DEC 3 0 1983

Docket Nos. 50 -348
and 50-364

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Mr. F. L. Clayton
Senior Vice President
Alabama Power Company
P. O. Box 2641
Birmingham, Alabama 35291

Dear Mr. Clayton:

The Commission has issued the enclosed Amendment No. 37 to Facility Operating License No. NPF-2 and Amendment No. 27 to NPF-8 for the Joseph M. Farley Nuclear Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application dated October 13, 1983.

The amendments consist of changes to the Technical Specifications to allow operation with a slightly positive moderator temperature coefficient at low power levels and an increased hot channel factor limit below full power.

A copy of our Safety Evaluation is also enclosed.

Sincerely,

/s/ Edward A. Reeves

Edward A. Reeves, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 37 to NPF-2
2. Amendment No. 27 to NPF-8
3. Safety Evaluation

cc: w/enclosures:
See next page

| | | | | |
|---|---|---|---|---|
| ORB#1:DL | ORB#1:DL | C-ORB#1:DL | OELD | AD:OR:DL |
| CParrish | EReeves;ps | SVarga | | GLainas |
| 12/20/83 | 12/20/83 | 12/20/83 | 12/29/83 | 12/29/83 |

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Mr. F. L. Clayton
Alabama Power Company

Joseph M. Farley Nuclear Plant
Units 1 and 2

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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 NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 37
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 13, 1983, 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. 37 , 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. However, the changes to the Technical Specifications will be effective prior to entry into Mode 2 following the fifth refueling outage scheduled to start January 10, 1984.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 30, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 37

FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

Revised Appendix A as follows:

| Remove | Insert |
|---|---|
| 2-2 | 2-2 |
| 2-8 | 2-8 |
| 2-9 | 2-9 |
| B2-1 | B2-1 |
| 3/4 1-4 | 3/4 1-4 |
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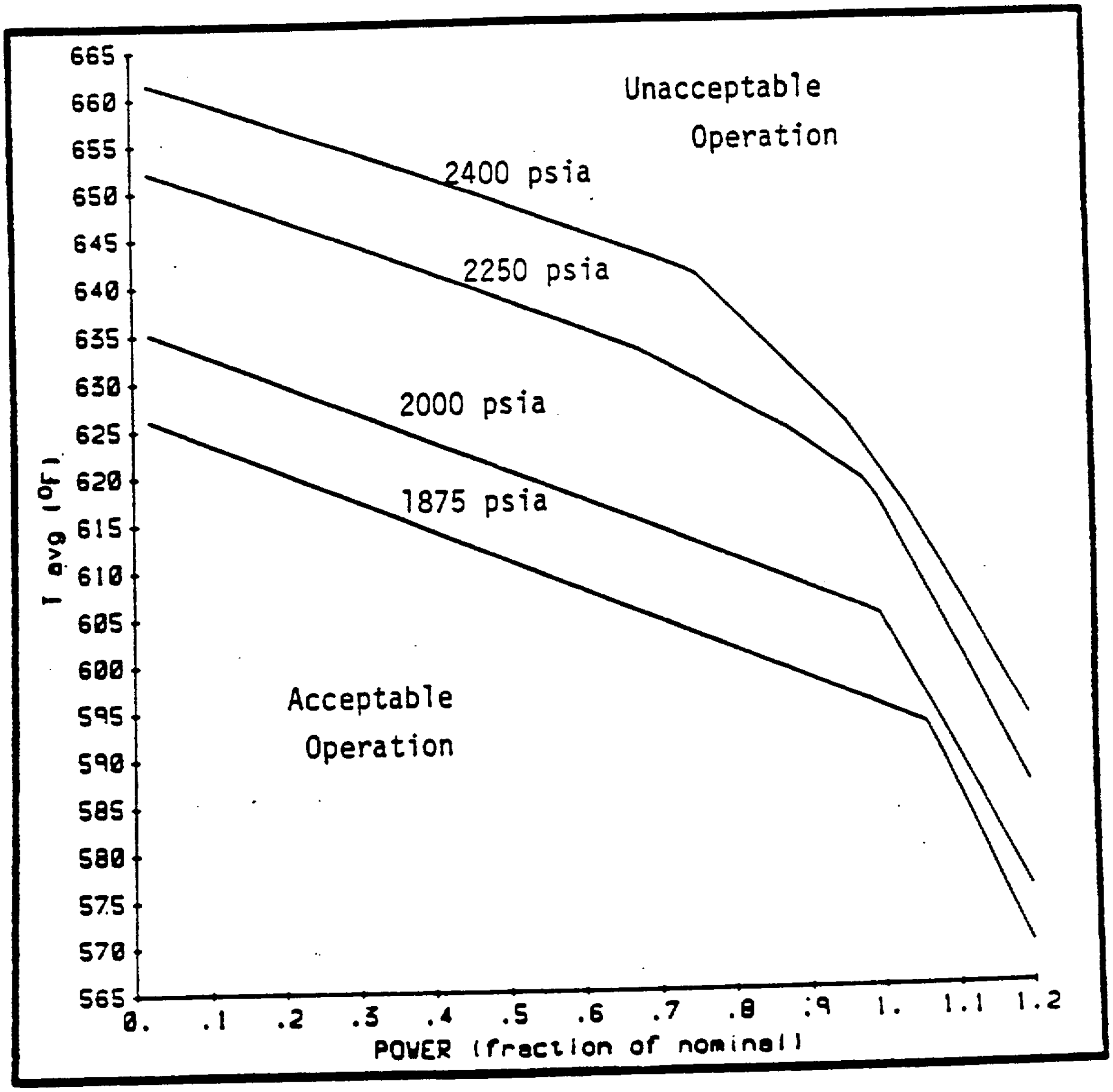


