October 1, 1992

Docket No. 50-348

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Mr. W. G. Hairston, III
Executive Vice President
Southern Nuclear Operating
Company, Inc.
Post Office Box 1295
Birmingham, Alabama 35201-1295

Dear Mr. Hairston:

SUBJECT: ISSUANCE OF AMENDMENT NO. 94 TO FACILITY OPERATING LICENSE NO. NPF-2 REGARDING STEAM GENERATOR PRIMARY-TO-SECONDARY LEAKAGE LIMIT - JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1 (TAC NO. M82034)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 94 to Facility Operating License NPF-2 for the Joseph M. Farley Nuclear Plant, Unit 1. The amendment consists of changes to the Technical Specifications in response to your submittal dated October 29, 1991, as supplemented July 1, 1992.

The amendment changes the Technical Specifications to reduce the steam generator primary-to-secondary leakage limit. The total primary-to-secondary leakage through all steam generators is lowered from one gallon per minute (1440 gallons/day) to 420 gallons per day. The limit through any one steam generator is reduced from 500 gallons per day to 140 gallons per day.

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

Sincerely,

Original signed by:

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

070005

Enclosures:
1. Amendment No. 94 to NPF-2
2. Safety Evaluation

cc w/enclosures:
See next page

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|---|---|---|---|---|
| LA:PD21:DRPE | PE:PD21:DRPE | PM:PD21:DRPE | OGC | D:PD21:DRPE |
| PAnderson | MWebb:dt | SHoffman | | EAdensam |
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Joseph M. Farley Nuclear Plant

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AMENDMENT NO. 94 TO FACILITY OPERATING LICENSE NO. NPF-2 - FARLEY, UNIT 1

Docket File
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cc: Farley Service List

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

DOCKET NO. 50-348

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 94
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 October 29, 1991, as supplemented July 1, 1992, 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. 94 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 1, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 94

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 |
|---|---|
| 3/4 4-17 | 3/4 4-17 |
| B 3/4 4-3 | B 3/4 4-3 |
| B 3/4 4-4 | B 3/4 4-4 |

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.7.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 420 gallons per day total primary-to-secondary leakage through all steam generators and 140 gallons per day through any one steam generator,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 31 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 $\pm$ 20 psig.

f. The maximum allowable leakage of any Reactor Coolant System Pressure Isolation Valve shall be as specified in Table 3.4-1 at a pressure of 2235 $\pm$ 20 psig.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the limit specified in Table 3.4-1, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.7.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment air cooler condensate level system or containment atmosphere gaseous radioactivity monitor at least once per 12 hours.

## 3/4.4.6 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage - 140 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operational leakage of this magnitude can be readily detected by existing Farley Unit 1 radiation monitors. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged or repaired.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or repair will be required for all tubes with imperfections exceeding 40% of the tube nominal wall thickness. If a sleeved tube is found to have through wall penetration of greater than or equal to 31% for the mechanical sleeve and 37% for the laser welded sleeve of sleeve nominal wall thickness in the sleeve, it must be plugged. The 31% and 37% limits are derived from R.G. 1.121 calculations with 20% added for conservatism. The portion of the tube and the sleeve for which indications of wall degradation must be evaluated can be summarized as follows:

a. Mechanical

   1. Indications of degradation in the entire length of the sleeve must be evaluated against the sleeve plugging limit.

   2. Indication of tube degradation of any type including a complete guillotine break in the tube between the bottom of the upper joint and the top of the lower roll expansion does not require that the tube be removed from service.

## 3/4.4.7 REACTOR COOLANT SYSTEM LEAKAGE

### 3/4.4.7.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

### 3/4.4.7.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 31 GPM with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the accident analyses.

The surveillance requirements for RCS Pressure Isolation Valves provide added assurance of valve integrity, thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS Pressure Isolation valves is IDENTIFIED LEAKAGE and will be considered a portion of the allowed limit.

