Mr. D. N. Morey
Vice President - Farley Project
Southern Nuclear Operating
Company, Inc.
Post Office Box 1295
Birmingham, Alabama 35201-1295

September 3, 19

~~See Report~~
See amendment to License

SUBJECT: ISSUANCE OF AMENDMENTS - JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. M95700 AND M95701)

Dear Mr. Morey:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 121 to Facility Operating License No. NPF-2 and Amendment No. 113 to Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant, Units 1 and 2. The amendments change the Technical Specifications (TS) in response to your submittal dated June 12, 1996.

The amendments revise the reactor core safety limits, Overtemperature delta T (OTDT) and Overpressure delta T (OPDT) reactor trip setpoints and allowable values, and the power distribution limits associated with implementation of Relaxed Axial Offset Control (RAOC) and F_Q surveillance. The amendments also include changes to the Bases associated with these specifications and surveillances.

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

Sincerely,

Original signed by:

Byron L. Siegel, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-348 and 50-364

Enclosures:
1. Amendment No. 121 to NPF-2
2. Amendment No. 113 to NPF-8
3. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 3, 1996

Mr. D. N. Morey
Vice President - Farley Project
Southern Nuclear Operating
Company, Inc.
Post Office Box 1295
Birmingham, Alabama 35201-1295

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cc w/encls: See next page

Mr. D. N. Morey
Southern Nuclear Operating
Company, Inc.

Joseph M. Farley Nuclear Plant

cc:
Mr. R. D. Hill, Jr.
General Manager -
Southern Nuclear Operating Company
Post Office Box 470
Ashford, Alabama 36312

Mr. Mark Ajluni, Licensing Manager
Southern Nuclear Operating Company
Post Office Box 1295
Birmingham, Alabama 35201-1295

Mr. M. Stanford Blanton
Balch and Bingham Law Firm
Post Office Box 306
1710 Sixth Avenue North
Birmingham, Alabama 35201

Mr. J. D. Woodard
Executive Vice President
Southern Nuclear Operating Company
Post Office Box 1295
Birmingham, Alabama 35201

State Health Officer
Alabama Department of Public Health
434 Monroe Street
Montgomery, Alabama 36130-1701

Chairman
Houston County Commission
Post Office Box 6406
Dothan, Alabama 36302

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

Resident Inspector
U.S. Nuclear Regulatory Commission
7388 N. State Highway 95
Columbia, Alabama 36319

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

ALABAMA POWER COMPANY

DOCKET NO. 50-348

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 121
License No. NPF-2

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

   A. The application for amendment by Southern Nuclear Operating Company, Inc. (Southern Nuclear), dated June 12, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-2 is hereby amended to read as follows:

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P PDR

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 121, are hereby incorporated in the license. Southern Nuclear 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 issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 3, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 121

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. The revised areas are indicated by marginal lines.

Remove Pages	Insert Pages
XIX	XIX
2-2	2-2
2-8	2-8
2-9	2-9
2-10	2-10
B 2-2	B 2-2
B 2-5	B 2-5
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-3	3/4 2-3
3/4 2-4	3/4 2-4
3/4 2-5	3/4 2-5
3/4 2-6	3/4 2-6
3/4 2-8	3/4 2-8
3/4 7-2	3/4 7-2
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 2-3	B 3/4 2-3
B 3/4 2-4	B 3/4 2-4
B 3/4 2-5	B 3/4 2-5
6-19	6-19

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6.9 REPORTING REQUIREMENTS

6.9.1 ROUTINE REPORTS

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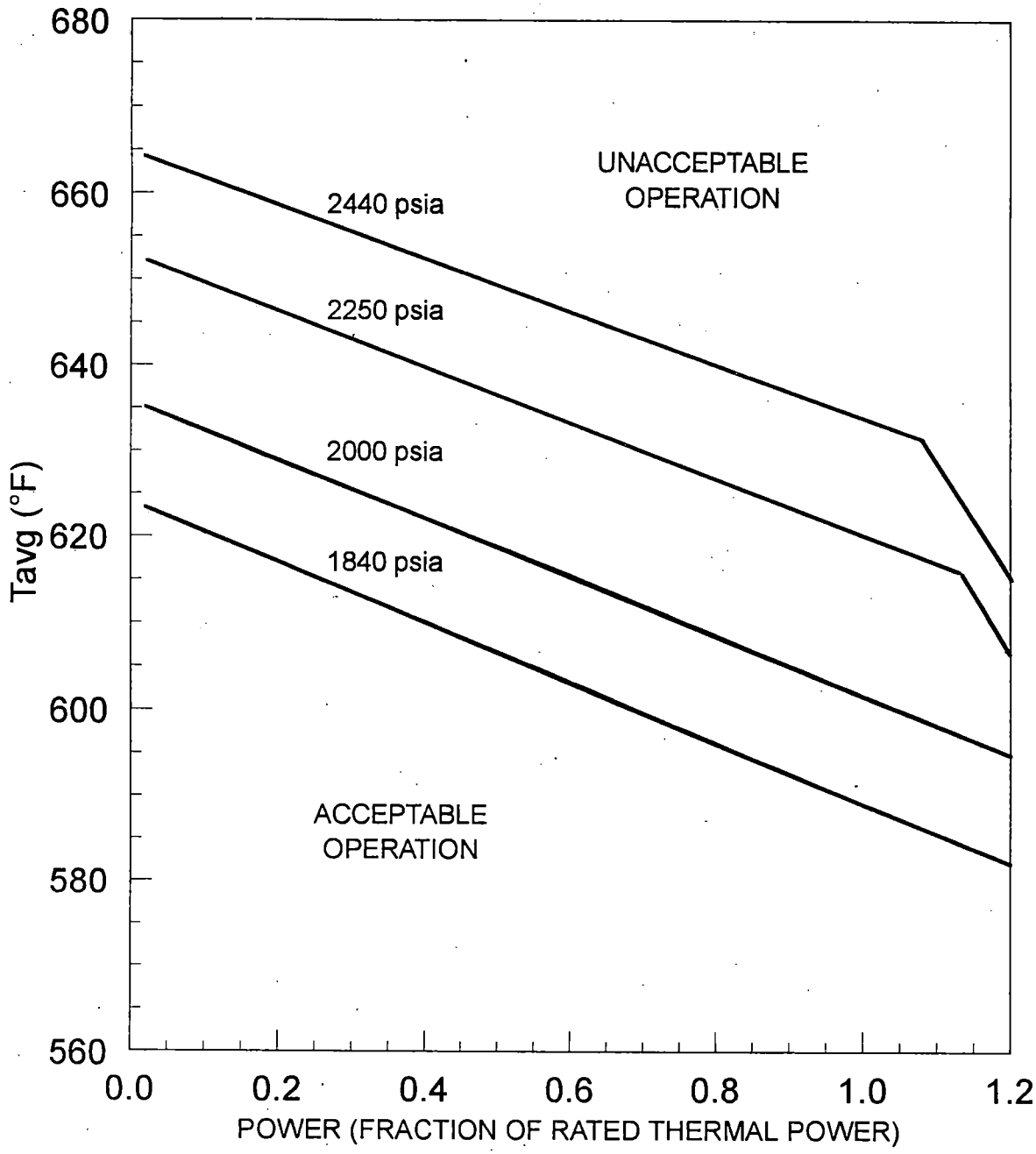

