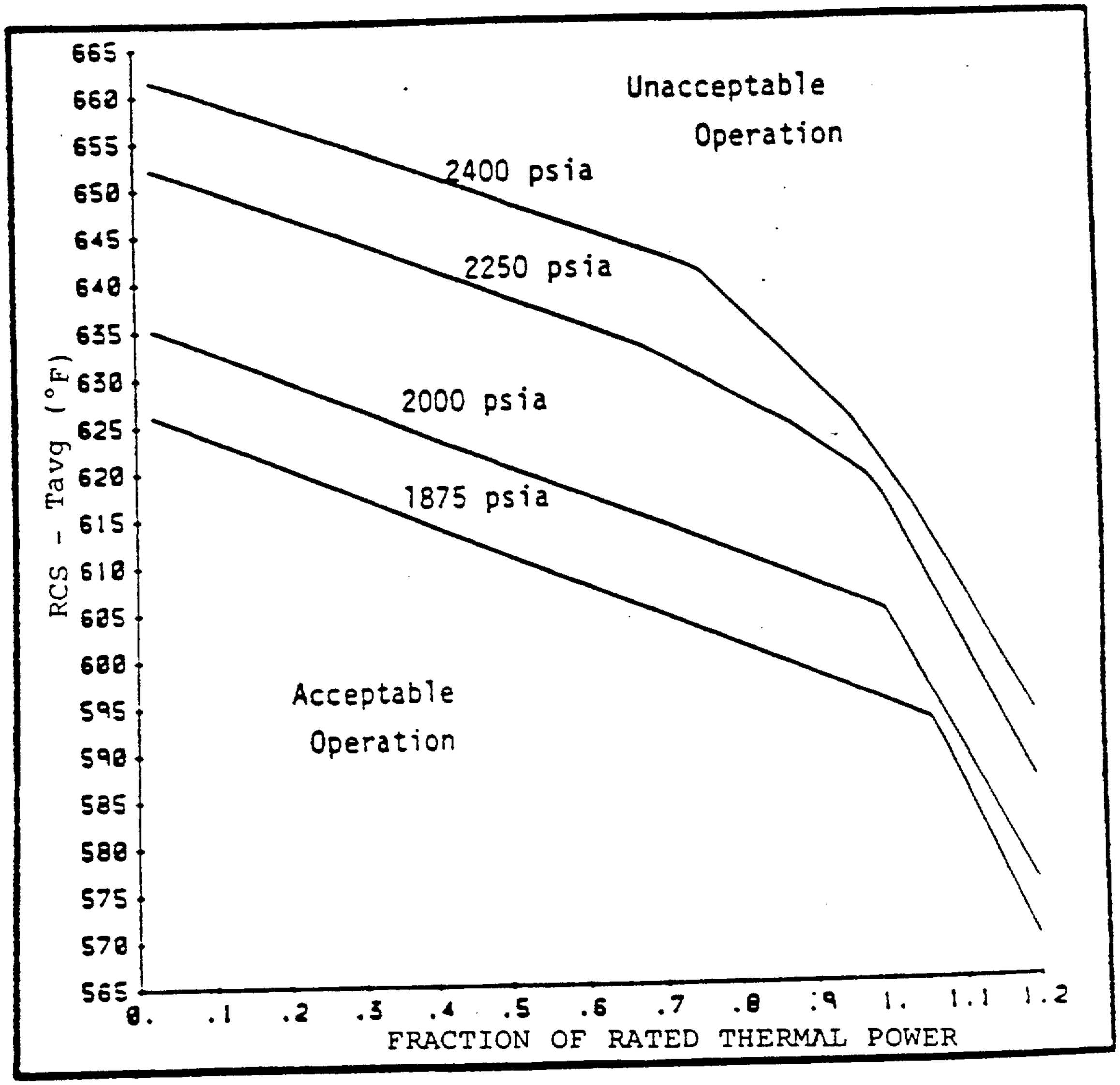

Figure 2.1-1 Reactor Core Safety Limit

Three Loops in Operation

Applicability: $\leq$ 10% Steam Generator Tube Plugging

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \left[\frac{2.32}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [4.64] [K(Z)] \text{ for } P \leq 0.5$$

where $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

and K(Z) is the function obtained from Figure (3.2-2) for a given core height location.

