March 24, 1997

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

SUBJECT: ISSUANCE OF AMENDMENT - JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1 (TAC NO. M97510)

Dear Mr. Morey:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 124 to Facility Operating License No. NPF-2 for the Joseph M. Farley Nuclear Plant, Unit 1. The amendment changes the Technical Specifications (TS) in response to your submittal dated December 26, 1996, as supplemented by letters dated February 6, March 7, and March 21, 1997.

The amendment changes TS 3/4.4.6, "Steam Generators" and associated Bases to implement the voltage-based alternate repair criteria for steam generator tubes in Farley Unit 1 in accordance with Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking."

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:

Jacob I. Zimmerman, Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-348

Enclosures: 1. Amendment No. 124 to NPF-2
2. Safety Evaluation

cc w/encls: See next page

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COPY	YES NO	YES NO	YES NO	YES NO	

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

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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. 124
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 December 26, 1996, as supplemented by letters dated February 6, March 7, and March 21, 1997, 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. 124, 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

Kalton N. Jabbour for

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: March 24, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 124

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
3/4 4-10	3/4 4-10
3/4 4-12	3/4 4-12
3/4 4-12a	3/4 4-12a
-----	3/4 4-12b
3/4 4-13	3/4 4-13
B 3/4 4-3	B 3/4 4-3
B 3/4 4-3a	B 3/4 4-3a
-----	B 3/4 4-3b

1. All nonplugged tubes that previously had detectable wall penetrations greater than 20%.

2. Tubes in those areas where experience has indicated potential problems.

3. At least 3% of the total number of sleeved tubes in all three steam generators or all of the sleeved tubes in the generator chosen for the inspection program, whichever is less. These inspections will include both the tube and the sleeve.

4. A tube inspection (pursuant to Specification 4.4.6.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube or sleeve inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

5. Indications left in service as a result of application of the tube support plate voltage-based repair criteria shall be inspected by bobbin coil probe during all future refueling outages.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

2. The inspections include those portions of the tubes where imperfections were previously found.

d. Implementation of the steam generator tube/tube support plate repair criteria requires a 100 percent bobbin coil inspection for hot-leg and cold-leg tube support plate intersections down to the lowest cold-leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20 percent random sampling of tubes inspected over their full length.

The results of each sample inspection shall be classified into one of the following three categories:

4.4.6.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube or sleeve from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube or sleeve.

3. Degraded Tube means a tube, including the sleeve if the tube has been repaired, that contains imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube or sleeve wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube or sleeve containing a defect is defective.

6. Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be repaired (i.e., sleeved) or removed from service by plugging and is greater than or equal to 40% of the nominal tube wall thickness. For a tube that has been sleeved with a mechanical joint sleeve, through wall penetration of greater than or equal to 31% of sleeve nominal wall thickness in the sleeve requires the tube to be removed from service by plugging. For a tube that has been sleeved with a welded joint sleeve, through wall penetration greater than or equal to 37% of sleeve nominal wall thickness in the sleeve between the weld joints requires the tube to be removed from service by plugging. This definition does not apply to tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to 4.4.6.4.a.11 for the repair limit applicable to these intersections.

7. Unserviceable describes the condition of a tube or sleeve if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.6.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg. For a tube that has been repaired by sleeving, the tube inspection should include the sleeved portion of the tube.

9. Tube Repair refers to mechanical sleeving, as described by Westinghouse report WCAP-11178, Rev. 1, or laser welded sleeving, as described by Westinghouse report WCAP-12672, which is used to maintain a tube in service or return a tube to service. This includes the removal of plugs that were installed as a corrective or preventive measure.

10. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

11. Tube Support Plate Repair Limit is used for the disposition of an alloy 600 steam generator tube for continued service that is experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the repair limit is based on maintaining steam generator tube serviceability as described below:

    a. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltage less than or equal to the lower voltage repair limit [2.0 volts], will be allowed to remain in service.

    b. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit [2.0 volts], will be repaired or plugged except as noted in 4.4.6.4.a.11.c below.

c. Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit [2.0 volts], but less than or equal to the upper voltage repair limit*, may remain in service if a rotating probe inspection does not detect degradation. Steam generator tubes, with indications of outside diameter stress corrosion cracking degradation with a bobbin voltage greater than the upper voltage repair limit*, will be plugged or repaired.

d. If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in 4.4.6.4.a.11.a, 4.4.6.4.a.11.b, and 4.4.6.4.a.11.c. The mid-cycle repair limits are determined from the following equations:

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr \left[\frac{CL - \Delta t}{CL} \right]}$$

$$V_{MLRL} = V_{MURL} - [V_{URL} - V_{LRL}] \left[\frac{CL - \Delta t}{CL} \right]$$

where:

V_{URL}	=	upper voltage repair limit
V_{LRL}	=	lower voltage repair limit
V_{MURL}	=	mid-cycle upper voltage repair limit based on time into cycle
V_{MLRL}	=	mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle
Δt	=	length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented
CL	=	cycle length (the time between two scheduled steam generator inspections)
V_{SL}	=	structural limit voltage
Gr	=	average growth rate per cycle length
NDE	=	95-percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20-percent has been approved by NRC)

Implementation of these mid-cycle repair limits should follow the same approach as in TS 4.4.6.4.a.11.a, 4.4.6.4.a.11.b, and 4.4.6.4.a.11.c.

* The upper voltage repair limit is calculated according to the methodology in Generic Letter 95-05 as supplemented.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair of all tubes exceeding the plugging or repair limit) required by Table 4.4-2.

4.4.6.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission within 15 days of the completion of the plugging or repair effort.

b. The complete results of the steam generator tube and sleeve inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1. Number and extent of tubes and sleeves inspected.
2. Location and percent of wall-thickness penetration for each indication of an imperfection.
3. Identification of tubes plugged or repaired.

c. Results of steam generator tube inspections which fall into Category C-3 shall be considered a REPORTABLE EVENT and shall be reported pursuant to 10CFR50.73 prior to resumption of plant operation. The written report shall provide a description of investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence.

d. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the NRC staff prior to returning the steam generators to service (Mode 4) should any of the following conditions arise:

1. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steam line break) for the next operating cycle.

2. If circumferential crack-like indications are detected at the tube support plate intersections.

3. If indications are identified that extend beyond the confines of the tube support plate.

4. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

5. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1×10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.

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.

The voltage-based repair limits of 4.4.6 implement the guidance in GL 95-05 and are applicable only to Westinghouse-designed steam generators (SGs) with outside diameter stress corrosion cracking (ODSCC) located at the tube-to-tube support plate intersections. The voltage-based repair limits are not applicable to other forms of SG tube degradation nor are they applicable to ODSCC that occurs at other locations within the SG. Additionally, the repair criteria apply only to indications where the degradation mechanism is dominantly axial ODSCC with no significant cracks extending outside the thickness of the support plate. Refer to GL 95-05 for additional description of the degradation morphology.

Implementation of 4.4.6 requires a derivation of the voltage structural limit from the burst versus voltage empirical correlation and then the subsequent derivation of the voltage repair limit from the structural limit (which is then implemented by this surveillance).

The voltage structural limit is the voltage from the burst pressure/bobbin voltage correlation at the 95-percent prediction interval curve reduced to account for the lower 95/95-percent tolerance for tubing material properties at 650 °F (i.e., the 95-percent LTL curve). The voltage structural limit must be adjusted downward to account for potential flaw growth during an operating interval and to account for NDE uncertainty. The upper voltage repair limit, V_{URL}, is determined from the structural voltage limit by applying the following equation:

$$V_{URL} = V_{SL} - V_{Gr} - V_{NDE}$$

where V_{Gr} represents the allowance for flaw growth between inspections and V_{NDE} represents the allowance for potential sources of error in the measurement of the bobbin coil voltage. Further discussion of the assumptions necessary to determine the voltage repair limit is contained in GL 95-05.

