May 20, 1996

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 2 (TAC NO. M95248)

Dear Mr. Morey:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 110 to Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant, Unit 2. On April 22, 1996 Southern Nuclear Operating Company requested and the NRC staff approved Enforcement Discretion for Farley Unit 2. This Technical Specification (TS) amendment, in response to your submittal dated April 23, 1996, changes the TS to be consistent with this Enforcement Discretion previously granted.

The amendment allows steam generator tubes to remain in service with bands of axial degradation in the tube sheet region, for the remainder of Cycle 11, provided sufficient non-degraded tubing remains to satisfy the L^*-type criteria restrictions established by the licensee.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly <u>Federal Register</u> notice.

Sincerely,

Original signed by:

Byron L. Siegel, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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Docket No. 50-364

Enclosures: 1. Amendment No. 110 to NPF-8
2. Safety Evaluation

cc w/encls: See next page

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

May 20, 1996

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Vice President - Farley Project
Southern Nuclear Operating
Company, Inc.
Post Office Box 1295
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Byron L. Siegel, Senior Project Manager
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Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-364

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2. Safety Evaluation

cc w/encls: See next page

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Company, Inc.

Joseph M. Farley Nuclear Plant

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

ALABAMA POWER COMPANY

DOCKET NO. 50-364

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 110
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 April 23, 1996, 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) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 110 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

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: May 20, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 110

TO FACILITY OPERATING LICENSE NO. NPF-8

DOCKET NO. 50-364

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 4-9	3/4 4-9
3/4 4-11	3/4 4-11
3/4 4-12a	3/4 4-12a
3/4 4-12b	3/4 4-12b
3/4 4-13	3/4 4-13
3/4 4-13a	3/4 4-13a
B3/4 4-3a	B3/4 4-3a
B3/4 4-3b	B3/4 4-3b

3/4.4.6 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.6 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing Tavg above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.6.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.6.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.6.2.1 Steam Generator Tube# Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.6.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.6.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators. Selection of tubes to be inspected is not affected by the F*/L*## designation. When applying the exceptions of 4.4.6.2.1.a through 4.4.6.2.1.c, previous defects or imperfections in the area repaired by sleeving are not considered an area requiring re-inspection. The tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

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

When referring to a steam generator tube, the sleeve shall be considered a part of the tube if the tube has been repaired per Specification 4.4.6.4.a.9.

L* Criteria is applicable to Cycle 11 only.

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes or sleeves must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

4.4.6.2.2 Steam Generator F*/L*## Tube Inspection - In addition to the minimum sample size as determined by Specification 4.4.6.2.1, all F*/L* tubes will be inspected within the tubesheet region. The results of this inspection will not be a cause for additional inspections per Table 4.4-2.

4.4.6.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. 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.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.6.3.a; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

L* Criteria is applicable to Cycle 11 only.

b. 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.

F* tubes do not have to be plugged or repaired provided the remainder of the tube within the tubesheet that is above the F* distance is not degraded. The F* distance is equal to 1.79 inches and is measured down from the top of the tubesheet or the bottom of the roll transition, whichever is lower in elevation. Included in this distance is an allowance of 0.25 inch for eddy current elevation measurement uncertainty.

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. This definition does not apply for tubes that meet the F*/L*## criteria. 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 plugging criteria are being applied. Refer to 4.4.6.4.a.14 for the plugging 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.

L* Criteria is applicable to Cycle 11 only.

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. F* Distance is the distance of the expanded portion of a tube which provides a sufficient length of undegraded tube expansion to resist pullout of the tube from the tubesheet. The F* distance is equal to 1.79 inches and is measured down from the top of the tubesheet or the bottom of the roll transition, whichever is lower in elevation.

12. F* Tube is a tube:

a) with degradation equal to or greater than 40% below the F* distance, and b) which has no indication of imperfections greater than or equal to 20% of nominal wall thickness within the F* distance, and c) that remains inservice.

If the above criteria cannot be met, then the L* tube criteria may be applied or the tube must be plugged or repaired.