Figure 2.1-1 Reactor Core Safety Limits
Three Loops in Operation

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

Note 1: Overtemperature ΔT

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

where:

ΔT	=	Measured ΔT by RTD instrumentation;
ΔT_0	=	Indicated ΔT at RATED THERMAL POWER and reference T_{avg};
T	=	Average temperature, °F;
T'	=	Reference T_{avg} at RATED THERMAL POWER (≤ 577.2°F);
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 sec, τ_2 = 4 sec;
$\frac{1+\tau_4 s}{1+\tau_5 s}$	=	The function generated by the lead-lag controller for ΔT dynamic compensation;
τ_4 & τ_5	=	Time constants utilized in the lead-lag controller for ΔT, τ_4 = 0 sec, $\tau_5 \le 6$ sec;
$\frac{1}{1+\tau_6 s}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 \le 6$ sec;
s	=	Laplace transform operator, sec^{-1};

Operation with 3 loops	Operation with 2 loops
K_1 = 1.17;	K_1 = (values blank pending
K_2 = 0.017;	K_2 = NRC approval of
K_3 = 0.000825;	K_3 = 2 loop operation)

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

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

(i) for $q_t - q_b$ between -23 percent and +15 percent, f_1 (ΔI) = 0 (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER);

(ii) for each percent that the magnitude of ($q_t - q_b$) exceeds -23 percent, the ΔT trip setpoint shall be automatically reduced by 2.48 percent of its value at RATED THERMAL POWER; and

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds +15 percent, the ΔT trip setpoint shall be automatically reduced by 2.05 percent of its value at RATED THERMAL POWER.

Note 2: Overpower ΔT

$$\Delta T\left(\frac{1+\tau_4 s}{1+\tau_5 s}\right) \le \Delta T_0\left[K_4 - K_5\left(\frac{\tau_3 s}{1+\tau_3 s}\right)T\left(\frac{1}{1+\tau_6 s}\right) - K_6\left\{T\left(\frac{1}{1+\tau_6 s}\right) - T''\right\} - f_2(\Delta I)\right]$$

where: ΔT = Measured ΔT by RTD instrumentation;

ΔT_0 = Indicated ΔT at RATED THERMAL POWER and reference T_{avg};

T = Average temperature, °F;

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

K_4 = 1.10;

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 $T \le T''$;

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

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

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

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

τ_4 & τ_5 = Time constants utilized in the lead-lag controller for ΔT, τ_4 = 0 sec, $\tau_5 \leq 6$ sec;

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

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 \leq 6$ sec;

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

$f_2(\Delta I)$ = 0 for all ΔI.

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 0.4 percent ΔT span.

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

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

Note 6: The channel's maximum trip point shall not exceed its computed trip point by more than 0.4 percent ΔT span.

The curves of Figures 2.1-1 and 2.1-2 are based on the most limiting result using an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.70 for VANTAGE 5 fuel and an $F^N_{\Delta H}$ of 1.30 for LOPAR fuel 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:

$$F^N_{\Delta H} = 1.70\ [1 + 0.3\ (1 - P)] \text{ for VANTAGE 5 fuel and}$$

$$F^N_{\Delta H} = 1.30\ [1 + 0.3\ (1 - P)] \text{ for LOPAR fuel}$$

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 f_1 (delta I) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature delta T trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel, pressurizer and the reactor coolant system piping and fittings are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

Overpower ΔT

The Overpower delta T reactor trip provides assurance of fuel integrity (e.g., no fuel pellet melting) under all possible overpower conditions, limits the required range for Overtemperature delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for axial power distribution, changes in density and heat capacity of water with temperature, and dynamic compensation for transport, thermowell, and RTD response time delays from the core to RTD output indication.

Pressurizer Pressure

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

Pressurizer Water Level

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

Loss of Flow

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

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

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the limits specified in Figure 3.2-1.*

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER**

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the limits specified in Figure 3.2-1:

   1. Either restore the indicated AFD to within the limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes.

SURVEILLANCE REQUIREMENTS

4.2.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

   1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

   2. At least once per hour with the AFD Monitor Alarm inoperable.

* The indicated AFD shall be considered outside of its limits when at least 2 OPERABLE excore channels are indicating the AFD to be outside its limits.

