July 22, 1994

Docket No. 50-348

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

Dear Mr. Morey:

SUBJECT: ISSUANCE OF AMENDMENT NO. 109 TO FACILITY OPERATING LICENSE NO. NPF-2 REGARDING NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1 (TAC NO. M89665)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 109 to Facility Operating License NPF-2 for the Joseph M. Farley Nuclear Plant, Unit 1. The amendment changes the Technical Specifications in response to your submittal dated June 17, 1994.

The amendment changes the Technical Specifications to revise the nuclear enthalpy rise hot channel factor (F delta H) from equal to or less than 1.65 [1 plus 0.3(1-P)] to equal to or less than 1.70 [1 plus 0.3(1-P)] where P is a fraction of rated power. The amendment also revises the action statement to reflect guidance contained in the improved standard technical specifications.

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

Sincerely,

ORIGINAL SIGNED BY:

Byron L. Siegel, Senior Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 109 to NPF-2
2. Safety Evaluation

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AMENDMENT NO. 109 TO FACILITY OPERATING LICENSE NO. NPF-2 - FARLEY, UNIT 1

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

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Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 109 to NPF-2
2. Safety Evaluation

cc w/enclosures:
See next page

cc:

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

DOCKET NO. 50-348

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 109
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 17, 1994, 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:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 109 , are hereby incorporated into 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

David B. Matthews, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 22, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 109

FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

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

Remove Pages	Insert Pages
B 2-2	B 2-2
3/4 2-8	3/4 2-8
B3/4 2-4	B3/4 2-4

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 1.55 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:

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

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

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.

POWER DISTRIBUTION LIMITS

3/4.2.3 NUCLEAR ENTHALPY HOT CHANNEL FACTOR -

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

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

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.

$F^N_{\Delta H}$ will be maintained within its limits provided conditions a. though 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.

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 $F^N_{\Delta H}$.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 109 TO FACILITY OPERATING LICENSE NO. NPF-2

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-348

1.0 INTRODUCTION

By letter dated June 17, 1994, Southern Nuclear Operating Company (SNC) submitted a request for changes to the Joseph M. Farley Nuclear Plant, Unit 1 (Farley 1), Technical Specification (TS) to (1) increase the limit of the nuclear enthalpy rise hot channel factor ($F^N_{\Delta H}$) for VANTAGE 5 fuel as given in TS 3.2.3, from 1.65 to 1.70 (for full power operation) and (2) change the associated action statement to more closely follow the guidance of the Westinghouse improved standard TS (NUREG-1431) for this specification. Also proposed were changes to the Bases for TS 2.11 (Safety Limits) to reflect the $F^N_{\Delta H}$ increase, and the Bases for TS 3.2.3 (Nuclear Enthalpy Hot Channel Factor) to reflect the changes to the action statement. The current Farley 1 core contains both VANTAGE 5 and LOPAR fuel assemblies. The change is proposed because increased margin is available for VANTAGE 5 fuel and it provides additional flexibility in core design and operation, including the current cycle in which measured $F^N_{\Delta H}$ values have closely approached the 1.65 limit.

2.0 EVALUATION

To justify the proposed change, SNC has evaluated all relevant transient/accident analyses, and has reanalyzed the large break loss-of-coolant accident (LBLOCA) and fuel handling accident after determining that all current analyses of events involving $F^N_{\Delta H}$, except for these two events, bound the effects of a 1.70 $F^N_{\Delta H}$.

The most limiting LBLOCA was reanalyzed using the Westinghouse 1981 evaluation model with BART and BASH, which is the current model of record for Farley 1 VANTAGE 5 fuel. The revised peak clad temperature is 1957 °F. The metal-water reaction amounts are also within 10 CFR Part 50 limits.

The fuel handling accident analyses was determined to be the only radiological event not already bounding the use of a 1.70 $F^N_{\Delta H}$. The accident was reanalyzed with the new value and all acceptance criteria were met.

It was determined that the increase in $F^N_{\Delta H}$ will cause no significant increase in dose above the refueling canal and spent fuel pool. Thermal calculations were done for the fuel pool and no significant increase in clad temperature was found.

The staff review has concluded that SNC has appropriately examined the current status of Farley 1 safety analyses with a full power $F^N_{\Delta H}$ value of 1.70 and has provided appropriate analyses for events and conditions not already covered by existing analyses.

The SNC has also proposed to change the action section of TS 3.2.3. The changed statement closely parallels the action statement of the new Westinghouse Standard TS (NUREG-1431). The primary effect of the change is to increase the time to reduce power to less than 50 percent, when $F^N_{\Delta H}$ is above the (power dependent) limit, from 2 to 4 hours. The time change and the language of the statement are in accordance with the staff approved version in NUREG-1431 and are acceptable.

3.0 SUMMARY

The staff has reviewed the information submitted by SNC for Farley 1 proposing TS changes relating to $F^N_{\Delta H}$. Based on this review, it has been concluded that appropriate information was submitted and the proposed changes to $F^N_{\Delta H}$ and associated TS 3.2.3 action statements 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 amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (59 FR 32249). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

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

Principal Contributor: H. Richings

Date: July 22, 1994