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower delta T Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower delta T Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a, above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core greater than or equal to 1.30 during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods and measurement uncertainty.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of 2.32 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

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. 65
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 August 25, 1986, superceded June 2, 1987, and supplemented September 16, and 23, 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 set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   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. 65, 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 and shall be implemented within 30 days of receipt of the amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

/s/ for

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 26, 1987

LA:PD21:DRPR	PM:PD21:DRPR	OGC-B	D:PD21:DRPR
PAnderson	EReeves/sh		EAdensam
10/13/87	10/13/87	10/16/87	10/26/87

ATTACHMENT TO LICENSE AMENDMENT NO. 65

TO FACILITY OPERATING LICENSE NO. NPF-8

DOCKET NO. 50-364

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
2-2	2-2
3/4 2-4	3/4 2-1
B3/4 2-1	B3/4 2-1

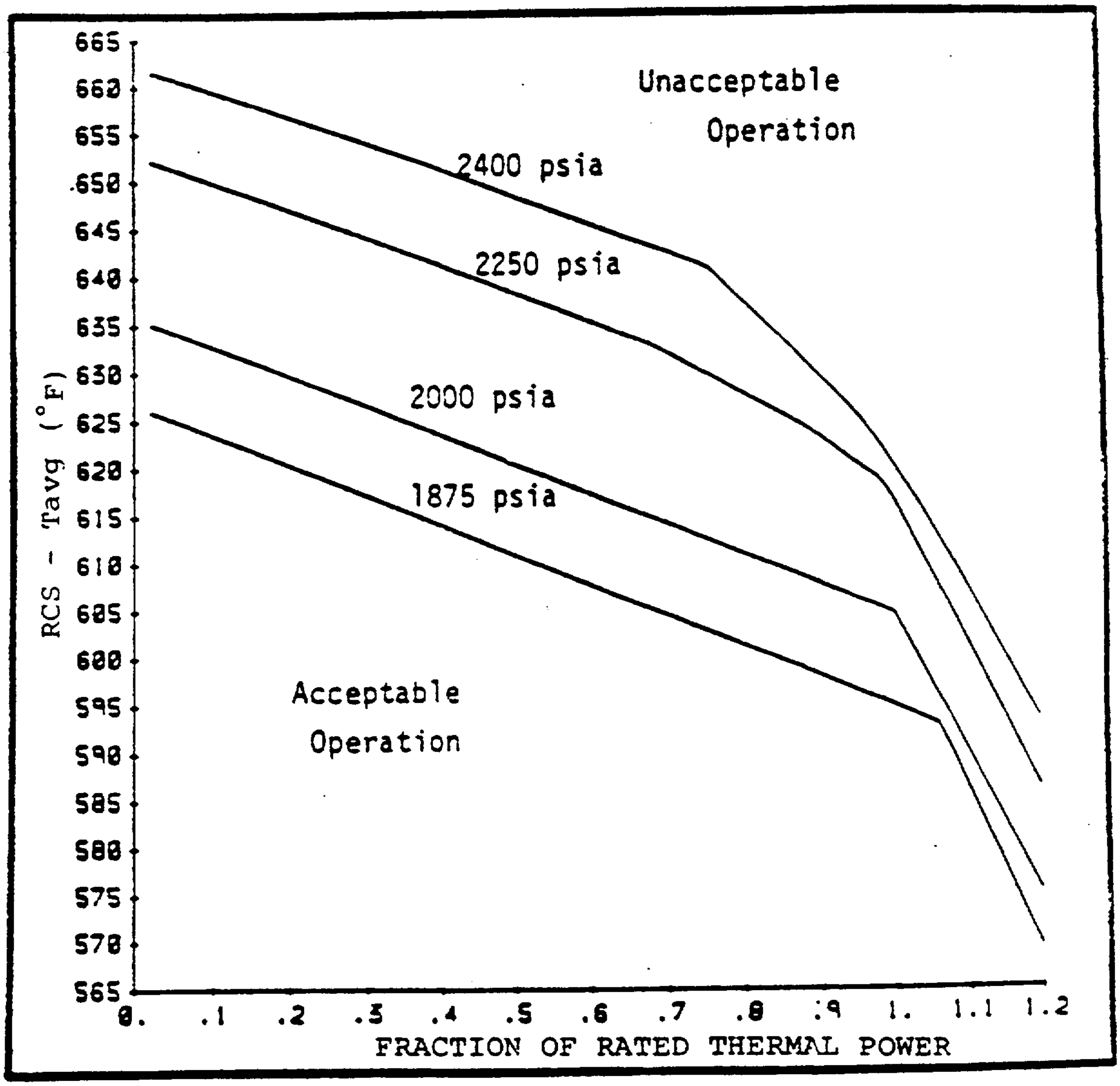

Figure 2.1-1 Reactor Core Safety Limit

Three Loops in Operation

Applicability: $\leq$ 10% Steam Generator Tube Plugging

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \left[\frac{2.32}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [4.64] [K(Z)] \text{ for } P \leq 0.5$$

$$\text{where } P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

and K(Z) is the function obtained from Figure (3.2-2) for a given core height location.

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower delta T Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower delta T Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a, above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core greater than or equal to 1.30 during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods and measurement uncertainty.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of 2.32 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 73 TO FACILITY OPERATING LICENSE NO. NPF-2