**See Special Test Exception 3.10.2.

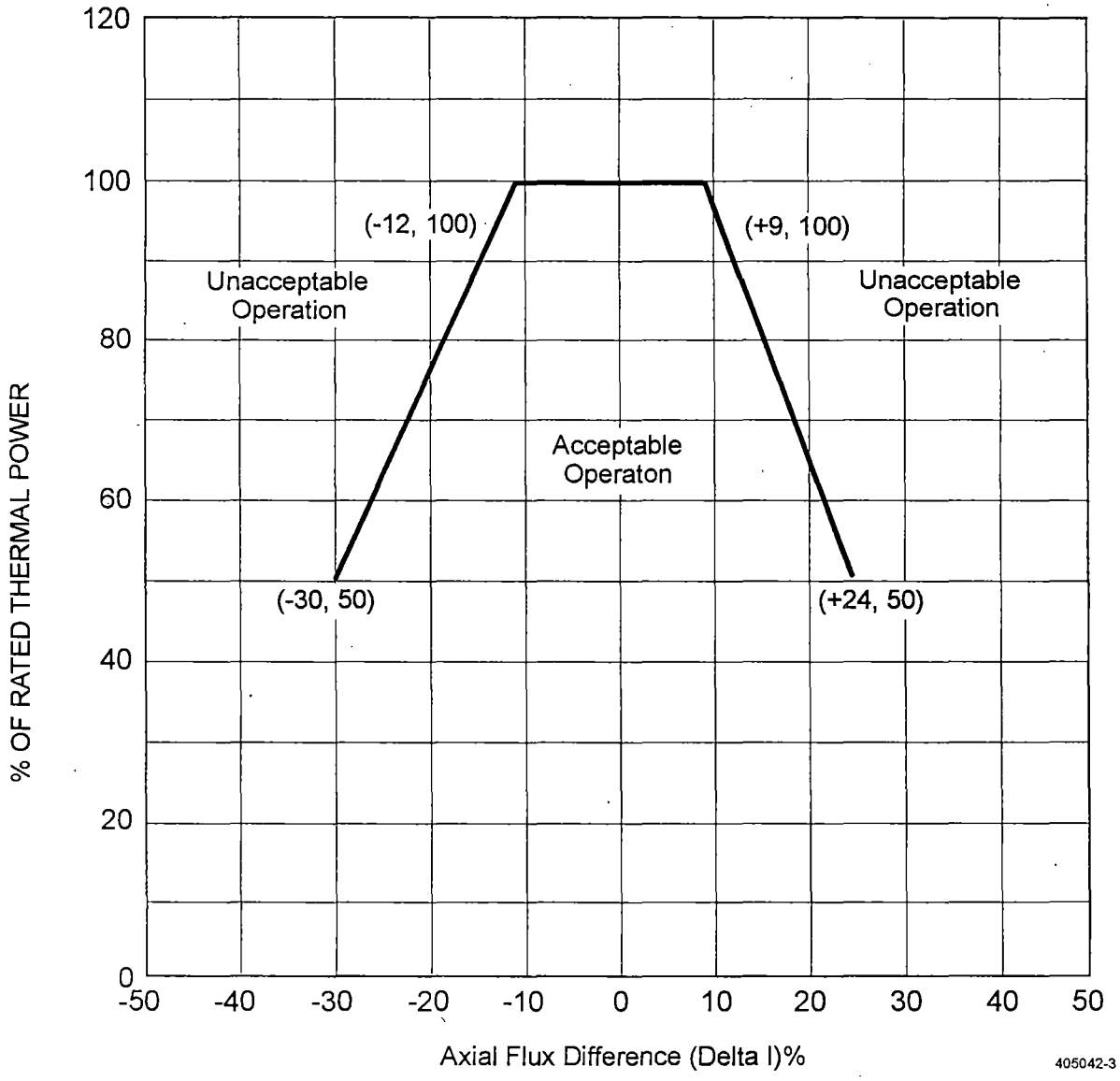

FIGURE 3.2-1 AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER FOR RAOC

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:

$$F_Q(Z) \leq \left[\frac{2.45}{P}\right] [K(Z)] \text{ for } P > 0.5 \text{ for VANTAGE 5 fuel}$$

$$F_Q(Z) \leq [4.9] [K(Z)] \text{ for } P \leq 0.5 \text{ for VANTAGE 5 fuel and}$$

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

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

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.

b. THERMAL POWER may be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q(Z)$ shall be evaluated to determine if it is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS (Continued)

b. Determining the computed heat flux hot channel factor $F_Q^C(Z)$, as follows:

Increase the measured $F_Q(Z)$ obtained from the power distribution map by 3% to account for manufacturing tolerances and further increase the value by 5% to account for measurement uncertainties.

c. Verifying that $F_Q^C(Z)$, obtained in Specification 4.2.2.2b above, satisfies the relationship in Specification 3.2.2.

d. Satisfying the following relationship:

$$F_Q^C(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{P \times W(Z)} \text{ for } P > 0.5$$

$$F_Q^C(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{0.5 \times W(Z)} \text{ for } P \leq 0.5$$

Where $F_Q^C(Z)$ is obtained in Specification 4.2.2.2b above, F_Q^{RTP} is the F_Q limit, K(Z) is the normalized $F_Q(Z)$ as a function of core height, P is the fraction of RATED THERMAL POWER, and W(Z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation.

F_Q^{RTP}	= 2.45 (VANTAGE 5 fuel) = 2.32 (LOPAR fuel)
K(Z)	provided in Figure 3.2-2
W(Z)	provided in the Radial Peaking Factor Limit Report

e. Measuring $F_Q(Z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 20% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(Z)$ was last determined*, or

2. At least once per 31 Effective Full Power Days, whichever occurs first.

*During power escalation after each fuel loading, power level may be increased until equilibrium conditions at any power level greater than or equal to 50% of RATED THERMAL POWER have been achieved and a power distribution map obtained.

f. With measurements indicating

$$\underset{\text{over }(Z)}{\text{maximum}} \left(\frac{F_Q{}^C(Z)}{K(Z)} \right)$$

has increased since the previous determination of $F_Q^C(Z)$ either of the following actions shall be taken:

1) Increase $F_Q^C(Z)$ by the $F_Q^C(Z)$ penalty factor specified in the Peaking Factor Limit Report and verify that this value satisfies the relationship in Specification 4.2.2.2d, or

2) $F_Q^C(Z)$ shall be measured at least once per 7 Effective Full Power Days until two successive maps indicate that

$$\underset{\text{over }(Z)}{\text{maximum}} \left(\frac{F_Q{}^C(Z)}{K(Z)} \right) \text{ is not increasing.}$$

g. With the relationships specified in Specification 4.2.2.2d above not being satisfied:

1) Calculate the percent $F_Q(Z)$ exceeds its limits by the following expression:

$$\left\{ \left(\underset{\text{over } Z}{\text{maximum}} \left[\frac{F_Q^C(Z) \times W(Z)}{\frac{F_Q^{RTP}}{P} \times K(Z)} \right] \right) - 1 \right\} \times 100 \text{ for } P > 0.5$$

$$\left\{ \left(\underset{\text{over } Z}{\text{maximum}} \left[\frac{F_Q^C(Z) \times W(Z)}{\frac{F_Q^{RTP}}{0.5} \times K(Z)} \right] \right) - 1 \right\} \times 100 \text{ for } P \leq 0.5, \text{ and}$$

2) The following action shall be taken:

Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the AFD limits specified in LCO 3.2.1, Axial Flux Difference, by 1% AFD for each percent $F_Q(Z)$ exceeds its limits as determined in Specification 4.2.2.2g.1. Within 8 hours, reset the AFD alarm setpoints to these modified limits.

SURVEILLANCE REQUIREMENTS (Continued)

h. The limits specified in Specification 4.2.2.2c are applicable in all core plane regions, i.e., 0 - 100%, inclusive.

i. The limits specified in Specifications 4.2.2.2d, 4.2.2.2f, and 4.2.2.2g above are not applicable in the following core plane regions:

1) Lower core region from 0 to 15%, inclusive.
2) Upper core region from 85 to 100%, inclusive.

4.2.2.3 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

3/4.2.3 NUCLEAR ENTHALPY 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:

$F^N_{\Delta H} \leq 1.70\ [1 + 0.3\ (1 - P)]$ for VANTAGE 5 fuel and

$F^N_{\Delta H} \leq 1.30\ [1 + 0.3\ (1 - P)]$ for LOPAR fuel

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

APPLICABILITY: MODE 1

ACTION:

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

a. Within 4 hours either:

1. Restore $F^N_{\Delta H}$ to within the above limit; and demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours of exceeding the limit, or

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

b. Demonstrate through in-core mapping, if not previously performed per a.1 above, 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.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 3 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	60***
2	43
3	24

TABLE 3.7-2

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 2 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator*	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	**
2	**
3	**

*At least two safety valves shall be OPERABLE on the non-operating steam generator.