13. L* Length## is the length of the expanded portion of the tube into the tube sheet from the bottom of the rolled transition or the top of the tube sheet, which ever is lower, that has been determined to be 1.45 inches.

14. L* Tube ##: a) is a tube with degradation equal or greater than 40% below the L* length and not degraded within the L* length; b) the eddy current indication of degradation below the L* length must be determined to be the result of cracks with an orientation no greater than 15 degrees from axial; c) the L* criteria shall be limited to a maximum of 600 tube ends per steam generator; d) tubes qualifying as F* tubes are not classified as L* tubes; e) a minimum of 3.1 inches of the tube into the tubesheet from the top of tubesheet or bottom of the rolled transition, which ever is lower, shall be inspected using rotating pancake coil eddy current technique or an inspection method shown to give equivalent or better information on the orientation and length of axial cracks; f) a minimum aggregate of 2.07 inches of sound roll expansion; g) a maximum crack length of .39 inches; h) a maximum of 5 distinct indications with a single band of tube degradation; and i) that remains in service.

L* Criteria is applicable to Cycle 11 only.

15. Tube Expansion is that portion of a tube which has been increased in diameter by a rolling process such that no crevice exists between the outside diameter of the tube and the hole in the tubesheet.

16. Tube Support Plate Plugging Limit is used for the disposition of a steam generator tube for continued service that is experiencing outside diameter stress corrosion cracking confined within the thickness of the tube support plates. These criteria are applicable for the Eleventh Cycle only. At tube support plate intersections, the repair limit is based on maintaining steam generator tube serviceability as described below:

    a. Degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltage less than or equal to 2.0 volts will be allowed to remain in service.

    b. Degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltage greater than 2.0 volts will be repaired or plugged except as noted in 4.4.6.4.a.14.c below.

    c. 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 2.0 volts but less than or equal to 5.6 volts may remain in service if a rotating pancake coil inspection does not detect degradation. Indications of outside diameter stress corrosion cracking degradation with a bobbin voltage greater than 5.6 volts will be plugged or repaired.

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, repaired or designated F*/L*## in each steam generator shall be reported to the Commission within 15 days of the completion of the inspection, 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 staff prior to returning the steam generator to service (Mode 4) should any of the following conditions arise:

1. If estimated leakage based on the actual end-of-cycle voltage distribution would have exceeded the leak limit (for the postulated main steam line break utilizing licensing basis assumptions) during the previous 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 the calculated conditional burst probability exceeds 1×10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.

L* Criteria is applicable to Cycle 11 only.

L* is similar to F*; however, bands of axial degradation are allowed as long as sufficient non-degraded tubing is available to ensure structural and leakage integrity. L* criterion is only applicable for Unit 2 Cycle 11. Provided below is the Unit 2 Cycle 11 specific L* criterion:

Unit 2 Cycle 11 Specific L* Criterion

Parameter	Value
Minimum distance of SRE	2.07 inches
Maximum number of distinct degradation areas in a band	5
Maximum inclination angle within a single band	15 degrees
Maximum crack length	.39 inches
Minimum distance of SRE from the bottom of the transition roll to the top of the indication	1.45 inches

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. 110 TO FACILITY OPERATING LICENSE NO. NPF-8

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-364

1.0 INTRODUCTION

By letter dated April 23, 1996, the Southern Nuclear Operating Company, Inc., et al. (the licensee), submitted a request for changes to the Joseph M. Farley Nuclear Plant, Unit 2, Technical Specifications (TS). The requested changes would allow steam generator tubes to remain in service with bands of axial degradation in the tube sheet region, for the remainder of Cycle 11, provided sufficient non-degraded tubing remains to satisfy the L^*-type criteria restrictions established by the licensee.