AND AMENDMENT NO. 65 TO FACILITY OPERATING LICENSE NO. NPF-8

ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-348 AND 50-364

1.0 INTRODUCTION

By letter dated August 25, 1986, superceded June 2, 1987, supplemented September 16, and 23, 1987, the Alabama Power Company (APCo, or the licensee), submitted a request for changes to the Joseph M. Farley Nuclear Plant, Units 1 and 2, Technical Specifications. This amendment request was noticed on October 7, 1986 (51 FR 36082) and July 15, 1987 (52 FR 26582). The supplements did not change the amendment requested or the determination noticed; therefore, the amendment was not renoticed a third time.

The Amendment would revise the Technical Specifications (TS) to allow an increase in the allowed steam generator (SG) tube plugging limit from 5% to 10% and an increase in the Heat Flux Hot Channel Factor (F_Q) limits. The F_Q change was from 2.31 to 2.32 for greater than 50% Rated Thermal Power (RTP) and from 4.62 to 4.64 for less than or equal to 50% RTP. The licensee had previously provided a sumbittal, dated August 25, 1986, which contained proposed TS changes and explanations of why the effects of the proposed changes on plant transients would not jeopardize safe operation of the plant. However, the staff advised the licensee to provide a reanalysis of the emergency core cooling system (ECCS) analysis for Farley Units 1 and 2 that supports the large break loss-of-coolant accident (LOCA) with a corrected BART code methodology. By letter dated June 2, 1987, the licensee superceded the August 25, 1986 submittal and addressed the concerns.

2.0 BACKGROUND

Farley Nuclear Plant currently has a steam generator tube plugging (SGTP) limit of 5% as shown on TS Figure 2.1-1. This limit is based on the Large Break LOCA/ECCS analysis in the FSAR Section 15.4 which assumes a 5% SGTP limit. Approximately 2.9% of the steam generator tubes have been plugged in Unit 1 and approximately 3.7% of the steam generator tubes have been plugged in Unit 2. This level of SGTP includes all row 1 tubes in each steam generator, which were done as a precautionary measure by the licensee. Based on the degradation identified during the last Unit 2 inspection, the expected tube plugging during the October 1987 refueling outage could exceed the current limit of 5%.

Therefore, APCo has proposed the change to increase the steam generator tube plugging limit to 10% to provide additional margin to the limit.