**These values left blank pending NRC approval of 2 loop operation.

***For plant operation approaching end of cycle (i.e., core average burnup ≥ 14,000 MWD/MTU), with one inoperable safety valve on any steam generator, the maximum allowable Power Range Neutron Flux setpoint may be increased from 60% to 87% RTP.

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) meeting the DNB design criterion 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.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of 2.45 for VANTAGE 5 and 2.32 for LOPAR times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for 2 or more OPERABLE excore channels is outside the allowed ΔI operating space for RAOC operation specified in Figure 3.2-1 and the THERMAL POWER is greater than 50% RATED THERMAL POWER.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, NUCLEAR ENTHALPY HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, and nuclear enthalpy rise hot channel factor ensure that 1) the design limit on peak local power density is not exceeded, 2) the DNB design criterion is met, and 3) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position.

b. Control rod banks are sequenced with overlapping groups as described in Specification 3.1.3.6.

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided conditions a. through d. above are maintained. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.

The heat flux hot channel factor $F_Q(Z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to RAOC operation, W(Z), to provide assurance that the limit on the heat flux hot channel factor $F_Q(Z)$ is met. W(Z) accounts for the effects of normal operational transients within the AFD limits and was determined from expected power control maneuvers over the full range of burnup conditions in the core.

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When $F^N_{\Delta H}$ is measured, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system. The specified limit for $F^N_{\Delta H}$ contains an 8% allowance for uncertainties. The 8% allowance is based on the following considerations:

a. Abnormal perturbations in the radial power shape, such as from rod misalignment, affect $F^N_{\Delta H}$ more directly than F_Q,

b. Although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. Errors in prediction for control power shape detected during startup physics tests can be compensated for in F_Q by restricting axial flux distribution. This compensation for $F^N_{\Delta H}$ is less readily available.

If $F^N_{\Delta H}$ exceeds its limit, the unit will be allowed 4 hours to restore $F^N_{\Delta H}$ to within its limits. This restoration may, for example, involve realigning any misaligned rods or reducing power enough to bring $F^N_{\Delta H}$ within its power dependent limit. When the $F^N_{\Delta H}$ limit is exceeded, the DNBR limit is not likely violated in steady state operation, because events that could significantly perturb the $F^N_{\Delta H}$ value, e.g., static control rod misalignment, are considered in the safety analyses. However, the DNBR limit may be violated if a DNB limiting event occurs while $F^N_{\Delta H}$ is above its limit. The increased allowed action time of 4 hours provides an acceptable time to restore $F^N_{\Delta H}$ to within its limits without allowing the plant to remain in an unacceptable condition for an extended period of time.

Once corrective action has been taken, e.g., realignment of misaligned rods or reduction of power, an incore flux map must be obtained and the measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit. Twenty additional hours are provided to perform this task above the four hours allowed by Action Statement 3/4.2.3.a. The completion time of 24 hours is acceptable because of the low probability of having a DNB limiting event within this 24 hour period and, in the event that power is reduced, an increase in DNB margin is obtained at lower power levels. Additionally, operating experience has indicated that this completion time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate $F^N_{\Delta H}$.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3 percent for each percent of tilt in excess of 1.0.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to meet the DNB design criterion throughout each analyzed transient. The indicated T_{avg} value of 580.7°F is based on the average of two control board readings and an indication uncertainty of 2.5°F. The indicated pressure value of 2205 psig is based on the average of two control board readings and an indication uncertainty of 20 psi. The indicated total RCS flow rate is based on one elbow tap measurement from each loop and an uncertainty of 2.4% flow (0.1% flow is included for feedwater venturi fouling).

The 12 hour surveillance of T_{avg} and pressurizer pressure through the control board readings are sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

The 18 month surveillance of the total RCS flow rate is a precision measurement that verifies the RCS flow requirement at the beginning of each fuel cycle and ensures correlation of the flow indication channels with the measured loop flows. The monthly surveillance of the total RCS flow rate is a reverification of the RCS flow requirement using loop elbow tap measurements that are correlated to the precision RCS flow measurement at the beginning of the fuel cycle. The 12 hour RCS flow surveillance is a qualitative verification of significant flow degradation using the control board indicators and the loop elbow tap measurements that are correlated to the precision RCS flow measurement at the beginning of each fuel cycle.

MONTHLY OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORV's or safety valves, shall be submitted on a monthly basis to the Commission, pursuant to 10 CFR 50.4, no later than the 15th of each month following the calendar month covered by the report.

PEAKING FACTOR LIMIT REPORT

6.9.1.11 The cycle dependent function W(Z) and the burnup dependent $F_Q^C(Z)$ penalty factors, required for calculation of $F_Q^C(Z)$ specified in LCO 3.2.2, "Heat Flux Hot Channel Factor - $F_Q(Z)$," shall be documented in the Peaking Factor Limit Report in accordance with the methodology in WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control FQ Surveillance Technical Specification," Rev. 1, February 1994 (W Proprietary).

The Peaking Factor Limit Report shall be provided to the Commission, pursuant to 10 CFR 50.4, upon issuance prior to each reload cycle (prior to MODE 2). In the event that the limit would be submitted at some other time during core life, it will be submitted upon issuance, unless otherwise exempted by the Commission.

ANNUAL DIESEL GENERATOR RELIABILITY DATA REPORT

6.9.1.12 The number of tests (valid or invalid) and the number of failures to start on demand for each diesel generator shall be submitted to the NRC annually. This report shall contain the information identified in Regulatory Position C.3.b of NRC Regulatory Guide 1.108, Revision 1, 1977.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

ALABAMA POWER COMPANY

DOCKET NO. 50-364

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 113
License No. NPF-8

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

   A. The application for amendment by Southern Nuclear Operating Company, Inc. (Southern Nuclear), dated June 12, 1996, 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.113 , are hereby incorporated in the license. Southern Nuclear 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 issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 3, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 113

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. The revised areas are indicated by marginal lines.