2.0 BACKGROUND

Surveillance Requirement 4.4.6.4.11 of TS 3/4.4.6, "F^* Distance," requires a tube to be repaired if this F^* distance is not satisfied. The F^* distance is the distance of the expanded portion of a steam generator tube which provides a sufficient length of undegraded tube expansion to resist pullout of the tube from the tubesheet. For Farley, this distance equals to 1.79 inches and is measured down from the top of the tubesheet or the bottom of the roll transition, whichever is lower in elevation. As stated in TS 4.4.6.4.12, an F^* tube is a tube that (1) has degradation equal to or greater than 40 percent below the F^* distance, (2) has no indication of greater than or equal to 20 percent of nominal wall thickness within the F* distance, and (3) remains in service. The licensee reported that as a result of the misapplication of the TS requirements, six tubes were left in service that have indications that do not satisfy the above F^* criteria.

By a letter dated April 23, 1996, the licensee requested that the NRC exercise discretion not to enforce compliance with the actions required in the TS on the basis of proposed L^*-type criteria. The L^*-type criteria would allow those tubes that do not satisfy the F^* criteria to remain in service.

In a second letter dated April 23, 1996, the licensee submitted a proposed TS change that utilized an L^*-type criteria that is limited to the remainder of Cycle 11 which is scheduled to end by the fall of 1996. The L^*-type criteria are more restrictive and conservative than the more generic L^* criteria proposed TS change that was submitted by the licensee in a letter dated April 22, 1996.

On April 22, 1996, the staff verbally approved, and by a letter dated April 25, 1996, the staff officially notified the licensee that it was exercising discretion not to enforce compliance with the F* criteria for the period from April 23, 1996, until the issuance of a license amendment for the L*-type criteria.

Farley Unit 2 uses three Westinghouse 51 Series steam generators. The tubes are hard rolled with full expansion in the tubesheet. The tubes have an outside diameter of 7/8 inch with a wall thickness of 0.050 inch. The proposed L*-type criteria defines the L* length as the length of the expanded portion of the tube into the tubesheet from the bottom of the rolled transition or the top of the tubesheet, whichever is lower, and has been determined to be 1.45 inches.

For Farley Unit 2, the proposed TS L*-type criteria define the L* tube as (1) a tube with degradation equal or greater than 40% below the L* length and not degraded within the L* length; (2) the eddy current indication of degradation below the L* length must be determined to be the result of cracks with an orientation no greater than 15 degrees from axial; (3) the L* criteria shall be limited to a maximum of 600 tube ends per steam generator; (4) tubes qualifying as F* tubes are not classified as L* tubes; (5) a minimum of 3.1 inches of the tube into the tubesheet from the top of tubesheet or bottom of the rolled transition, whichever is lower, shall be inspected using rotating pancake coil eddy current technique or an inspection method shown to give equivalent or better information on the orientation and length of axial cracks; (6) a minimum aggregate of 2.07 inches of sound roll expansion; (7) a maximum crack length of 0.39 inches; and (8) a maximum of 5 distinct indications with a single band of tube degradation.

3.0 EVALUATION

3.1 Discussion

General Design Criterion (GDC) 14 of Appendix A to 10 CFR Part 50 requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. Draft Regulatory Guide (RG) 1.121 provides guidance on an acceptable method for establishing repair limits for steam generator tubes. Surveillance requirements in the plant TS require periodic inspections of steam generator tubes. If a tube is found to have degradation in excess of the plugging limits (i.e., 40 percent through-wall), it is required to be repaired or removed from service.

Steam generator tubes comprise a significant portion of the reactor coolant pressure boundary. Maintenance of this boundary is provided by the integrity of the steam generator tube wall and tube-to-tubesheet joint. The joint between the tube and tubesheet is an interference fit constructed by roll expanding the tube into the bore of the tubesheet. The tubes are inserted into the bore of the tubesheet followed by seal welding at the primary face of the tubesheet. Step rolls are then performed on the tube to fully expand the tube in the bore of the tubesheet. The tubes are restrained in the bore of the tubesheet by the elastic springback of the tubesheet. The tube-to-tubesheet joint provides sufficient strength to maintain adequate structural and pressure boundary integrity.