The licensee's original submittal, dated August 25, 1986, also provided a revised ECCS analysis for Farley Units 1 and 2. Changes in the analysis assumptions included an increase in SGTP from 5% to 10%, and an increase in F_Q from 2.31 to 2.32 for RTP above 50% and an increase in F_Q from 4.62 to 4.64 for RTP of 50% or less. The lower values for F_Q were the result of a penalty imposed by the NRC against the 1978 version of the Westinghouse ECCS Evaluation Model. We were informed by the licensee that the K(z) value had originally been evaluated for a F_Q of 2.32. This value remained the same with the previous reduction to F_Q of 2.31. Therefore, the change up to F_Q of 2.32 did not require a change for K(z). The present model, BART, no longer requires a penalty. However, Westinghouse had recently identified nonconservative assumptions in the BART model regarding the effects of control rod thimble filling during reflood and hot-assembly power effects. Reassessment of the overall BART model conducted by Westinghouse and described in WCAP-9561-P-A, Addendum 3, determined that other conservatisms contained in BART compensate for the nonconservative thimble filling and hot assembly assumptions. However, the staff required that licensing actions be supported by a reanalysis using revised versions of the BART code (WCAP-9561, Addendum 3, Revision 1) to comply with 10 CFR 50.46.

Accordingly, APCo resubmitted the proposed changes to the TS in a letter dated June 2, 1987, which supersedes the August 25, 1986 submittal. A new Large Break LOCA analysis was performed by Westinghouse for the Farley Nuclear Plant utilizing the 1981 Evaluation Model (WCAP-9220-P-A and WCAP-9221) with BASH (WCAP-10266, Revision 2). The use of the BASH methodology has been approved by the NRC staff in a letter dated November 13, 1986, from Mr. Charles E. Rossi to Mr. E. P. Rahe, Jr. (Westinghouse). In a telcon of August 20, 1987, the licensee reaffirmed that WCAP-9561, Addendum 3, Revision 1, was used as required in the evaluation. The NRC staff evaluation follows.

3.0 EVALUATION

Large Break LOCA

The analysis to meet the requirements of Appendix K and 10 CFR 50.46 for Large Break LOCA was performed using the Westinghouse Evaluation model with BART-1A (WCAP-9561-P-A, 1984 (Proprietary)) and BASH (WCAP-10266, Rev. 2 with Addenda, 1986 (Proprietary)) for a spectrum of break coefficients. Subsequent to the completion of the Farley Large Break LOCA analysis with BASH, Westinghouse notified Alabama Power Company of enhancements to the BASH code and methodology that were made to improve the reliability and performance of the code in certain circumstances. The modifications to the BASH methodology which incorporate these enhancements (described in Addendum 2 to WCAP-10266, Revision 2) were submitted to the NRC in letter NS-NRC-87-3212, dated March 26, 1987. This topical report has been reviewed by the staff and is approved for application to Farley Units 1 and 2. APCo together with Westinghouse has evaluated the impact of the BASH code modifications on the Farley Large Break LOCA analysis with BASH in Attachment 4 to the June 2, 1987, APCo letter and has concluded the Farley analysis remains conservative and bounding.

The new Large Break LOCA analysis assumes an F_Q of 2.40. The present F_Q coefficient of 2.31 for greater than 50% RTP and 4.62 for less than or equal to 50% RTP was required as a result of penalties assessed by the NRC against the 1978 version of the Westinghouse ECCS Evaluation Model.

Since the current Small Break LOCA analysis assumes an F_Q of 2.32 and the proposed increase in F_Q is conservatively bounded by the assumptions of the non-LOCA transient analyses, the proposed changes to increase the F_Q coefficients of TS 3.2.2 to 2.32 for greater than 50% RTP and 4.64 for less than or equal to 50% RTP are consistent with the design/licensing basis for Farley Nuclear Plant.