Remove Pages	Insert Pages
XIX	XIX
2-2	2-2
2-8	2-8
2-9	2-9
2-10	2-10
B 2-2	B 2-2
B 2-5	B 2-5
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-3	3/4 2-3
3/4 2-4	3/4 2-4
3/4 2-5	3/4 2-5
3/4 2-6	3/4 2-6
3/4 2-8	3/4 2-8
3/4 7-2	3/4 7-2
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 2-3	B 3/4 2-3
B 3/4 2-4	B 3/4 2-4
B 3/4 2-5	B 3/4 2-5
6-19	6-19

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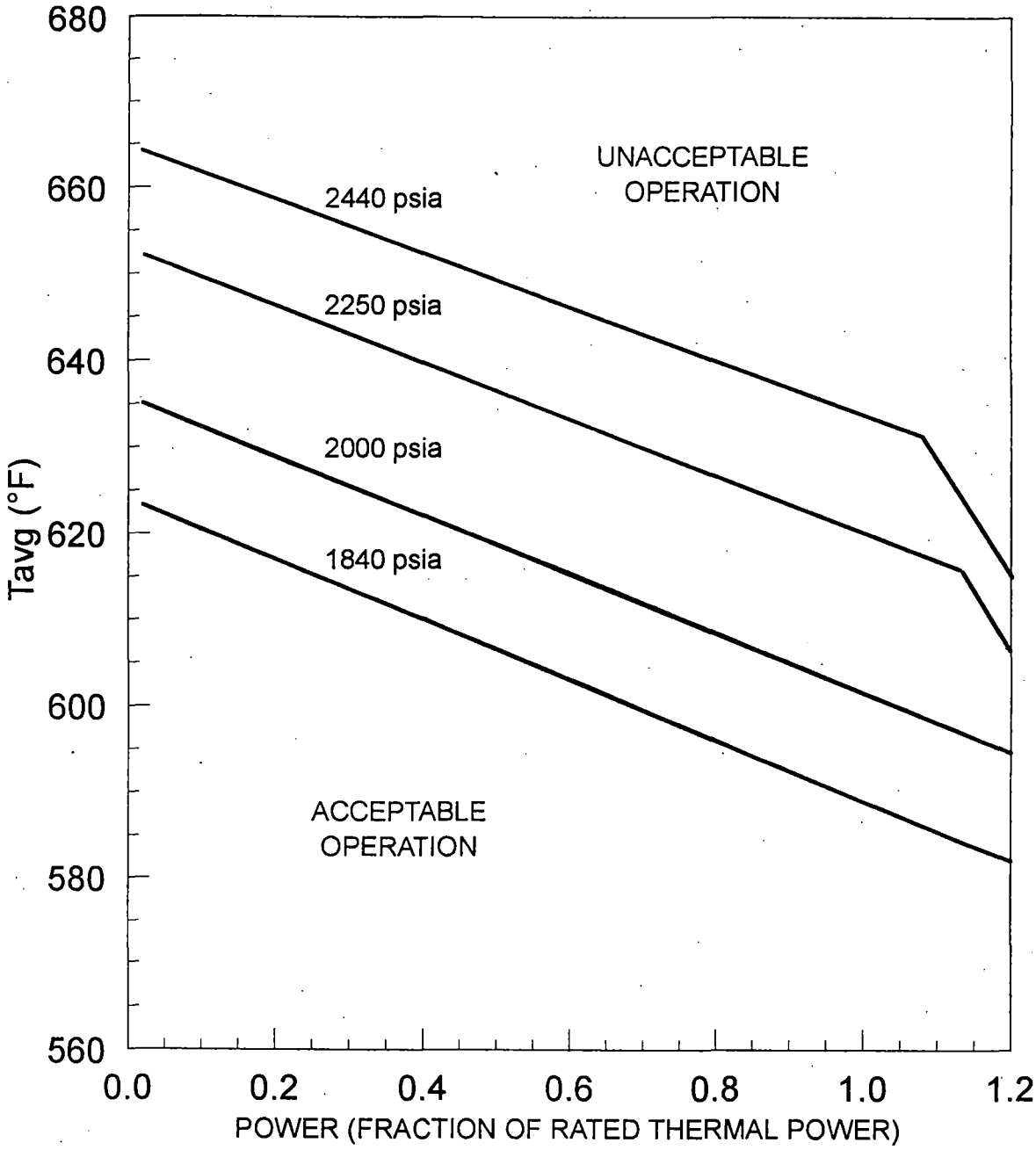

Figure 2.1-1 Reactor Core Safety Limits
Three Loops in Operation

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

Note 1: Overtemperature ΔT

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

where:

ΔT	=	Measured ΔT by RTD instrumentation;
ΔT_0	=	Indicated ΔT at RATED THERMAL POWER and reference T_{avg};
T	=	Average temperature, °F;
T'	=	Reference T_{avg} at RATED THERMAL POWER (≤ 577.2°F);
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 sec, τ_2 = 4 sec;
$\frac{1+\tau_4 s}{1+\tau_5 s}$	=	The function generated by the lead-lag controller for ΔT dynamic compensation;
τ_4 & τ_5	=	Time constants utilized in the lead-lag controller for ΔT, τ_4 = 0 sec, $\tau_5 \le 6$ sec;
$\frac{1}{1+\tau_6 s}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 \le 6$ sec;
s	=	Laplace transform operator, sec^{-1};

Operation with 3 loops	Operation with 2 loops
K_1 = 1.17;	K_1 = (values blank pending
K_2 = 0.017;	K_2 = NRC approval of
K_3 = 0.000825;	K_3 = 2 loop operation)

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

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

(i) for $q_t - q_b$ between -23 percent and +15 percent, f_1 (ΔI) = 0 (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER);

(ii) for each percent that the magnitude of ($q_t - q_b$) exceeds -23 percent, the ΔT trip setpoint shall be automatically reduced by 2.48 percent of its value at RATED THERMAL POWER; and

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds +15 percent, the ΔT trip setpoint shall be automatically reduced by 2.05 percent of its value at RATED THERMAL POWER.

Note 2: Overpower ΔT

$$\Delta T\left(\frac{1+\tau_4 s}{1+\tau_5 s}\right) \le \Delta T_0\left[K_4 - K_5\left(\frac{\tau_3 s}{1+\tau_3 s}\right)T\left(\frac{1}{1+\tau_6 s}\right) - K_6\left\{T\left(\frac{1}{1+\tau_6 s}\right) - T''\right\} - f_2(\Delta I)\right]$$

where: ΔT = Measured ΔT by RTD instrumentation;

ΔT_0 = Indicated ΔT at RATED THERMAL POWER and reference T_{avg};

T = Average temperature, °F;

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

K_4 = 1.10;

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 $T \le T''$;

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

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

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

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

τ_4 & τ_5 = Time constants utilized in the lead-lag controller for ΔT, τ_4 = 0 sec, $\tau_5 \leq 6$ sec;

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

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 \leq 6$ sec;

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

$f_2(\Delta I)$ = 0 for all ΔI.

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 0.4 percent ΔT span.

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

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

Note 6: The channel's maximum trip point shall not exceed its computed trip point by more than 0.4 percent ΔT span.

The curves of Figures 2.1-1 and 2.1-2 are based on the most limiting result using an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.65 for VANTAGE 5 fuel and an $F^N_{\Delta H}$ of 1.30 for LOPAR fuel 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:

$$F^N_{\Delta H} = 1.65\ [1 + 0.3\ (1 - P)] \text{ for VANTAGE 5 fuel and}$$

$$F^N_{\Delta H} = 1.30\ [1 + 0.3\ (1 - P)] \text{ for LOPAR fuel}$$

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 f_1 (delta I) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature delta T trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel, pressurizer and the reactor coolant system piping and fittings are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

Overpower ΔT

The Overpower delta T reactor trip provides assurance of fuel integrity (e.g., no fuel pellet melting) under all possible overpower conditions, limits the required range for Overtemperature delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for axial power distribution, changes in density and heat capacity of water with temperature, and dynamic compensation for transport, thermowell, and RTD response time delays from the core to RTD output indication.

