Mr. J. B. Beasley, Jr. July 14, 2003
Vice President - Farley Project
Southern Nuclear Operating
Company, Inc.
Post Office Box 1295
Birmingham, Alabama 35201-1295

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2 RE: REVISING THE STEAM GENERATOR INSPECTION FREQUENCY (TAC NO. MB7938)

Dear Mr. Beasley:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 153 to Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant, Unit 2 (Farley-2). The amendment consists of changes to the Technical Specifications in response to your application dated February 11, 2003.

The amendment revises TS 5.5.9.3.a, "Steam Generator Tube Surveillance Program, Inspection Frequencies." Specifically, the proposed TS changes would allow a 40-month inspection interval for Farley-2 after the completion of the first post-replacement in-service inspection, rather than after the completion of two consecutive inspections resulting in a classification of C-1.

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. 50-364

Enclosures:
1. Amendment No. 153 to NPF-8
2. Safety Evaluation

cc w/encl: See next page

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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. 153
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 February 11, 2003, 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. 153, 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: July 14, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 153

TO FACILITY OPERATING LICENSE NO. NPF-8

DOCKET NO. 50-364

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
5.5-8	5.5-8

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 153 TO FACILITY OPERATING LICENSE NO. NPF-8

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

FARLEY NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-364

1.0 INTRODUCTION

By letter dated February 11, 2003, Southern Nuclear Operating Company, Inc. (the licensee) submitted a proposed amendment to the Technical Specifications (TS) for Farley Nuclear Plant, Unit 2 (Farley-2). The requested changes would revise TS 5.5.9.3.a, "Steam Generator Tube Surveillance Program, Inspection Frequencies." Specifically, the proposed TS changes would allow a 40-month inspection interval for Farley-2 after the completion of the first post-replacement in-service inspection, rather than after the completion of two consecutive inspections resulting in a classification of C-1.

2.0 REGULATORY EVALUATION

Steam generator (SG) replacement for Farley-2 was completed during the Spring 2001 refueling outage. The replacement steam generators (RSG) are Westinghouse Model 54F. The RSG incorporate a number of design and material changes. These changes will be discussed in Section 3.1 of this Safety Evaluation. During the Fall 2002 refueling outage the licensee performed the first post-replacement inservice inspection for Farley-2. No service induced steam generator tube degradation was identified during this inspection. The licensee's bases for the proposed TS amendment are the improved SG design features, improved SG materials, scope and results of the Fall 2002 refueling outage inspection, and related industry experience. The following revision to TS 5.5.9.3.a was proposed:

Current TS 5.5.9.3.a

> ". . . If two consecutive inspections following service under AVT conditions, 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. [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 on only one inspection result falling into the C-1 category.]"

Enclosure

Revised TS 5.5.9.3.a

> "... If two consecutive inspections following service under AVT conditions, 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. **[An exception to this Extension Criteria is for each unit, a one-time inspection interval extension of a maximum of once per 40 months is allowed for the inspection performed immediately after the Unit 1 1R17 inspection and for the inspection performed immediately after the Unit 2 2R15 inspection. These are exceptions to the Extension Criteria in that the inspection interval for each unit is based on the results of only one inspection falling into the C-1 category for that unit.]"**

3.0 <u>TECHNICAL EVALUATION</u>

3.1 <u>Steam Generator Design Features and Material Improvements</u>

Problems associated with the original steam generator design features were addressed by design and material improvements incorporated into the RSGs. Several of the Westinghouse RSG design and material improvements are discussed below.

1. The RSGs consist of thermally-treated Alloy 690 tubing that exhibits greater resistance to stress corrosion cracking (SCC) than the original SG (OSG) tubing material (mill annealed Alloy 600). Thermally-treated Alloy 690 consists of 13 percent more chromium than mill annealed Alloy 600 and correspondingly reduced nickel content. The increased chromium content reduces the material's degree of sensitization, which in turn increases corrosion attack resistance. The increased resistance to SCC that Alloy 690 exhibits is also due to heat treatment optimization. Extensive industry testing demonstrates that thermally-treated Alloy 690 exhibits greater resistance to primary and secondary system SCC, general corrosion, and pitting than mill annealed Alloy 600.

2. Additional stress relief on all U-bends up to a 12-inch centerline radius was performed in addition to the thermal treatment process. Also, the RSG incorporated a larger minimum radius U-bend design. Both the additional stress relief and larger radius decreases the residual stresses in the U-bend region providing additional assurance of cracking resistance.

3. The RSG incorporated an enhanced anti-vibration bar (AVB) design that provides for a more stable tube bundle and limits the potential for both wear and high-cycle fatigue of the SG tubes. The enhanced AVBs are stainless steel Type 405.

4. The carbon steel tube support plates (TSPs) used in the OSG were replaced with stainless steel Type 405 TSPs. The stainless steel Type 405 TSPs exhibit increased resistance to crevice corrosion product buildup and subsequent denting and degradation of the SG tube than the carbon steel TSPs.