The fuel parameters used as input for the LOCA analysis were generated using the Revised PAD Thermal Safety Model, WCAP-8720, Addendum 2, which we approved by letter from C.O. Thomas (NRC) to E. P. Rahe, Jr. (Westinghouse), dated December 9, 1983. The hydraulic analyses and core thermal transient analyses for the Joseph M. Farley Large Break LOCA analysis were performed using 102 percent of licensed NSSS core power, 2652 Mwt. Other pertinent assumptions included a 10% SGTP level, minimum and maximum safeguards ECCS capabilities, 17 x 17 standard Westinghouse fuel design, which is the current design for both Farley units, and an upflow barrel-baffle configuration. The upflow barrel-baffle configuration was previously shown to represent a small peak clad temperature (PCT) penalty; hence, the use of this configuration is conservative and bounding on both units. This analysis also incorporated a conservative total reactor coolant system flow (1% below TS limit). Pertinent input parameters are listed below:

INPUT PARAMETERS

NSSS Power, MWt, 102% of licensed power	2652
Peak Linear Power, kw/ft, 102% of design	12.49
Peaking Factor (At Design Rating)	2.40
Hot Channel Enthalpy Rise Factor	1.62
Accumulator Water Volume (Cubic Feet per Tank)	1025.0
Accumulator Pressure, psi	600.0
Number of Safety Injection Charging Pumps Operating (Min ECCS/Max ECCS)	2/3 *
Steam Generator Tubes Plugged	10% (uniform)**

*Minimum safeguards analysis assumes 2 charging pumps are operating and one RHR pump. Maximum safequards analysis assumes three charging pumps and two RHR pumps are operating.

**Uniform 10% Steam Generator Tube Plugging assumes 10% SG tubes plugged in each steam generator and corresponds to the worst plugging level in any steam generator and will bound all combinations of non-uniform plugging as long as no one steam generator plugging level exceeds 10%.

Of the three break sizes evaluated, C_D= 0.4, C_D= 0.6, and C_D= 0.8, the C_D= 0.4 break with minimum ECCS safeguards proved to be the limiting (highest PCT) case. The resulting peak clad temperature was 2013°F, which is well below the 2200°F allowable limit.

APCo presented the following conclusions from their analysis which demonstrate that for breaks up to and including the double ended severance of a reactor coolant pipe, the ECCS design at Farley Nuclear Plant will meet the acceptance criteria as presented in 10 CFR 50.46. These are:

1. The calculated peak clad temperature does not exceed 2200°F based on a large break LOCA total peaking factor of 2.40 and a hot channel enthalpy rise factor of 1.62.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of Zircaloy in the reactor.

3. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling.

4. The cladding oxidation limits of 17% are not exceeded during or after quenching.

5. The core temperature is reduced and the decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

Small Break LOCA

As previously stated, APCo requested an increase in SGTP limit from 5% to 10%. In justifying the increase to 10% tube plugging, the licensee stated in the June 2, 1987, letter that there is evidence that for low steam generator plugging levels (up to 20%), Small Break LOCA transients would not be affected by the proposed tube plugging limit. We questioned the licensee about the evidence. In response, the licensee explained in letters dated August 18, and September 16, 1987, their conclusion that for up to 20% tube plugging there would be no affect in the Small Break LOCA analysis. This explanation is based upon an evaluation performed in 1985 for the Westinghouse designed Almaraz plant which is similar to the Farley design. Both plants are of identical Westinghouse vessel design. Common features include three coolant loops, 157 fuel assemblies, standard fuel design (0.374 inch OD), 48 control rods, 144 inch active fuel length, and Model 93 reactor coolant pumps. A few small deviations exist with respect to operating parameters, but these are insignificant for comparison of response for a small break accident. For example, reactor power is 2686 MWt for Almaraz; whereas, reactor power for Farley is 2652 MWt. The licensee concluded that evaluations based on the Almaraz plant for establishing trends and sensitivities are equally applicable for the Farley units.

Three specific effects of SGTP were identified by the licensee and evaluated for the Almaraz plant as follows:

1) the impact of the reduction of the steam generator tube area on the small break transient as it relates to the ability to transfer heat from primary to secondary and, thus, dissipate core stored energy and decay heat,

2) the effect of changes to operating temperatures (primary and secondary) as a result of SGTP, and

3) the effect that could be exerted on the draining of the steam generator tubes as this has a direct effect on water inventory in the vessel and potential for core uncovery.