The industry experience has shown that defects have developed in the tube-to-tubesheet joints by various degradation processes. The staff believes that some tubes having degradation in the joint may remain in service because the reinforcing effect of the tubesheet on the undegraded portion of the tube would maintain structural and leakage integrity. However, an undegraded portion of the tube must be available to maintain adequate structural and leakage integrity under loadings from normal operation, anticipated operational occurrences, and postulated accident conditions. RG 1.121 recommends that the margin of safety against tube rupture under normal operating conditions should not be less than three for any tube location where defects have been detected. For postulated accidents, RG 1.121 recommends that the margin of safety against tube rupture be consistent with the margin of safety determined by the stress limits specified in NB-3225 of Section III of the Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers (ASME).

The staff has previously approved two alternative tube repair criteria, F^* and L^*, at other PWR facilities to allow tubes with degradation within the tubesheet to remain in service provided the degradation within the tubes has been shown to meet certain criteria. The F^* criteria allow degraded tubes to remain in service when defects exist at a specified distance below the bottom of the roll transition. The L^* criteria permit certain defects to exist within the F^* distance provided additional conditions are satisfied.

Structural loads imposed on the tube-to-tubesheet joint primarily result from the differential pressure between the primary and secondary sides of the tubes. The peak postulated loading occurs during a main steamline break due to a lowering of the secondary side pressure. However, normal operating loads, cyclic loading from transients (e.g., startup/shutdown), and potential thermal expansion loads can also be significant. The analysis supporting these alternative criteria must address all loading conditions necessary to maintain adequate integrity of the tube-to-tubesheet joint.

The elastic preload between the tube and tubesheet not only prevents pullout of the tube from the tubesheet, but also provides a leak tight barrier minimizing the potential for primary to secondary coolant leakage. With sufficient length of undegraded hardroll, the tube-to-tubesheet joint will not allow any leakage under normal and faulted conditions. Under the proposed L^*-type criteria, the licensee has to demonstrate that leakage integrity of the joint is maintained under all analyzed conditions.

3.2 Structural Integrity of the L*-Type Criteria

The licensee developed the L*-type criteria for the 7/8-inch diameter tubes in the Farley steam generators using a combination of analyses and adjustment of test data from an L* program that was developed for 3/4-inch diameter tubes. The licensee considered various loads that would affect the structural integrity of the degraded tubes. The primary load considered was the maximum pullout loads under the normal operation and faulted (feed line break) conditions. The licensee applied a safety factor of 3 as recommended by RG 1.121 to the axial load under the normal operating conditions. A safety factor of 1.43 was applied to the axial load for the feed line break case. The licensee also considered axial compression loads caused by large break loss-of-coolant accident and bending and torsion loads about the tube vertical axis. The licensee stated that significant margin exists between tube collapse strength and the limiting secondary-to-primary pressure differential. The licensee used appropriate safety factors recommended by RG 1.121 in determining the pullout load of the degraded tubes.

In addition, the F* criteria approved for Farley Unit 2 was developed on the assumption that the portion of the tube below the F* distance does not contribute to the tube-to-tubesheet joint integrity. The L*-type criteria allow tubes with a small number of short predominantly axial cracks to remain in service provided a minimum aggregate of approximately 2 inches of sound roll existed below the bottom of the roll transition. This limited degradation would be expected to have an insignificant effect on the structural integrity of the joint within the inspected region. Based on this criteria in comparison with the staff's detailed review and evaluation of the structural integrity tests performed for the F* criteria, the staff concludes that the L*-type criteria for Farley Unit 2 provide adequate assurance of structural integrity.

3.3 Leakage Integrity of the L*-Type Criteria

The licensee stated that the L* length of the sound roll expansion is sufficient to preclude significant leakage from tube degradation located below the L* distance. The existing TS leak rate requirements and accident analysis assumptions remain unchanged in the unlikely event that significant leakage from the L* tubes does occur. Any leakage from the tube within the tubesheet at any elevation in the tubesheet is fully bounded by the existing steam generator tube rupture analysis included in the Farley Nuclear Plant Final Safety Analysis Report. A conservative leakage allowance for each L* tube is provided to determine the impact of the L*-type criteria on offsite radiological doses in the event of a postulated double ended guillotine break of the main steamline outside of containment. The licensee determined that the primary-to-secondary leakage will result in offsite doses that are a small fraction of the 10 CFR Part 100 guidelines. The licensee stated that the proposed L* criteria do not adversely impact any other previously evaluated design basis accident.