5. The TSPs are designed with quatrefoil shaped cutouts that improve the axial fluid flow through the tube bundle and minimize the tube-to-tube support contact area. The improved TSP design also decreases tube dryout and chemical concentration where the tubes pass through the tube support plates.

6. The RSG design enhances the secondary side access for foreign object and sludge removal capabilities by increasing the number and types of external shell penetrations.

7. A sludge collector will be utilized to decrease the sludge deposition rate on the tube bundle, enhancing SG performance and reliability.

8. A stainless steel Type 405 flow distribution baffle plate with an open central region consisting of octafoil shaped holes produces increased flow velocity across the tubesheet and minimizes the sludge deposition zone.

9. The RSG's design incorporates full depth hydraulic tube expansions that minimize the crevice depth between the tubes and the top-of-the-tubesheet. The full depth expansion decreases contaminant accumulation within the tubesheet crevice, while the hydraulic expansion reduces the residual stresses in the SG tubes.

The NRC staff concludes that the RSG design and material improvements support the licensee's request for an inspection interval extension for Farley-2.

3.2 Fall 2002 Inservice Inspection Scope and Results

The licensee reported that during the first post-replacement inservice inspection, Fall 2002, 100 percent of the inservice tubes in all three SGs were tested full-length (i.e., hot leg tube end to cold leg tube end, including the U-bends) with a bobbin coil eddy current probe. Also, for all three SGs, 100 percent of the low row tubes (rows 1 and 2) in the U-bend region and approximately 20 percent of the hot leg top-of-tubesheet expansion transitions were inspected with +Point™ rotating probes. All possible bobbin probe indications not resolved by historical review of the pre-service inspection results were inspected with a +Point™ probe for further signal characterization. The +Point™ probes were used to inspect new or pre-existing bobbin signals that exhibited a significant magnitude change.

As a result of the inspection, no degraded or defective tubes were reported. A C-1 classification (as defined by technical specification) was assigned to each SG. The licensee reported that no service induced defects <20 percent through wall were detected or sized. No tubes were plugged due to the inspection results.

To evaluate the predicted condition of the tubing for the SGs at the end of the proposed inspection interval extension, the licensee performed an operational assessment. The licensee concluded that the structural and accident leakage integrity performance criteria would be met at the end of two cycles of operation (i.e., the next planned SG inspection).

The NRC staff concluded that the inspection scope and results, the condition monitoring assessment, and the operational assessment provide assurance that SG tube integrity will be maintained over the proposed inspection interval extension.

3.3 Related Industry Experience

Sludge lancing was performed for all three SGs on the secondary side tubesheet region during the Fall 2002 refueling outage. Following the sludge lancing process, foreign objects in this region were identified by an inspection video. No foreign objects detrimental to the SG tubing were identified. As part of the inspection process, minor debris was removed from the SGs. Based on eddy current inspections, the licensee determined the removed debris did not produce any tube wear indications. The eddy current inspection did not identify any loose parts. The licensee concluded that the SGs contain no known foreign objects at this time.

Some tubes contained pre-existing dings that were not attributed to service induced degradation and the tubes were left in service. The licensee’s decision to leave these indications in service was based on either the signal being identified during pre-service inspections or the signal does not exhibit flaw like characteristics and was identified in the pre-service data based on re-review of the data. Historical review of these signals were performed to confirm existence in the pre-service inspection records.

A condition monitoring assessment was performed by the licensee in order to evaluate the condition of the tubing for the three SGs based on the eddy current results. Based on the Fall 2002 outage inspection results, the licensee concluded that all performance criteria had been met.

The licensee reported that there are 54 plants consisting of SGs with thermally-treated Alloy 690 tubing that contain no degradation mechanism other than mechanical wear. Forty-nine (49) of the 54 plants were placed in service prior to the Farley-2 RSG being placed in service. This supports the licensee’s conclusion that the Farley-2 RSGs will not experience degradation due to corrosion this early in their life. The licensee also reviewed the operating experience from 16 units with the Westinghouse advanced AVB design and found that there has been no reported instances of AVB wear. Eleven (11) of the 16 units have completed at least one post-replacement inservice inspection. The licensee concluded there is reasonable assurance that during the proposed inspection interval extension, any wear indications that may develop in the Farley-2 SGs will not become structurally significant.

The NRC staff concluded that the licensee provided sufficient related industry experience that supports the licensee’s proposed inspection interval extension for Farley-2.

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 (68 FR 25657). 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

After careful consideration of the improved SG design features and SG materials, scope and results of the Fall 2002 refueling outage inspection, and related industry experience, the staff has concluded that the proposed TS changes are acceptable.

Principal Contributor: L. Miller, EMCB/DE

Date: July 14, 2003

Joseph M. Farley Nuclear Plant

cc:

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