The reference evaluation of the Almaraz plant considered other studies made on small break LOCA. These studies, "Simulation of Small Break Type Behavior of PUN and SPES using the NORTRUMP Code" and "Limiting Counter Current Flow Phenomenon in Small Break LOCA Transients," were included in Proceedings of the Specialists meeting on Small Break LOCA Analyses in LWR's, Pisa, Italy, June 1985. It was concluded that no effect would be expected in the Small Break analysis for SGTP levels up to 20% for the three relevant phenomena identified.

These phenomena are summarized as follows:

1) only a small portion of the steam generator tube heat transfer area is sufficient in a small break transient to provide an effective heat sink to the primary side,

2) operating temperature differences as a result of plugging disappear shortly after the break because the secondary side pressure reaches steam generator safety valve setpoints almost immediately, and

3) the counter current flow limit (CCFL) characteristics would be such that the CCFL would still be dominant and limiting in the inclined pipe connecting the steam generator inlet plenum to the hot leg for SGTP levels up to 20%. For steam generator tube plugging levels beyond 20%, a CCFL calculation in steam generator tube locations would increase. This would reduce the dominance of the CCFL in the inclined pipe, in which case the plugging level would exert an influence.

The licensee stated that Farley's current Small Break LOCA analysis is based on the WFLASH code. From the reference evaluation of 1985 for the Almaraz plant, they conclude that the effect of SGTP would not be seen in WFLASH analyses because WFLASH does not take credit for the CCFL phenomenon.

Following the incident at Three Mile Island, Unit 2, Westinghouse and the Westinghouse Owners Group developed the NOTRUMP computer codes (WCAP-10079-P-A and WCAP-10054-P-A (both Proprietary), August 1985) as the new Small Break LOCA evaluation model, which the NRC staff approved in May 1985, to meet the requirements of NUREG-0737, Section II.K.3.30. Small break LOCA analysis performed using NOTRUMP for NUREG-0737 demonstrated that, in general, the NOTRUMP evaluation model calculated lower peak cladding temperatures than the WFLASH evaluation model. This allowed the WFLASH analyses contained in the Joseph M. Farley Final Safety Analysis Report (FSAR) to remain the licensing basis analysis of record in accordance with NRC Generic Letter 83-35.

APCo concluded that the reference Almaraz plant evaluation is directly applicable to the Farley plants and that for a 10% SGTP there would be no adverse effect on the WFLASH small break analysis of record. By projecting effects, if analyzed with NOTRUMP, APCo stated that minimal SGTP effects in PCT would be expected to be observable at a level near 15% to as high as 20% SGTP. But this would be insignificant compared to the PCT improvement that would be expected by applying the NOTRUMP Evaluation Model. Therefore, APCo concludes that the results of the Farley analysis of record continue to be bounding.

Based on the preceding evaluation, the NRC staff concludes that for the SGTP, as APCo requested (up to 10%), the proposed changes to the TS are acceptable and will not impact or invalidate the current licensing basis for the Small Break LOCA analysis as represented in the Farley FSAR. In addition, the peak cladding temperature for the Small Break LOCA is 193°F (1820°F vs. 2013°F) lower than that for the Large Break LOCA; thereby, providing the NRC staff with additional assurance that the criteria of 10 CFR 50.46 are satisfied.

Reactor Coolant System Flow

The licensee stated that an analysis was performed to determine the effects on the core flow due to the increase of 5% in SGTP. The analysis determined that the increase to 10% SGTP would not decrease reactor coolent system (RCS) flow, below the thermal design flow (TDF) for the Farley Nuclear Plant.