Figure 2.1-1 Reactor Core Safety Limit

Three Loops in Operation

Applicability: ≤ 5% Steam Generator Tube Plugging

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: $\Delta T_o$ = Indicated $\Delta T$ 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

$\tau_1$ & $\tau_2$ = Time constants utilized in the lead-lag controller for $T_{avg}$ $\tau_1$ = 30 secs,

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

| Operation with 3 Loops | Operation with 2 Loops |
|---|---|
| $K_1$ = 1.22 | $K_1$ = (values blank pending |
| $K_2$ = 0.0154 | $K_2$ = NRC approval of |
| $K_3$ = 0.000635 | $K_3$ = 2 loop operation) |

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

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 $\Delta T$ 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 $\Delta T$ trip setpoint shall be automatically reduced by 1.60 percent of its value at RATED THERMAL POWER.

Note 2: Overpower $\Delta T \leq \Delta T_o \left[ Ky - K_5 \frac{\tau_3 S}{1+\tau_3 S} - K_6 (T-T'') - f_2(\Delta I) \right]$

where: $\Delta T_o$ = Indicated $\Delta T$ at RATED THERMAL POWER

$T$ = Average temperature, °F

$T''$ = Reference $T_{avg}$ at RATED THERMAL POWER (Calibration temperature for $\Delta T$ 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

## 2.1 SAFETY LIMITS

BASES

### 2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which wold result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figures 2.1-1 and 2.1-2 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

These curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.55\ [1+0.3\ (1-P)]$$

where P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

## LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less than or equal to $0.5 \times 10^{-4}$ delta k/k/°F for the all rods withdrawn, beginning of cycle life (BOL), below 70 % THERMAL POWER condition. Less than or equal to 0 delta k/k/°F at or above 70% THERMAL POWER.

b. Less negative than $-3.9 \times 10^{-4}$ delta k/k/°F for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.3.a - MODES 1 and 2* only#
Specification 3.1.1.3.b - MODES 1, 2 and 3 only#

ACTION:

b. With the MTC more positive than the limit of 3.1.1.3.a above, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than 0 delta k/k/°F within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

3. In lieu of any other report required by Specification 6.9.1, a Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of 3.1.1.3.b above, be in HOT SHUTDOWN within 12 hours.

*With $K_{eff}$ greater than or equal to 1.0
#See Special Test Exception 3.10.3

## 3/4.2.3 NUCLEAR HOT CHANNEL FACTOR - $F^N_{\Delta H}$

### LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$F^N_{\Delta H} \leq 1.55\ [1 + 0.3\ (1-P)]\ [1-RBP(BU)]$

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

RPB(BU) = Rod Bow Penalty as a function of region average burnup as shown in Figure 3.2-3, where a region is defined as those assemblies with the same loading date (reloads) or enrichment (first cores).

