September 20, 2002

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

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1 RE: ISSUANCE OF AMENDMENT (TAC NO. MB4310)

Dear Mr. Morey:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 157 to Facility Operating License No. NPF-2 for the Joseph M. Farley Nuclear Plant, Unit 1. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated March 4, 2002, as supplemented by letter dated July 11, 2002.

The amendment revises TS 5.5.9.3.a, “Steam Generator Tube Surveillance Program, Inspection Frequencies.” Specifically, the proposed changes would revise the Farley Nuclear Plant, Unit 1 TS to allow a 40-month inspection interval after its first (post-replacement) inservice inspection, rather than after two consecutive inspections resulting in C-1 classification.

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,

/RA/

Frank Rinaldi, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-348

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

cc w/encl: See next page

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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.157
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 March 4, 2002, as supplemented by letter dated July 11, 2002, 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) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 157 , 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

/RA/

John A. Nakoski, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 20, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 157

TO FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
5.5-7	5.5-7
5.5-8	5.5-8
5.5-9	5.5-9
5.5-10	5.5-10

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 157 TO FACILITY OPERATING LICENSE NO. NPF-2

SOUTHERN NUCLEAR OPERATING COMPANY, INC., ET AL.

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-348

1.0 INTRODUCTION

By letter dated March 4, 2002, as supplemented by letter dated July 11, 2002, the Southern Nuclear Operating Company, Inc. et al., submitted a request for changes to the Joseph M. Farley Nuclear Plant, Unit 1, Technical Specifications (TS). The requested changes would revise TS 5.5.9.3.a, "Steam Generator Tube Surveillance Program, Inspection Frequencies." Specifically, the proposed changes would revise the Farley Nuclear Plant, Unit 1 TS to allow a 40-month inspection interval after its first (post-replacement) inservice inspection, rather than after two consecutive inspections resulting in C-1 classification. The July 11, 2002, letter provided clarifying information that did not change the March 4, 2002, application nor the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND AND PROPOSED TECHNICAL SPECIFICATION CHANGE

Farley Nuclear Plant, Unit 1, replaced steam generators during the 1R16 refueling outage that was completed in the Spring of 2000. The replacement steam generators are Westinghouse design, Model 54F, that incorporated significant design improvements, including thermally treated Alloy 690 tubing. Since the replacement of the steam generators, the licensee performed an inservice inspection during the 1R17, Fall 2001, refueling outage. No service induced degradation of the steam generator tubes was identified during this inspection. The licensee stated that the 1R17 inspection results along with the improved Westinghouse replacement steam generator design provide the basis for the proposed TS amendment.

Currently, TS 5.5.9.3.a requires that subsequent inservice inspection of steam generator tubes after the first inservice inspection be performed "at intervals of not less than 12 nor more than 24 calendar months after the previous inspection." In accordance with the extension criteria in TS 5.5.9.3.a, if two consecutive inspections, not including the preservice inspection, result in all inspection results falling into the C-1 category, or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months. A C-1 category is defined as less than 5 percent of the total tubes inspected are degraded (i.e., contain defects greater than or equal to 20 percent throughwall) and none of the inspected tubes are defective (i.e., contain defects greater than or equal to 40 percent throughwall). The proposed TS amendment requests a one time exception to the extension criteria. The licensee proposed adding the following to its TS: "An exception to this Extension Criteria is that for Farley Unit 1 only, a one-time inspection interval extension of a maximum of once per 40 months is allowed for the inspection performed immediately after the Farley 1 1R17 inspection. This is an exception to the Extension Criteria in that the inspection interval is based on the result of only one inspection result falling into the C-1 category." Approval of this TS change request would allow the licensee to omit the inspection of the steam generators during the next scheduled refueling outage, 1R18, scheduled for Spring 2003.

The licensee estimated that approval of this TS change would reduce the radiation dose by approximately 7 person-REM.