The staff has reviewed F^* leakage testing data from steam generator tubes with undegraded hardroll which indicated no leakage occurred. Similar lengths of undegraded hardroll exists in the six tubes that the proposed L^*-type criteria are being applied to. Based on this comparison, for the degradation in the six tubes covered by this amendment, any leakage resulting from the use of proposed L^*-type criteria is acceptable.

The staff concludes that the proposed L^*-type criteria are acceptable for Farley Unit 2 for the limited number of tubes affected based on the adequacy of the structural and leakage integrity provided by the L^*-type criteria. Therefore the staff has determined that the licensee's proposed TS change that incorporates the L^*-type criteria into the Farley Unit 2 TS for the duration of the present operating cycle is acceptable.

4.0 EXIGENT CIRCUMSTANCES

The Commission's regulations, 10 CFR 50.91(a)(6), contain provisions for issuance of amendments when the usual 30-day public notice period cannot be met. One type of special exception is an exigency. Under such circumstances, the Commission notifies the public in one of two ways: by issuing a Federal Register notice providing an opportunity for hearing and allowing at least 2 weeks for prior public comments, or by issuing a press release discussing the proposed changes, using local media. In this case, the Commission used the first approach.

The licensee, on April 22, 1996, identified six tubes that were left in service which did not satisfy the F^* distance requirements of TS 4.4.6.4.11. Non-compliance with this TS required the licensee to declare the steam generators inoperable and to take action within one hour to be in hot standby within the next 6 hours in accordance with TS 3.0.3. The licensee promptly notified the NRC and requested that the NRC exercise its enforcement discretion. The licensee proposed a revision to the TS, as stated in its April 23, 1996, submittal, which, if granted, would bring the plant into compliance with its TS for the remainder of the operating cycle. Based on the information provided by the licensee, the staff concluded that continued operation was acceptable and granted a Notice of Enforcement Discretion (NOED) to avoid an undesirable transient as a result of forcing compliance with the TS. The NOED was granted until the prompt issuance of the proposed license amendment.

Accordingly, pursuant to 10 CFR 50.91(a)(6), the Commission has determined that an exigent situation exists in which it must act before the expiration of the 30-day comment period to bring the plant into compliance with its TSs.

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92(c) state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility in accordance with the amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

Operation of the facility in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. As discussed in Section 3.0 of this SE, the L^* criterion provides adequate assurance of structural integrity to prevent tube pullout or collapse. In addition, the licensee assumed conservative leakage allowances for each L^* tube to assure that the primary-to-secondary leakage would not result in offsite doses that exceed 10 CFR Part 100 guidelines during a postulated double ended guillotine break of the main steamline outside of containment.

Operation of the facility in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. The implementation of the L^* criterion does not provide a mechanism that would result in an accident initiated outside of the tubesheet. Verification of the L^* criterion prevents tube displacement of any magnitude and postulated axial cracks existing a minimum of 0.5 inch from either the bottom of the roll transition or top of the tube sheet, whichever is lower, from migrating out of the tubesheet.

Operation of the facility in accordance with the amendment will not involve a significant reduction in a margin of safety. The implementation of the L^* criterion maintains the integrity of the tube bundle commensurate with the guidance in draft Regulatory Guide 1.121 under normal and accident conditions. In addition, the L^* length has been verified by testing that sufficient roll expansion length is available to preclude significant leakage and to provide structural integrity during normal and postulated accident conditions.

Based upon the above considerations, the staff concludes that the amendment meets the three criteria of 10 CFR 50.92. Therefore, the staff has made a final determination that the proposed amendment does not involve a significant hazards consideration.

6.0 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.

7.0 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 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 made a final finding that the amendment involves no significant hazards consideration. 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 this amendment.

8.0 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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Date: May 20, 1996