Pressurizer Pressure

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

Pressurizer Water Level

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

Loss of Flow

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

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

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the limits specified in Figure 3.2-1.*

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER**

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the limits specified in Figure 3.2-1:

   1. Either restore the indicated AFD to within the limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes.

SURVEILLANCE REQUIREMENTS

4.2.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

   1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

   2. At least once per hour with the AFD Monitor Alarm inoperable.

* The indicated AFD shall be considered outside of its limits when at least 2 OPERABLE excore channels are indicating the AFD to be outside its limits.

**See Special Test Exception 3.10.2.

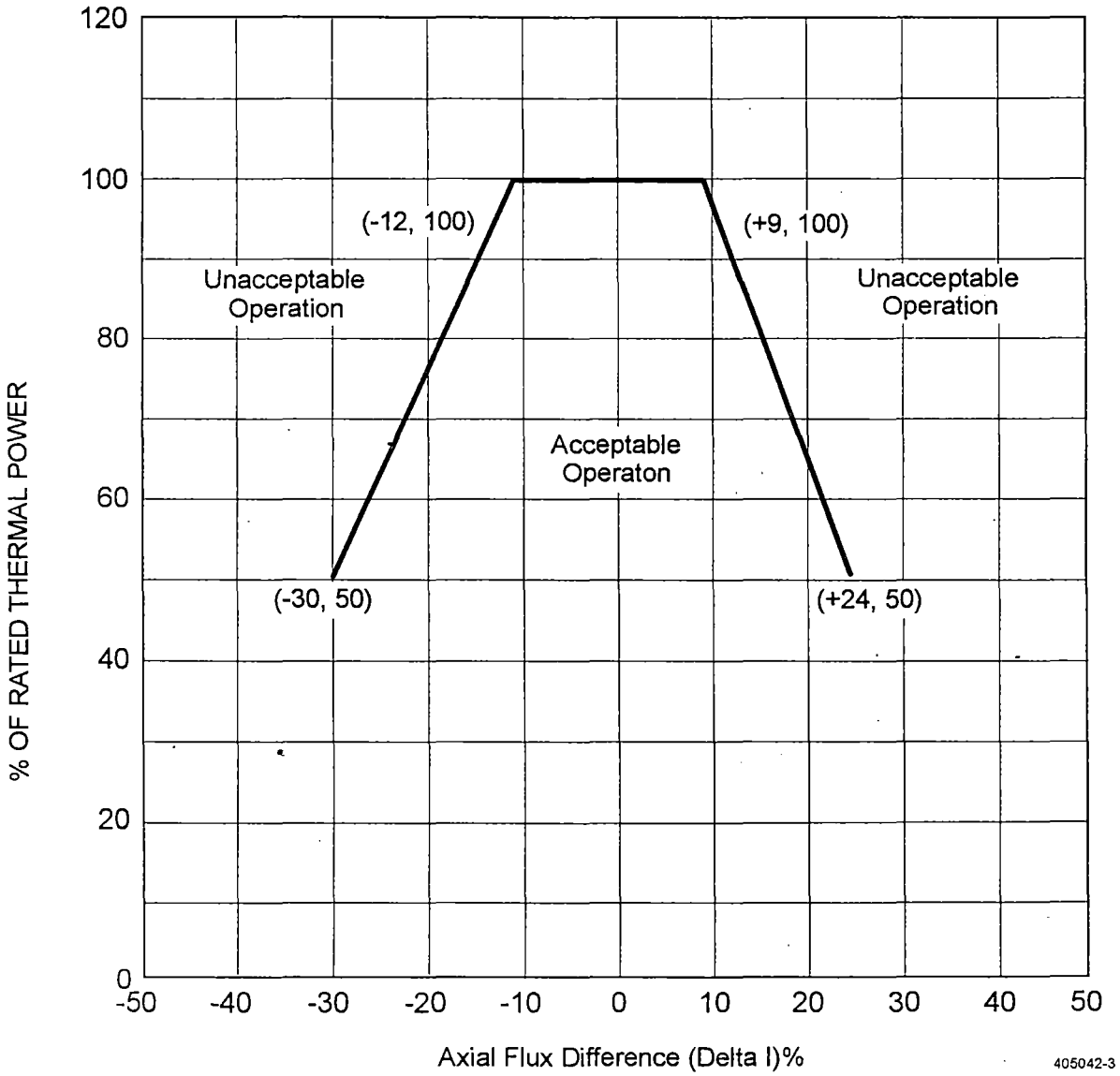

FIGURE 3.2-1 AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER FOR RAOC

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:

$$F_Q(Z) \leq \left[\frac{2.45}{P}\right] [K(Z)] \text{ for } P > 0.5 \text{ for VANTAGE 5 fuel}$$

$$F_Q(Z) \leq [4.9] [K(Z)] \text{ for } P \leq 0.5 \text{ for VANTAGE 5 fuel and}$$

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

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

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

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.

b. THERMAL POWER may be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q(Z)$ shall be evaluated to determine if it is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS (Continued)

b. Determining the computed heat flux hot channel factor $F_Q^C(Z)$, as follows:

Increase the measured $F_Q(Z)$ obtained from the power distribution map by 3% to account for manufacturing tolerances and further increase the value by 5% to account for measurement uncertainties.

c. Verifying that $F_Q^C(Z)$, obtained in Specification 4.2.2.2b above, satisfies the relationship in Specification 3.2.2.

d. Satisfying the following relationship:

$$F_Q^C(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{P \times W(Z)} \text{ for } P > 0.5$$

$$F_Q^C(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{0.5 \times W(Z)} \text{ for } P \leq 0.5$$

Where $F_Q^C(Z)$ is obtained in Specification 4.2.2.2b above, F_Q^{RTP} is the F_Q limit, K(Z) is the normalized $F_Q(Z)$ as a function of core height, P is the fraction of RATED THERMAL POWER, and W(Z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation.