The mid-cycle equation in 4.4.6.4.a.11.d should only be used during unplanned inspections in which eddy current data is acquired for indications at the tube support plates.

4.4.6.5 implements several reporting requirements recommended by GL 95-05 for situations in which the NRC wants to be notified prior to returning the SGs to service. For the purposes of this reporting requirement, leakage and conditional burst probability can be calculated based on the as-found voltage distribution rather than the projected end-of-cycle voltage distribution (refer to GL 95-05 for more information) when it is not practical to complete these calculations using the projected EOC voltage distributions prior to returning the SGs to service. Note that if leakage and conditional burst probability were calculated using the measured EOC voltage distribution for the purposes of addressing the GL section 6.a.1 and 6.a.3 reporting criteria, then the results of the projected EOC voltage distribution should be provided per the GL section 6.b(c) criteria.

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. The tube plugging limit continues to apply to the portion of the tube in the entire upper joint region and in the lower roll expansion. As noted above, the sleeve plugging limit applies to these areas also.

4. The tube plugging limit continues to apply to that portion of the tube above the top of the upper joint.

b. Laser Welded

1. Indications of degradation in the length of the sleeve between the weld joints must be evaluated against the sleeve plugging limit.

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

3. At the weld joint, degradation must be evaluated in both the sleeve and tube.

4. In a joint with more than one weld, the weld closest to the end of the sleeve represents the joint to be inspected and the limit of the sleeve inspection.

5. The tube plugging limit continues to apply to the portion of the tube above the upper weld joint and below the lower weld joint.

Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect wastage type degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission pursuant to 10 CFR 50.73 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision to the Technical Specifications, if necessary.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 124 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 December 26, 1996, as supplemented by letters dated February 6, March 7, March 21, 1997, Southern Nuclear Operating Company, Inc., et al. (the licensee), submitted for staff review a license amendment to change the Technical Specifications (TS) for the Joseph M. Farley Nuclear Plant, Unit 1. The licensee proposed to implement permanent voltage-based alternate repair criteria for steam generator tubes in the TS. The proposed alternate repair criteria would allow steam generator tubes having outside diameter stress corrosion cracking (ODSCC) that is predominately axially oriented and confined within the tube support plates to remain in service on the basis of bobbin coil voltage response. The NRC guidance on the alternate repair criteria is specified in Generic Letter (GL) 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," August 3, 1995.

In addition to the TS amendment, the licensee proposed the following additional items: (1) to include 50% of the bobbin coil indications not confirmed by rotating pancake coil in the determination of the beginning-of-cycle voltage distribution instead of 100% as specified in GL 95-05; (2) proposed to use probability of detection that is voltage dependent, instead of a constant 60%; and (3) revising the steam line break leakage limit from 11.4 gpm to 20 gpm. By letter dated January 27, 1997, the staff requested from the licensee additional information related to items (1) and (2). In the licensee's February 6, 1997, response, they requested that the proposed TS changes be approved without approval of items (1) and (2). Also, by letter dated March 21, 1997, the licensee withdrew their request for approval of item (3).

By letters dated February 6, March 7, and March 21, 1997, the licensee submitted additional information to clarify the changes to the proposed repair criteria, which did not change the scope of the December 26, 1996, application and the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

Steam generator tube flaw acceptance criteria (i.e., plugging limits) are specified in the plant TS. The traditional strategy for achieving adequate structural and leakage integrity of the tubes has been to establish a minimum wall thickness requirement in accordance with NRC Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes." Development of minimum wall thickness requirements to satisfy RG 1.121 was governed by analyses assuming a uniform thinning of the tube wall. This assumed degradation mode is inherently conservative for most other forms of steam generator tube degradation. Conservative repair limits may lead to plugging tubes with adequate structural and leakage integrity for further service.