In response to NRC staff questions, the licensee stated that Farley TDF values for Units 1 and 2 are 265,500 gpm for each unit. The licensee also stated that the latest measured total RCS flow values were 283,963 gpm for Unit 1 and 285,767 gpm for Unit 2. Since the flow measurement uncertainty for the Farley Units is 3.5%, these measured flow values correspond to actual flows of at least 274,024 gpm for Unit 1 and 275,765 gpm for Unit 2. These values are well above the minimum flow requirement of 265,500 gpm in the TS. With 10% of the steam generator tubes plugged, the calculated RCS flow is 94,500 gpm per loop or 283,500 gpm total for Unit 1 and 93,800 gpm per loop or 281,400 gpm total for Unit 2.

These values are based on best estimate flow calculations. These flows are acceptable since all are above the minimum flow requirements. For reference only, the licensee stated that the calculated RCS flow with 0% steam generator tube plugging is 289,200 gpm total for Unit 1 and 287,100 gpm total for Unit 2.

Non-LOCA Accidents

The licensee stated that, since the non-LOCA departure from nucleate boiling (DNB) transients are based on TDF, which remains applicable, a 10% SGTP limit was determined to have no impact on the non-LOCA DNB transients. The effect of 10% SGTP upon those non-LOCA accidents which are not DNB related, or for which DNB is not the only safety criterion was also evaluated. The only accident of this group which is affected by 10% SGTP is the boron dilution analysis. An input to the boron dilution analysis for Modes 1 and 2 is the RCS active volume, i.e., the total RCS volume minus the volumes of the pressurizer, the pressurizer surge line, the dead volume of the reactor vessel head, and plugged steam generator tubes. Reduction of the RCS active volume is directly proportional to the reduction in operator response time for the boron dilution event described in the Farley FSAR. APCo estimated that 10% tube plugging will reduce the Farley active volume by approximately 4%. However, the licensee stated that, from the boron dilution analysis done for Farley, it can be shown that the RCS active volume can be reduced by more than 4%; thus, the required operator action time (of at least 15 minutes) would still be adequate. Therefore, the 10% SGTP limit for the Farley Nuclear Plant will not change the conclusions of the safety analysis.

In response to a question on the effect of tube plugging on pump coastdown, the licensee in a letter dated October 13, 1986, stated that an analysis was performed which determined that a 10% SGTP limit would not decrease RCS flow below the TDF for the Farley Nuclear Plant. The initial RCS flow used in the pump coastdown analysis is based on TDF which must be maintained to comply with the TS. Therefore, the modeled pump coastdown used in the current non-LOCA analysis will not become more severe. The pump coastdown for, Farley Nuclear Plant is modeled using the PHOENIX computer code. A description of the model is found in WCAP-7993, "Calculation of Flow Coastdown After Loss of Reactor Coolant Pump (PHOENIX Code)."

Technical Specification Changes

The Technical Specification changes for Farley Units 1 & 2 are as follows:

1. In Figure 2.1-1, Reactor Core Safety Limit, applicability was changed to account for the increase in the SGTP limit from 5% to 10%. This was discussed in the evaluation above and is acceptable.

2. In TS 3/4.2.2 Heat Flux Hot Channel Factor - $F_Q(z)$ the value for F_Q was changed from 2.31 to 2.32 for greater than 50% RTP and from 4.62 to 4.64 for less than original to 50% RTP. These changes are acceptable for the reasons explained herein.

3. In TS 3/4.2.1, Bases, Axial Flux Difference, an editorial change was made to account for the new F_Q value of 2.32 instead of 2.31. This is acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to installation or use of a facility component located within the restricted areas, as defined in 10 CFR Part 20, and change the surveillance requirements. The staff has determined that these amendments involve 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 these amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, these 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.

5.0 CONCLUSION

The Commission made a proposed determination that this amendment involves no significant hazards consideration which was published in the Federal Register (51 FR 36082) on October 7, 1986, and (52 FR 26582) on July 15, 1987, 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, 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: October 26, 1987

AMENDMENT NO. 73 TO FACILITY OPERATING LICENSE NO. NPR-2 - FARLEY, UNIT 1
AMENDMENT NO. 65 TO FACILITY OPERATING LICENSE NO. NPF-8 - FARLEY, UNIT 2

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