F_Q^{RTP}	= 2.45 (VANTAGE 5 fuel) = 2.32 (LOPAR fuel)
K(Z)	provided in Figure 3.2-2
W(Z)	provided in the Radial Peaking Factor Limit Report

e. Measuring $F_Q(Z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 20% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(Z)$ was last determined*, or

2. At least once per 31 Effective Full Power Days, whichever occurs first.

*During power escalation after each fuel loading, power level may be increased until equilibrium conditions at any power level greater than or equal to 50% of RATED THERMAL POWER have been achieved and a power distribution map obtained.

f. With measurements indicating

$$\underset{\text{over }(Z)}{\text{maximum}} \left(\frac{F_Q{}^C(Z)}{K(Z)} \right)$$

has increased since the previous determination of $F_Q^C(Z)$ either of the following actions shall be taken:

1) Increase $F_Q^C(Z)$ by the $F_Q^C(Z)$ penalty factor specified in the Peaking Factor Limit Report and verify that this value satisfies the relationship in Specification 4.2.2.2d, or

2) $F_Q^C(Z)$ shall be measured at least once per 7 Effective Full Power Days until two successive maps indicate that

$$\underset{\text{over }(Z)}{\text{maximum}} \left(\frac{F_Q{}^C(Z)}{K(Z)} \right) \text{ is not increasing.}$$

g. With the relationships specified in Specification 4.2.2.2d above not being satisfied:

1) Calculate the percent $F_Q(Z)$ exceeds its limits by the following expression:

$$\left\{ \left(\underset{\text{over } Z}{\text{maximum}} \left[\frac{F_Q^C(Z) \times W(Z)}{\frac{F_Q^{RTP}}{P} \times K(Z)} \right] \right) - 1 \right\} \times 100 \text{ for } P > 0.5$$

$$\left\{ \left(\underset{\text{over } Z}{\text{maximum}} \left[\frac{F_Q^C(Z) \times W(Z)}{\frac{F_Q^{RTP}}{0.5} \times K(Z)} \right] \right) - 1 \right\} \times 100 \text{ for } P \leq 0.5, \text{ and}$$

2) The following action shall be taken:

Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the AFD limits specified in LCO 3.2.1, Axial Flux Difference, by 1% AFD for each percent $F_Q(Z)$ exceeds its limits as determined in Specification 4.2.2.2g.1. Within 8 hours, reset the AFD alarm setpoints to these modified limits.

SURVEILLANCE REQUIREMENTS (Continued)

h. The limits specified in Specification 4.2.2.2c are applicable in all core plane regions, i.e., 0 - 100%, inclusive.

i. The limits specified in Specifications 4.2.2.2d, 4.2.2.2f, and 4.2.2.2g above are not applicable in the following core plane regions:

1) Lower core region from 0 to 15%, inclusive.
2) Upper core region from 85 to 100%, inclusive.

4.2.2.3 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

3/4.2.3 NUCLEAR ENTHALPY 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:

$F^N_{\Delta H} \leq 1.65\ [1 + 0.3\ (1 - P)]$ for VANTAGE 5 fuel and

$F^N_{\Delta H} \leq 1.30\ [1 + 0.3\ (1 - P)]$ for LOPAR fuel

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

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.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 3 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	60***
2	43
3	24

TABLE 3.7-2

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 2 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator*	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	**
2	**
3	**

*At least two safety valves shall be OPERABLE on the non-operating steam generator.

**These values left blank pending NRC approval of 2 loop operation.

***For plant operation approaching end of cycle (i.e., core average burnup ≥ 14,000 MWD/MTU), with one inoperable safety valve on any steam generator, the maximum allowable Power Range Neutron Flux setpoint may be increased from 60% to 87% RTP.

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) meeting the DNB design criterion 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.

$F_{\Delta H}^N$ 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.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of 2.45 for VANTAGE 5 and 2.32 for LOPAR times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for 2 or more OPERABLE excore channels is outside the allowed ΔI operating space for RAOC operation specified in Figure 3.2-1 and the THERMAL POWER is greater than 50% RATED THERMAL POWER.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, NUCLEAR ENTHALPY HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, and nuclear enthalpy rise hot channel factor ensure that 1) the design limit on peak local power density is not exceeded, 2) the DNB design criterion is met, and 3) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position.

b. Control rod banks are sequenced with overlapping groups as described in Specification 3.1.3.6.

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided conditions a. through d. above are maintained. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.

The heat flux hot channel factor $F_Q(Z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to RAOC operation, W(Z), to provide assurance that the limit on the heat flux hot channel factor $F_Q(Z)$ is met. W(Z) accounts for the effects of normal operational transients within the AFD limits and was determined from expected power control maneuvers over the full range of burnup conditions in the core.

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When $F^N_{\Delta H}$ is measured, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system. The specified limit for $F^N_{\Delta H}$ contains an 8% allowance for uncertainties. The 8% allowance is based on the following considerations:

a. Abnormal perturbations in the radial power shape, such as from rod misalignment, affect $F^N_{\Delta H}$ more directly than F_Q,

b. Although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. Errors in prediction for control power shape detected during startup physics tests can be compensated for in F_Q by restricting axial flux distribution. This compensation for $F^N_{\Delta H}$ is less readily available.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3 percent for each percent of tilt in excess of 1.0.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to meet the DNB design criterion throughout each analyzed transient. The indicated T_{avg} value of 580.7°F is based on the average of two control board readings and an indication uncertainty of 2.5°F. The indicated pressure value of 2205 psig is based on the average of two control board readings and an indication uncertainty of 20 psi. The indicated total RCS flow rate is based on one elbow tap measurement from each loop and an uncertainty of 2.4% flow (0.1% flow is included for feedwater venturi fouling).

The 12 hour surveillance of T_{avg} and pressurizer pressure through the control board readings are sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

The 18 month surveillance of the total RCS flow rate is a precision measurement that verifies the RCS flow requirement at the beginning of each fuel cycle and ensures correlation of the flow indication channels with the measured loop flows. The monthly surveillance of the total RCS flow rate is a reverification of the RCS flow requirement using loop elbow tap measurements that are correlated to the precision RCS flow measurement at the beginning of the fuel cycle. The 12 hour RCS flow surveillance is a qualitative verification of significant flow degradation using the control board indicators and the loop elbow tap measurements that are correlated to the precision RCS flow measurement at the beginning of each fuel cycle.

MONTHLY OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORV's or safety valves, shall be submitted on a monthly basis to the Commission, pursuant to 10 CFR 50.4, no later than the 15th of each month following the calendar month covered by the report.

PEAKING FACTOR LIMIT REPORT

6.9.1.11 The cycle dependent function, W(Z), and the burnup dependent $F_Q^C(Z)$ penalty factors, required for calculation of $F_Q^C(Z)$ specified in LCO 3.2.2, "Heat Flux Hot Channel Factor - $F_Q(Z)$," shall be documented in the Peaking Factor Limit Report in accordance with the methodology in WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control FQ Surveillance Technical Specification," Rev. 1, February 1994 (W Proprietary).

The Peaking Factor Limit Report shall be provided to the Commission, pursuant to 10 CFR 50.4, upon issuance prior to each reload cycle (prior to MODE 2). In the event that the limit would be submitted at some other time during core life, it will be submitted upon issuance, unless otherwise exempted by the Commission.