The staff developed generic criteria for voltage-based limits for ODSCC confined within the thickness of the tube support plates. The staff published several conclusions regarding voltage-based repair criteria in draft NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes" and in a draft GL titled "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes." The latter document was published for public comment in the *Federal Register* on August 12, 1994 (59 FR 41520). On August 3, 1995, the staff issued GL 95-05 that took into consideration public comments on the draft GL cited above, domestic operating experience under the voltage-based repair criteria, and additional data made available from European nuclear power plants.

The guidance of GL 95-05 does not set depth-based limits on predominantly axially oriented ODSCC at tube support plate locations; rather it relies on empirically derived correlations between a nondestructive inspection parameter, the bobbin coil voltage, and tube burst pressure and leak rate. The staff recognizes that although the total tube integrity margins may be reduced following application of a voltage-based repair criteria, the guidance in GL 95-05 ensures structural and leakage integrity continue to be maintained at acceptable levels consistent with the requirements of 10 CFR Part 50 and the guideline values in 10 CFR Part 100. Since the voltage-based repair criteria do not incorporate a minimum tube wall thickness requirement, there is the possibility for tubes with through-wall cracks to remain in service. Because of the increased likelihood of such flaws, the staff included provisions for augmented steam generator tube inspections and more restrictive operational leakage limits.

GL 95-05 specifies, in part, that: (1) the repair criteria is only applicable to predominantly axially oriented ODSCC located within the bounds of the tube support plates; (2) licensees perform an evaluation to confirm that the steam generator tubes will retain adequate structural and leakage integrity from cycle to cycle; (3) licensees adhere to specific inspection criteria to ensure consistency in methods between inspections; (4) tubes must be periodically removed from the steam generators, examined, and destructively tested to verify the morphology of the degradation and provide additional data for structural and leakage integrity evaluations; (5) the operational leakage limit be reduced; (6) licensees implement an operational leakage monitoring program; and (7) specific reporting requirements shall be incorporated into the plant technical specifications.

The licensee has applied for the voltage-based alternate repair criteria on an interim basis and the staff has approved the licensee's interim repair criteria for the Farley Unit 1 TS as documented in license Amendment No. 95, issued on October 8, 1992; license Amendment No. 106 on April 5, 1994; and license Amendment No. 117 on September 28, 1995. Each interim criteria amendment was approved for a specific operating cycle. The proposed permanent alternate repair criteria will replace the interim criteria and will eliminate the need for applying periodic license amendments for the tube repair criteria.

Farley Unit 1 uses three Westinghouse model 51 steam generators. The tubes were fabricated using mill annealed alloy 600 material. Each steam generator has 3,388 tubes and the nominal outside diameter for each tube is 7/8 inch.

3.0 EVALUATION

The licensee has stated that it will comply with the guidance in GL 95-05 for its proposed permanent alternate repair criteria. In addition, the licensee has proposed to incorporate verbatim the model technical specifications in GL 95-05 into the Farley Unit 1 TS. The major issues related to the licensee's implementation of the alternate repair criteria are discussed below.

3.1 Tube Repair Limits

The proposed criteria will (1) permit indications confined to within the thickness of the tube support plates with bobbin voltages less than or equal to 2.0 volts to remain in service; (2) permit indications confined to within the thickness of the tube support plates with bobbin voltages greater than 2.0 volts but less than or equal to the upper voltage limit to remain in service if a motorized rotating pancake coil probe or acceptable alternative inspection does not detect degradation; and (3) require indications confined to within the thickness of the tube support plates with bobbin voltages greater than the upper voltage limit be plugged or repaired.

The proposed lower voltage limit of 2.0 volts is based on the use of a correlation between the burst pressure and the bobbin coil voltage of pulled tube and model boiler data and is consistent with the recommended value specified in GL 95-05 for 7/8-inch steam generator tubing. The upper voltage limit is based on the lower 95 percent prediction interval of the burst pressure versus bobbin voltage correlation, adjusted for lower bound material properties evaluated at the 95 percent confidence level. This voltage is further reduced to account for uncertainty in the nondestructive examination technique and flaw growth over the next operating cycle. Because licensees periodically update the burst pressure and bobbin voltage database when the destructive test data from pulled tube are available, the upper voltage limit may vary as additional data is included in the correlation.