3.0 <u>EVALUATION</u>

The staff's evaluation covers: the replacement steam generator's improved design features; the first inservice, Fall 2001, steam generator inspection scope; the first inservice, Fall 2001, steam generator inspection results; and related industry operating experience.

3.1 Steam Generator Design Improvements

The replacement steam generators incorporate both design and material improvements to address problems the industry experienced with the original steam generator design features. Several examples of these improvements are discussed below.

- The replacement steam generator tubing is made of thermally-treated Alloy 690 material that has an increased resistance to stress corrosion cracking (SCC) over the original mill annealed Alloy 600 steam generator tubing. The thermally-treated Alloy 690 material has a 13 percent higher chromium content and correspondingly reduced nickel content than the original mill annealed Alloy 600 steam generator tubing. The higher chromium content reduces the degree of sensitization of the material, thus increasing resistance to corrosion attack. In addition, heat treatment of the Alloy 690 material was optimized for SCC resistance. Extensive laboratory tests have been performed by the industry that has demonstrated that thermally-treated Alloy 690 material is superior to mill annealed Alloy 600 material in its resistance to both primary and secondary system SCC, pitting and general corrosion.

- An enhanced anti-vibration bar (AVB) design provides for more stable tube bundle, and limits potential for both wear and high cycle fatigue of tubes.

- The replacement steam generator tube support plate (TSP) material is Type 405 stainless steel that shows improved corrosion resistance over the carbon steel TSPs used in the original steam generators. Corrosion resistant tube support plate material limits the potential for crevice corrosion product buildup, and subsequent denting and degradation of the steam generator tube.

- The quatrefoil shaped cutouts in the TSPs improve axial fluid flow within the tube bundle and minimize tube-to-tube support contact area.

- Full depth hydraulic tube expansions minimize the depth of the crevice between the tubes and the top-of-the-tubesheet. The full depth expansion minimizes the accumulation of contaminants in the tubesheet crevice and the hydraulic expansion process minimizes the residual stresses in the steam generator tubes. Both these improvements reduce the susceptibility of the steam generator tube within the tubesheet to corrosion.

- The number and types of external shell penetrations have been increased. This provides for better secondary side access for sludge and foreign object removal capabilities.

The staff finds the replacement steam generator's design and material improvements should enhance the steam generator tubing's resistance to service induced degradation of the type experienced with the original steam generators, especially during the first several cycles of operation.

3.2 First Inservice, Fall 2001, Steam Generator Inspection Scope

The licensee stated that during the Fall 2001 refueling outage (1R17), following the first cycle of operation since the steam generator replacement, 100 percent of the tubing in all three steam generators was inspected full-length (i.e., hot leg tube end to cold leg tube end) with an eddy current probe containing a bobbin coil. In addition, the U-bend region of 100 percent of the low row tubes (rows 1 and 2) and a 20 percent sample of the hot leg top-of-tubesheet (TTS) transitions were inspected in all three steam generators with a rotating pancake coil (RPC) probe containing a plus point (+Point) coil. Additional RPC inspections were performed on possible bobbin indication locations not cleared by a historical review of the pre service inspection results. New or pre-existing signals that exhibited significant change in magnitude were inspected with an RPC probe containing a +Point coil. In the three steam generators, a total of thirty-seven bobbin indications were inspected with the +Point coil.

The staff concluded that the eddy current inspection scope (i.e., bobbin and RPC inspection) during the Fall 2001 outage was comprehensive and supports the requested inspection interval extension.

3.3 First Inservice, Fall 2001, Steam Generator Inspection Results

The licensee stated that although a number of eddy current signals were identified, no service induced degradation of the steam generator tubing was identified during the Fall 2001 (1R17) refueling outage. The licensee determined that the eddy current signals were due to bulges, freespan dings, dents, freespan signals unchanged from the preservice inspection, hot leg TTS transition anomalies within the tubesheet, and bobbin indications not confirmed by RPC examinations. The licensee provided a discussion on the basis used to determine that these signals were acceptable to leave in service. The staff did not identify any concerns.