ANNUAL DIESEL GENERATOR RELIABILITY DATA REPORT

6.9.1.12 The number of tests (valid or invalid) and the number of failures to start on demand for each diesel generator shall be submitted to the NRC annually. This report shall contain the information identified in Regulatory Position C.3.b of NRC Regulatory Guide 1.108, Revision 1, 1977.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 121 TO FACILITY OPERATING LICENSE NO. NPF-2

AND AMENDMENT NO. 113 TO FACILITY OPERATING LICENSE NO. NPF-8

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-348 AND 50-364

1.0 INTRODUCTION

By letter dated June 12, 1996, Southern Nuclear Operating Company, Inc., et al. (SNC or the licensee) proposed to amend the Farley Nuclear Plant (FNP) Units 1 and 2 Technical Specifications (TS). The requested changes revise the reactor core safety limits, Overtemperature delta T (OTDT) and Overpressure delta T (OPDT) reactor trip setpoints and allowable values, and the power distribution limits associated with implementation of Relaxed Axial Offset Control (RAOC) and F_Q surveillance. The changes also include revising the TS Bases associated with these specifications and surveillances, the nuclear enthalpy hot channel factor ($F^N_{\Delta H}$) limit for low parasitic (LOPAR) fuel, the neutron high flux setpoints for inoperable main steamline safety valves (MSSVs), and the administrative requirements for the Peaking Factor Limit Report.

In addition to the TS changes, the SNC letter also transmitted Westinghouse report NSD-NT-OPL-96-152, Revision 2 (Proprietary), "Joseph M. Farley Nuclear Plant Units 1 & 2 Licensing Report for Technical Specification Changes Associated With Revised Core Limits, Revised OTΔT/OPΔT Trip Setpoints and Inclusion of RAOC Control Strategy."

2.0 EVALUATION

The following specific TS changes have been proposed for the FNP Units 1 and 2:

(1) TS 2.1 Reactor Core Safety Limits (Figure 2.1-1 and Bases)

Since FNP is completing the transition from a full core of LOPAR fuel to a full core of VANTAGE 5 fuel, the reactor core safety limits given in Figure 2.1-1 are being revised to reflect VANTAGE 5 fuel as the most limiting fuel design. The revised limits are based on the approved Revised Thermal Design Procedure (RTDP) methodology and the WRB-2 Departure from Nucleate Boiling (DNB) correlation. Therefore, these proposed changes are acceptable.

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(2) TS 3/4.2.3 Nuclear Enthalpy Hot Channel Factor and Bases

In order to ensure that VANTAGE 5 fuel will remain limiting with respect to DNB and that any burned LOPAR assembly that may be reinserted into the core will not be limiting and thus invalidate the proposed new core limits, the $F^N_{\Delta H}$ limit for LOPAR fuel has been reduced from 1.55 to 1.30 in TS 3/4.2.3. Since the current loss of coolant accident (LOCA) analysis bounds this value, LOCAs are not affected by this reduced $F^N_{\Delta H}$. Any reinserted LOPAR assembly will have an $F^N_{\Delta H}$ lower than this limit, and SNC will verify the proposed core limits to be acceptable prior to reinsertion of any LOPAR assemblies. Therefore, these proposed changes are acceptable.

(3) TS 2.2 Reactor Trip System Instrument Trip Setpoints (Table 2.2-1 and Bases)

Changes to the OTDT and OPDT reactor trip setpoint equations are being made as a result of the changes to the core safety limits to reflect Farley specific uncertainties. These include changes in the K1, K2, K3, K4, and K6 constants defined in Notes 1 and 2 of Table 2.2-1 from 1.14, 0.0250, 0.001275, 1.07, and 0.00165 to 1.17, 0.017, 0.000825, 1.10, and 0.00109, respectively. The corresponding allowable values in Notes 3 and 6 will also be changed. A reanalysis or a reevaluation of all DNB-related events which use OTDT and OPDT as the primary reactor trip by FNP has shown that all acceptance criteria continue to be met. Therefore, these proposed changes are acceptable.

(4) TS 3/4.2.1 Axial Flux Difference (AFD) and Bases
TS 3/4.2.2 Heat Flux Hot Channel Factor - F_Q and Bases

The change in axial offset control strategy from the current Constant Axial Offset Control (CAOC) to RAOC requires changes to the axial flux difference (AFD) specification as well as the replacement of the F_{xy} surveillance methodology with the F_Q surveillance methodology and its associated W(z) function. RAOC is a calculational procedure that defines the allowed operational space of the AFD versus thermal power. The RAOC strategy has been reviewed and approved by the NRC in WCAP-10216-P-A. The proposed TS changes are consistent with those approved for use with RAOC and F_Q surveillance. The effects of axial power shapes resulting from the implementation of RAOC for normal operation, operational transients, and during transients arising from faults of moderate frequency have been evaluated by FNP and all acceptance criteria continue to be met. Therefore, these proposed changes are acceptable.

(5) TS 6.9.1.11 Radial Peaking Factor Limit Report

The change to RAOC requires changes to the penalty factors specified in the Radial Peaking Factor Limit Report (RPFLR). Since the cycle dependent function, W(z), and the burnup dependent $F_Q^c(z)$ penalty factors required for calculation of $F_Q^c(z)$ specified in LCO 3.2.2 will now be documented in the RPFLR in accordance with the RAOC methodology, these proposed changes are acceptable.

(6) TS 3.7 Maximum Allowable Power Range Flux High Setpoint with Inoperable Main Steamline Safety Valve During 3 Loop Operation (Table 3.7-1)

TS Table 3.7-1 defines the maximum power levels at which the FNP units can operate with 1, 2, or 3 inoperable main steam safety valves (MSSVs) on any loop. A burnup dependent setpoint for operation with one MSSV out of service was developed to provide maximum plant flexibility and operability. For core average burnups up to 14,000 megawatt days per metric ton uranium (MWD/MTU), a setpoint of 60% of rated thermal power (RTP) has been established. For higher burnups, the current setpoint of 87% RTP remains acceptable. The reduced high neutron flux reactor trip setpoints are based on a plant-specific analysis which model a direct trip on the revised OTDT equation. This analysis, performed by Westinghouse for FNP, confirmed that these setpoints ensure adequate protection throughout the entire fuel cycle. Therefore, this revision to the allowable high setpoints is acceptable.

3.0 SUMMARY

The staff has reviewed the proposed TS changes to the FNP safety limits, reactor trip setpoints, high neutron flux setpoints for MSSVs out of service, $F^N_{\Delta H}$ for LOPAR, and the use of RAOC strategy. The core safety limits and trip setpoints were determined using the NRC approved DNB methodologies, namely RTDP, and approved DNB correlations. All safety analyses transients and accidents that are protected by the OTDT and OPDT trips were reanalyzed or evaluated and continue to meet all acceptance criteria with the revised setpoints. LOCA analyses are not affected by any of the proposed changes. Based on this, the staff has determined that the proposed changes to the FNP TS are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of Alabama official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change 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 change the surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (61 FR 40029 dated July 31, 1996). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission 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 the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: L. Kopp

Date: September 3, 1996