Section 1.b.1 of Attachment 1 to GL 95-05 specifies that the repair criteria do not apply to tube-to-tube support plate intersections where the tube with degradation may potentially collapse or deform as a result of the combined postulated loss-of-coolant accident and safe shutdown earthquake loadings. Licensees should perform or reference an analysis that identifies which intersections are to be excluded. The licensee submitted an analysis, WCAP-12871, Revision 2, as a part of its application for the interim alternate repair criteria on May 28, 1993. As a result of the licensee's analysis, no tubes need to be excluded from application of the voltage-based repair criteria. The staff found the licensee's assessment acceptable.

3.2 Inspection Issues

Section 3.c.3 of Attachment 1 to GL 95-05 specifies guidance in regard to probe wear. The licensee proposed to use an alternative to GL 95-05. The industry approach, developed through the Nuclear Energy Institute, is such that if any of the probe wear standard signal amplitudes prior to probe replacement exceed the ± 15 percent limit, all tubes having indications with voltage responses measured at 75 percent or greater of the lower repair limit must be reinspected with a bobbin probe satisfying the ± 15 percent wear standard criterion. The voltages from the reinspection should be used as the basis for tube repair. The NRC staff completed a review of the Nuclear Energy Institute proposed alternative method and concluded that the approach is acceptable as discussed in the letter from Brian Sheron of the NRC to Alex Marion of the Nuclear Energy Institute dated March 18, 1996. Therefore, the licensee's proposal to follow the industry approach to address bobbin coil probe wear is acceptable.

In the laboratory and field studies supporting the alternative probe wear criteria, the correlation of worn probe voltages with new probe voltages shows that for all significant voltage levels, the worn probe voltages are never less than 75% of the new probe voltage as discussed in the letter from Alex Marion of the Nuclear Energy Institute to Brian Sheron of the NRC dated January 23, 1996. However, in a 90-day inspection report from Byron Unit 1 dated September 9, 1996, a comparison made between the worn probe voltage and the new probe voltage resulted in a few indications where the worn probe voltage was substantially less than 75% of the new probe voltage. The licensee for Byron Unit 1 evaluated these indications and concluded that the criteria to retest tubes with worn probe voltages above 75% of the repair limit is adequate and generally conservative due to the average trend for worn probe volts to exceed new probe voltages. Comparison of the actual and projected end-of-cycle voltages did not show anything unusual attributable to the alternate probe wear criteria. The staff concludes that the aforementioned probe wear results do not indicate an immediate need to modify the industry alternative probe wear criteria. However, the staff will continue to monitor the 90-day inspection reports of licensees using this approach to probe wear.

Section 3.b of Attachment 1 to GL 95-05 specifies guidance for tube inspection using the rotating pancake coil. The licensee stated that it may use a motorized rotating coil probe, e.g., the +Point coil, for dispositioning bobbin coil indications. The staff encourages licensees to use the most sensitive inspection techniques available and as such this proposal is acceptable.

3.3 Structural and Leakage Integrity Assessments

The staff guidance for the implementation of the voltage-based repair criteria focuses on maintaining tube structural integrity during the full range of normal, transient and postulated accident conditions with adequate allowance for eddy current test uncertainty and flaw growth projected to occur during the next operating cycle. Tube structural limits based on RG 1.121 criteria require maintaining a margin of safety of 1.43 against tube failure under postulated accident conditions and maintaining a margin of safety of 3 against burst during normal operation. Because GL 95-05 addresses tubes affected with ODSCC confined to within the thickness of the tube support plate during normal operation, the staff concluded that the structural constraint provided by the tube support plate ensures all tubes to which the voltage-based criteria applies will retain a margin of 3 with respect to burst under normal operating conditions. For a postulated main steam line break accident, however, the tube support plate may displace axially during steam generator blowdown such that the ODSCC affected portion of the tubing may no longer be fully constrained by the tube support plate. Accordingly, it is appropriate to consider the ODSCC affected regions of the tubes as free standing tubes for the purpose of assessing burst integrity under postulated main steam line break conditions.