The total steam generator tube leakage limit of 420 gallons per day for all steam generators ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. A 1 GPM limit is consistent with the assumptions used in the analysis of these accidents. The 140 gallons per day leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## RELATED TO AMENDMENT NO. 94 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 October 29, 1991, as supplemented July 1, 1992, the Southern Nuclear Operating Company, Inc. (the licensee), submitted a request for changes to Joseph M. Farley Nuclear Plant (Farley), Unit 1, Technical Specification (TS) Section 3.4.7.2 and Bases Sections 3/4.4.6 and 3/4.4.7.2. The requested changes would reduce the steam generator (SG) primary-to-secondary leakage limits. The change lowers the primary-to-secondary leakage limit through all SGs from one gallon per minute (gpm) (1440 gallons per day (gpd)) to 420 gpd, and lowers the primary-to-secondary leakage limit through any one SG from 500 gpd to 140 gpd. The July 1, 1992, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

## 2.0 EVALUATION

Inservice inspections of the steam generators at Farley, Unit 1, have revealed the presence of circumferentially oriented defects in the steam generator tubing. The licensee has proposed a change to its TS to adopt a more restrictive limit for primary-to-secondary leakage based on the discovery of these circumferentially oriented defects. The NRC staff reviewed the licensee's proposed changes and subsequently requested additional information from the licensee by letter dated June 3, 1992. The licensee responded to this request by letter dated July 1, 1992.

The current 500 gpd primary-to-secondary leakage limit per SG is intended to assure that cracks which leak at a rate up to this limit will have an adequate margin of safety, consistent with the guidance in Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," to withstand the load imposed during normal operation and by postulated accidents. The current one gpm primary-to-secondary leakage limit through all SGs is consistent with the assumptions used in the Farley, Units 1 and 2, Final Safety Analysis Report (FSAR) design basis accident analyses.

The circumferential defects are primarily located within the WEXTEX expansion transition area of tubes located in the central, sludge pile area and also in the tangent point and apex locations of the U-bend region of row 1 tubes. Several of the indications that were found could be traced back, using hindsight, to distorted indications in previous outages; however, a growth rate could not be calculated based on the previous inspection results because the earlier bobbin coil inspection data could not be directly correlated to the new rotating pancake coil probe data. Tubes identified with circumferential indications were removed from service by plugging. Row 1 U-bend tubes, in which circumferential cracking was not detected, were heat treated during the last outage to minimize any residual tensile stresses in the tubing and to minimize crack growth in subsequent plant operation.

The licensee has calculated a projected end-of-cycle circumferential crack angle using an estimate for the non-destructive examination (NDE) detection threshold, corrosion growth allowance, and NDE uncertainty. The maximum projected end-of-cycle circumferential crack angle is expected to be within the maximum allowable single, through-wall circumferential crack angle for meeting Regulatory Guide 1.121 guidance for normal, upset, and accident loading conditions with respect to tube burst. If the cracking were to continue at Farley, Unit 1, the licensee expects the development of a segmented crack morphology around the tube circumference with through-wall cracks of various arc lengths separated by ligaments. This expectation is based on pulled tube evidence from several plants with WEXTEX expansions. The leakage from this expected segmented crack morphology (approximately 35 degree cracks separated by 20 mil ligaments) would exceed the proposed 140 gpd per SG primary-to-secondary leakage limit. Similarly, the expected leakage if two of these individual cracks were to grow together (i.e. loss of a ligament) would exceed the 140 gpd limit. Therefore, the licensee concluded that the 140 gpd limit provides assurance that should a circumferential crack propagate at an unexpectedly high rate or if the ligaments separating individual cracks should rupture, sufficient time would exist to shutdown Farley, Unit 1, prior to a SG tube rupture. The licensee is capable of detecting operational leakage of this magnitude with existing radiation monitors.

## 3.0 SUMMARY

The licensee has determined that the proposed 140 gpd primary-to-secondary leakage limit is significantly less than the leakage expected due to the maximum allowable single, through-wall circumferential crack angle which meets Regulatory Guide 1.121 guidance for normal, upset, and accident loading conditions with respect to tube burst. Additionally, the proposed more restrictive primary-to-secondary leakage limit will require unit shutdown at a lower primary-to-secondary leakage level and provides added assurance that the leak rate limit under normal operation will be exceeded prior to exceeding the largest permissible crack. Therefore, the staff concludes that the proposed primary-to-secondary leakage limits should provide adequate assurance of structural and leakage integrity of the steam generator tubing at Farley, Unit 1.

## 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 effluent 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 (57 FR 2580). 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: K. Karwoski

Date: October 1, 1992