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a or b, above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL power and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

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

ALABAMA POWER COMPANY

DOCKET NO. 50-364

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 27
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 October 13, 1983, 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-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. 27 , 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. However, the changes to the Technical Specifications will be effective prior to entry into Mode 2 following the third refueling outage scheduled to start December 4, 1984.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 30, 1983

<u>ATTACHMENT TO LICENSE AMENDMENT NO. 27</u>

<u>FACILITY OPERATING LICENSE NO. NPF-8</u>

<u>DOCKET NO. 50-364</u>

Revised Appendix A as follows:

| Remove | Insert |
|---|---|
| 2-2 | 2-2 |
| 2-8 | 2-8 |
| 2-9 | 2-9 |
| B2-1 | B2-1 |
| 3/4 1-4 | 3/4 1-4 |
| 3/4 2-8 | 3/4 2-8 |

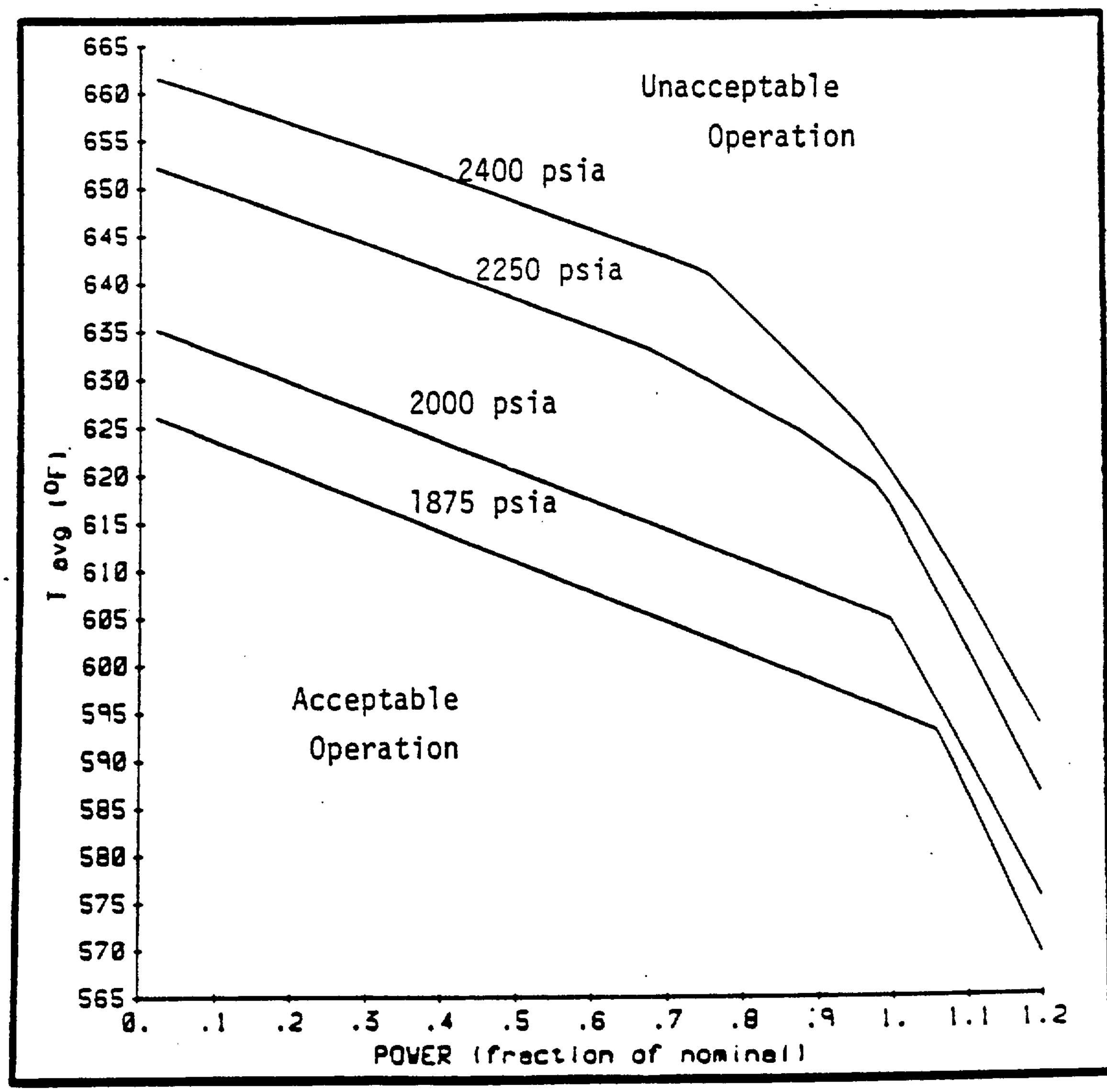


Figure 2.1-1 Reactor Core Safety Limit

Three Loops in Operation

Applicability: $\leq$ 5% Steam Generator Tube Plugging

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: $\Delta T_o$ = Indicated $\Delta T$ 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