In order to confirm the structural and leakage integrity of the tube until the next scheduled inspection, GL 95-05 specifies a methodology to determine the conditional burst probability and the total primary-to-secondary leak rate from an affected steam generator during a postulated main steam line break event. To complete GL 95-05 prescribed assessments, the licensee proposes to follow the methodology described in WCAP-14277, Revision 1, "SLB Leak Rate and Tube Burst Probability Analysis Methods for ODSCC at TSP Intersections," dated December 1996. Based on the staff's detailed review of WCAP-14277, Revision 1, including performance of confirmatory calculations, the staff finds the methodology acceptable.

GL 95-05 specifies that the structural and leakage integrity assessments should use the latest data from destructive examinations of tubes removed from licensees' steam generators. The licensee stated that the latest NRC-approved database, using the NRC-approved data exclusion criteria, will be applied to the tube integrity evaluations. For the upcoming Farley Unit 1 inspection and

GL 95-05 specified calculations, the licensee will use the database forwarded to the NRC by Duquesne Light Company for Beaver Valley Unit 1, dated March 27, 1996. The database contains the most currently available tube pull data from industry and also satisfies the exclusion criteria specified in GL 95-05. Therefore, the staff finds that the database submitted by Duquesne Light Company is acceptable for the GL 95-05 calculations for the upcoming Farley Unit 1 inspection.

For the long-term, Nuclear Energy Institute has developed a protocol for updating the steam generator degradation database. The staff will review the adequacy of the protocol. Pending the implementation of an NRC-approved process for updating a generic industry database for steam generator tube degradation, the licensee will provide the NRC with the database it intends to use prior to each refueling outage. The database will include the data from tubes that have been pulled and tested up to 2 months before the plant outage. The staff finds the licensee's proposal acceptable.

GL 95-05 specifies an alternative for licensees to calculate the primary-to-secondary leakage and probability of tube burst given a main steam line break using the projected end-of-cycle voltage distribution. The licensee will perform the calculations on the basis of the projected end-of-cycle distributions. In the event that the growth rate determinations cannot be completed before returning the steam generators to service, the licensee will use the actual end-of-cycle distributions as allowed in Section 2.c of Attachment 1 to GL 95-05. The licensee stated that even if the calculation made before returning the steam generators to service is based on the actual measured voltage distribution, the calculation based on the projected end-of-cycle voltage distribution will be submitted to the NRC in the 90-day report following the outage. This approach is consistent with Section 6.b.(c) of Attachment 1 to GL 95-05 and is acceptable.

3.3.1 Conditional Probability of Burst

The licensee will use the methodology described in Revision 1 of WCAP-14277 for performing a probabilistic analysis to quantify the potential for steam generator tube ruptures given an main steam line break event. The results of the probabilistic analysis will be compared to a threshold value of $1x10^{-2}$ per cycle in accordance with GL 95-05. This threshold value provides assurance that the probability of burst is acceptable considering the assumptions of the calculation and the results of the staff's generic risk assessment for steam generators contained in NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity." Failure to meet the threshold value indicates ODSCC confined to within the thickness of the tube support plate could contribute a significant fraction to the overall conditional probability of tube rupture from all forms of degradation assumed and evaluated as acceptable in NUREG-0844. The NRC staff concludes the licensee's proposed methodology for calculating the conditional burst probability is consistent with the guidance in GL 95-05 and is acceptable.

3.3.2 Accident Leakage

The licensee will use the methodology described in Revision 1 of WCAP-14277 for calculating the steam generator tube leakage from the faulted steam generator during a postulated main steam line break event. The model consists of two major components: (1) a model predicting the probability that a given indication will leak as a function of voltage (i.e., the probability of leakage model); and (2) a model predicting leak rate as a function of voltage, given that leakage occurs (i.e., the conditional leak rate model). The staff concludes that the licensee's proposed methodology for calculating the tube leakage is consistent with the guidance in GL 95-05 and is acceptable.