The licensee also performed inspections looking for foreign objects and degradation due to foreign objects. Inspections performed included: 1) eddy current inspections of all tubes; 2) foreign object search and retrieval of regions most likely to experience high levels of wear should an object be present in the steam generator; and 3) inspections of the material removed via sludge lancing from the steam generators.

The eddy current inspection program included both a 100 percent bobbin coil inspection (good at detecting volumetric indications) and a 20 percent +Point inspection of the hot leg the top of the tubesheet. Possible loose parts signals were reported in two tubes in steam generator C at the TTS on the hot leg. Visual examination did not confirm an object at these locations. The tubes immediately adjacent to these two tubes were all examined with an RPC probe containing a +Point coil. No signs of tube wear or possible loose parts were identified in these examinations.

Inspections of the material removed via sludge lancing identified thirteen small pieces of non-metallic material resembling flexitallic gasket-like material in steam generator A, one metal shaving approximately 1/16-inch long and one piece of wire approximately 3/4-inch long in steam generator B, and one nail (1.25 inches long) and two pieces of flexitallic gasket-like material in steam generator C. No tube wear was associated with any of these foreign objects.

The licensee performed a condition monitoring assessment to evaluate the as-found condition of the steam generator tubes based on eddy current inspection results. The licensee concluded that all performance criteria had been met.

The licensee also performed an operational assessment to evaluate the predicted condition of the steam generator tubing after the proposed extended inspection interval. They concluded that all structural and accident leakage performance criteria are predicted to be met through the end of the next two cycles of operation.

The staff concluded that the inspection results, condition monitoring assessment and operational assessment results provide assurance that unexpected degradation of steam generator tubing has not occurred and is not expected to occur over the proposed inspection interval extension.

3.4 Related Industry Operating Experience

The licensee reviewed industry data for fifty-four plants with steam generators containing thermally treated alloy 690 tubing and determined that no degradation mechanism, other than mechanical wear, has been identified. Of the fifty-four plants, forty-nine were placed in service prior to the Farley Nuclear Plant, Unit 1 replacement steam generators. This supports the licensee’s conclusion that corrosion related degradation is not expected, particularly this early in the life of the Farley Nuclear Plant, Unit 1 steam generators. The staff agrees with this assessment.

With regard to wear, the licensee stated that there have been no reported instances of AVB wear in replacement steam generators with the Westinghouse advanced AVB design incorporated in the Farley Nuclear Plant, Unit 1 steam generators. This industry experience covers sixteen units, eleven of which have completed at least one inservice inspection of their steam generators. These plants have up to 7 effective full power years of operation with replacement steam generators. Based on this information, the licensee concluded there is reasonable assurance that wear indications will not become structurally significant over the proposed inspection interval extension. The staff agrees with this assessment.

The staff concluded that the industry operating experience with replacement steam generators supports the licensee’s proposed inspection interval extension.

The staff evaluated the replacement steam generator's improved design features, the scope and results of the first inservice steam generator inspection, and related industry operating experience as part of their review of the proposal.

The staff concluded that the replacement steam generators incorporate both design and material improvements that are expected to improve the steam generator's tubing resistance to all forms of service induced degradation, especially during the first several cycles of operation. In addition, the comprehensive Fall 2001 inspection scope, the results of the inspection, and the conclusions of the operational assessment indicate the tubing is not experiencing any service induced degradation and can be safely operated during the proposed extension. Lastly, the industry operating experience with both the thermally treated alloy 690 tubing and the improved Westinghouse advanced AVB design provides added assurance that the steam generators can be safely operated over the proposed period of operation without an inspection of the steam generator tubing. Therefore, the staff has determined that the proposed TS changes 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 and changes the surveillance requirements. The NRC staff has determined that the amendment 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 amendment involve no significant hazards consideration, and there has been no public comment on such finding (67 FR 53991). Accordingly, the amendment 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 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: Cheryl Beardslee Khan

Date: September 20, 2002

Joseph M. Farley Nuclear Plant

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