$\tau_1$ & $\tau_2$ = Time constants utilized in the lead-lag controller for $T_{avg}$ $\tau_1$ = 30 secs,

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

Operation with 3 Loops

$K_1$ = 1.22

$K_2$ = 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)

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

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 $\Delta T$ 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 $\Delta T$ trip setpoint shall be automatically reduced by 1.60 percent of its value at RATED THERMAL POWER.

Note 2: Overpower $\Delta T \leq \Delta T_o \left[Ky - K_5 \frac{\tau_3 S}{1+\tau_3 S} - K_6 (T-T") - f_2 (\Delta I)\right]$

where: $\Delta T_o$ = Indicated $\Delta T$ at RATED THERMAL POWER

$T$ = Average temperature,°F

$T"$ = Reference $T_{avg}$ at RATED THERMAL POWER (Calibration temperature for $\Delta T$ 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

## 2.1 SAFETY LIMITS

BASES

### 2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which wold result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figures 2.1-1 and 2.1-2 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

These curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.55\ [1+0.3\ (1-P)]$$

where P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less than or equal to $0.5 \times 10^{-4}$ delta k/k/°F for the all rods withdrawn, beginning of cycle life (BOL), below 70 % THERMAL POWER condition. Less than or equal to 0 delta k/k/°F at or above 70% THERMAL POWER.

b. Less negative than $-3.9 \times 10^{-4}$ delta k/k/°F for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.3.a - MODES 1 and 2* only#
Specification 3.1.1.3.b - MODES 1, 2 and 3 only#

ACTION:

b. With the MTC more positive than the limit of 3.1.1.3.a above, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than 0 delta k/k/°F within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

3. In lieu of any other report required by Specification 6.9.1, a Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of 3.1.1.3.b above, be in HOT SHUTDOWN within 12 hours.

*With $K_{eff}$ greater than or equal to 1.0
#See Special Test Exception 3.10.3

## 3/4.2.3 NUCLEAR HOT CHANNEL FACTOR - $F^N_{\Delta H}$

### LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq 1.55\ [1 + 0.3\ (1-P)]\ [1-RBP(BU)]$$

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

RPB(BU) = Rod Bow Penalty as a function of region average burnup as shown in Figure 3.2-3, where a region is defined as those assemblies with the same loading date (reloads) or enrichment (first cores).

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a or b, above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL power and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 37 TO FACILITY OPERATING LICENSE NO. NPF-2

AND AMENDMENT NO. 27 TO FACILITY OPERATING LICENSE NO. NPF-8

ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-348 AND 50-364

## INTRODUCTION

By letter dated October 13, 1983, Alabama Power Company, licensee for Joseph M. Farley Nuclear Plant Units 1 and 2, submitted a request for an amendment of the Technical Specifications for both units. The Technical Specification changes are intended to accommodate a slightly positive moderator temperature coefficient (MTC) below 70 percent of rated power and an increased enthalpy hot channel factor ($F^N_{\Delta H}$) limit below full power. The licensee's submittal included a safety analysis for operation of the Farley units with new core safety limits which were established as a result of these proposed changes.

## DISCUSSION AND EVALUATION

### A. POSITIVE MODERATOR TEMPERATURE COEFFICIENT

The current MTC Technical Specification requires a non-positive value at all power levels. The proposed change allows a slightly positive MTC ($+0.5 \times 10^{-4}$ $\Delta k/k/°F$) below 70 percent of rated power and a non-positive value at or above 70 percent of rated power. Therefore, those transients and accidents which are sensitive to a positive MTC were reanalyzed. In general, these are transients which cause an increase in the reactor coolant temperature such as an uncontrolled rod cluster control assembly (RCCA) withdrawal, uncontrolled boron dilution, partial loss of forced reactor coolant flow, loss of external electrical load and/or turbine trip, accidental depressurization of the reactor coolant system (RCS), complete loss of forced reactor coolant flow, single reactor coolant pump locked rotor, and RCCA ejection.