3.3.3 Primary-to-Secondary Leakage During Normal Operation

When the voltage-based repair criteria is implemented, tubes may have or may develop through-wall or near through-wall cracks during an operational cycle, thus creating the potential for primary-to-secondary leakage during normal operation, transients, or postulated accidents. Postulated accident leak rates were discussed previously.

The staff concludes adequate leakage integrity during normal operation is reasonably assured by the TS limits on allowable primary-to-secondary leakage. GL 95-05 specifies the operational leakage limits of the plant TS should be reduced to 150 gallons per day. Farley Unit 1 TS currently limit the primary-to-secondary leakage through one steam generator to 150 gallons per day. This requirement is consistent with the guidance in GL 95-05 and is, therefore, acceptable.

3.4 Degradation Monitoring

To confirm the nature of the degradation occurring at the tube support plate elevations, tubes are periodically removed from the steam generators for destructive tests. The test data from removed tubes can confirm that the nature of the degradation observed at these locations is predominantly axially oriented ODSCC, provide data for assessing the reliability of the inspection methods, and supplement the existing databases (e.g., burst pressure, probability of leakage, and leak rate). GL 95-05 specifies that at least two tube be removed from steam generators with the objective of retrieving as many intersections as practical (minimum of four intersections) during the plant steam generator inspection outage preceding initial application of the voltage-based repair criteria. On an ongoing bases, additional tube specimen removals (minimum of two intersections) should be obtained at the first refueling outage following 34 effective full power months of operation or at the maximum interval of three refueling outages after the previous tube pull. Alternatively, the licensee may participate in an industry-sponsored tube pull program endorsed by the staff as described in GL 95-05.

The licensee has removed at least four tubes, including the required number of intersections, from the Unit 1 steam generators for burst and leak rate testing and metallographic examination as a part of the interim repair criteria. The metallurgical examination confirmed that the degradation mechanism for the indications at the tube support plates was predominantly axially oriented ODSCC. For the permanent alternate repair criteria, the licensee stated that it will comply with the tube pull guidance in GL 95-05. The staff concludes that the licensee satisfies the tube removal guidance of GL 95-05.

3.5 Technical Specification Changes

The proposed amendment revised TS 3/4.4.6, "Steam Generators" and associated Bases section as part of implementing the voltage-based repair criteria for steam generator tubes. Specifically, the licensee changed the following Surveillance Requirements sections in the TS: TS 4.4.6.2, "Steam Generator Tube Sample Selection and Inspecton;" TS 4.4.6.4, "Acceptance Criteria;" and TS 4.4.6.5, "Reports." The changes incorporate the methodology of calculating the upper voltage repair limit and mid-cycle repair limits. The licensee also changed TS Bases Section 3/4.4.6, "Steam Generators," consistent with these changes as stated above. The staff concludes that the proposed TS changes satisfy the model technical specifications for the voltage-based repair criteria as specified in GL 95-05 will ensure adequate structural and leakage integrity and, therefore, are acceptable.

4.0 STAFF CONCLUSION

The licensee submitted an application for a license amendment to permit the use of the permanent voltage-based repair criteria for steam generator tubes at Farley Unit 1. The staff has reviewed the proposed amendment and concludes that the proposed permanent alternate repair criteria are consistent with GL 95-05 and are acceptable. The staff also concludes that adequate structural and leakage integrity can be assured, consistent with applicable regulatory requirements, for indications to which the voltage-based repair criteria will be applied. The staff's approval of the proposed voltage-based repair criteria is based in part on the licensee being able to successfully demonstrate after each inspection outage the conditional probability of burst and the primary-to-secondary leakage during a postulated main steam line break will be acceptable in accordance with the guidance in GL 95-05. The licensee may incorporate the proposed permanent alternate repair criteria into the TS for Farley Unit 1.

5.0 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.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes the surveillance requirements. 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 (62 FR 4353 dated January 29, 1997). The amendment also changes reporting or recordkeeping requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). 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.

7.0 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.

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