B. NUCLEAR HOT CHANNEL FACTOR CHANGE

The nuclear hot channel factor is limited by the following relationship:

$$F^N_{\Delta H} \leq 1.55\ [1+K\ (1-P)]\ \ [1-RBP\ (BU)]$$

where P is the fraction of full power, RBP is the rod bow penalty, and K is a power correction constant. The proposed change will increase K from 0.2 to 0.3 and its effect on the accident analysis is realized through changes to the core safety limits. Since the overtemperature and overpower $\Delta T$ trip setpoints are based on the core safety limits, the setpoints must be verified as being applicable to the new core safety limits. The licensee has verified that the overpower and overtemperature $\Delta T$ protection setpoints are not affected by the proposed $F_{\Delta H}$ multiplier change. This also has been shown to be true for other Westinghouse cores where this power correction has been recently approved. However, since there exists excess conservatism between the current setpoints and the new core safety limits, the licensee has also proposed to modify the overpower and overtemperature $\Delta T$ setpoints in order to provide improved operating capability. Since there are no FSAR transients which rely on the overpower $\Delta T$ trip, only those transients which rely upon the overtemperature $\Delta T$ trip were reanalyzed by the licensee. These are the RCCA withdrawal at power, loss of external load, and RCS depressurization incidents.

C. REANALYZED TRANSIENTS

The reanalyzed transients which have been mentioned in the preceding sections as being impacted by the proposed changes in MTC or overtemperature $\Delta T$ trip setpoint were reviewed. The analysis of these transients used approved methods and computer programs consistent with those employed in the FSAR. In all cases the results indicate that the safety criteria are met. That is, peak fuel and clad

**temperatures remained acceptable, minimum DNBR remained above the limit value of 1.30, and/or reactor coolant system pressure remained below 110% of design pressure. The results of the reanalyzed transients are, therefore, acceptable.**

D. TECHNICAL SPECIFICATION CHANGES

(1) The MTC limits in Specification Limiting Condition of Operation (LCO) 3.1.1.3 have been changed to less than or equal to $0.5X10^{-4}\Delta k/k^{o}F$ for the all rods out, beginning of cycle, below 70 percent thermal power condition and to less than or equal to 0 at or above 70 percent thermal power. Westinghouse has performed the necessary transient and accident reanalyses based on the new MTC values and has shown that the limiting conditions and safety criteria are met. The change is, therefore, acceptable.

(2) The 0.2 partial power multiplier in Secifications Bases B.2.1.1 and LCO 3.2.3 has been changed to 0.3 in order to allow an increased hot channel factor below full power. Historically, increasing the allowable $F^{N}_{\Delta H}$ with decreasing power has been permitted for previously approved Westinghouse designs. The Westinghouse safety analysis has shown that the current Farley safety analyses are not affected by this change. The change is, therefore, acceptable.

(3) The overpower and overtemperature $\Delta T$ setpoints defined in Specification Table 2.2-1 were revised in order to optimize the available operational margin that exists between the setpoints and the new core safety limits (Figure 2.1-1) which were established due to the above-mentioned $F^{N}_{\Delta H}$ change. The constants in these $\Delta T$ setpoints were modified according to the setpoint methodology accepted for all previous Westinghouse cores and described in WCAP-8745, "Design Bases for the Thermal Overpower $\Delta T$ and Thermal Overtemperature $\Delta T$ Trip Functions." Those incidents which rely on these trips were reanalyzed to confirm that the limiting conditions and safety criteria are met. Therefore, the revised overpower and overtemperature $\Delta T$ protection setpoint modifications are acceptable.

SAFETY SUMMARY

We have reviewed the proposed changes to the Farley Units 1 and 2 Technical Specifications allowing operation with a positive moderator temperature coefficient at low power levels and an increased hot channel factor limit below full power. We conclude that reasonable assurance has been provided that these modifications will not pose a threat to the health and safety of the public and that the proposed Technical Specification changes are, therefore, acceptable.

ENVIRONMENTAL CONSIDERATION

We have determined that the two amendments do not authorize changes in effluent types or total amounts, nor increases in power levels, and will not result in an significant environmental impact. Having made this determination, we have further concluded that the amendments involve actions which are insignificant from the standpoint of environmental impact and, pursuant to 10 CFR 51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

CONCLUSION

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

Dated: DEC 3 0 1983

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