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 AMENDMENTS - JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. M97699 and M97700)

Dear Mr. Morey:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 125 to Facility Operating License No. NPF-2 and Amendment No. 119 to Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant, Units 1 and 2. The amendments change the Technical Specifications (TS) in response to your submittal dated January 10, 1997, as supplemented by letter dated February 24, 1997.

The amendments revise the TS to incorporate the latest revised topical reports governing the installation of laser welded steam generator tube sleeves. In addition, an administrative change for Unit 2 revises the TS by deleting the reference to a one-cycle implementation of L*, which expired at the last Unit 2 outage.

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 Nos. 50-348 and 50-364

DFol

Enclosures:
1. Amendment No. 125 to NPF-2
2. Amendment No. 119 to NPF-8
3. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION:

Docket File	PUBLIC
PDII-2 RF	S.Varga
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9PIH 3-10-97

OFFICE	PDII-2/PM	PDII-2/LA	EMCB	OGC	PD22/D
NAME	J.ZIMMERMAN:cn	L.BERRY	TSULLIVAN	M Young	H.BERKOW
DATE	3/10/97	3/6/97	3/10/97	3/20/97	3/24/97
COPY	YES NO	YES NO	YES NO	YES NO	YES NO

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March 24, 1997

Mr. D. N. Morey
Vice President - Farley Project
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Post Office Box 1295
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The amendments revise the TS to incorporate the latest revised topical reports governing the installation of laser welded steam generator tube sleeves. In addition, an administrative change for Unit 2 revises the TS by deleting the reference to a one-cycle implementation of L*, which expired at the last Unit 2 outage.

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OFFICE	PDII-2/PM	PDII-2/LA	EMCB	OGC	PD22/D
NAME	J.ZIMMERMAN:cn	L.BERRY	TSULLIVAN	U Young	H.BERKOW
DATE	3/10/97	3/6/97	3/10/97	3/20/97	3/24/97
COPY	YES NO	YES NO	YES NO	YES NO	YES NO

OFFICIAL RECORD COPY

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

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 AMENDMENTS - JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. M97699 and M97700)

Dear Mr. Morey:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 125 to Facility Operating License No. NPF-2 and Amendment No. 119 to Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant, Units 1 and 2. The amendments change the Technical Specifications (TS) in response to your submittal dated January 10, 1997, as supplemented by letter dated February 24, 1997.

The amendments revise the TS to incorporate the latest revised topical reports governing the installation of laser welded steam generator tube sleeves. In addition, an administrative change for Unit 2 revises the TS by deleting the reference to a one-cycle implementation of L*, which expired at the last Unit 2 outage.

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,

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

Docket Nos. 50-348 and 50-364

Enclosures:
1. Amendment No. 125 to NPF-2
2. Amendment No. 119 to NPF-8
3. Safety Evaluation

cc w/encls: See next page

cc:
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Mr. Mark Ajluni, Licensing Manager
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State Health Officer
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434 Monroe Street
Montgomery, Alabama 36130-1701

Chairman
Houston County Commission
Post Office Box 6406
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Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
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Atlanta, Georgia 30323

Resident Inspector
U.S. Nuclear Regulatory Commission
7388 N. State Highway 95
Columbia, Alabama 36319

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. 125
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 January 10, 1997, as supplemented by letter dated February 24, 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-2 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 125 , 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

Kahtan 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. 125

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 4-9	3/4 4-9
3/4 4-10	3/4 4-10
3/4 4-11	3/4 4-11
3/4 4-12	3/4 4-12
3/4 4-12a	3/4 4-12a
3/4 4-13	3/4 4-13
--	3/4 4-15a
B 3/4 4-3a	B 3/4 4-3a

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 T_{avg} 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 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 Tables 4.4-2 and 4.4-3. 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. When applying the exceptions of 4.4.6.2.a through 4.4.6.2.c, previous defects or imperfections in the area repaired by sleeving are not considered an area requiring reinspection. 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:

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.

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. 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.

4. 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 Tables 4.4-2 and 4.4-3) 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:

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.

4.4.6.3 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 an steam generator conducted in accordance with Tables 4.4-2 and 4.4-3 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 Tables 4.4-2 and 4.4-3 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.7.2.
   2. A seismic occurrence greater than the Operating Basis Earthquake.
   3. A loss-of-coolant accident requiring actuation of the engineered safeguards.
   4. A main steam line or feedwater line break.

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 24% 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 reports WCAP-130t Revision 4, and WCAP-14740 dated January 1997, 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.

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 Tables 4.4-2 and 4.4-3.

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.

TABLE 4.4-3

STEAM GENERATOR REPAIRED TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION	
Sample Size	Result	Actions Required	Result	Action Required
A minimum of 20% of repaired tubes (1)(2)	C-1	None	NA	NA
	C-2	Plug or repair defective repaired tubes and inspect 100% of the repaired tubes in this steam generator.	C-1	None
			C-2	Plug or repair defective repaired tubes.
			C-3	Perform action for C-3 result of first sample.
	C-3	Inspect all repaired tubes in this steam generator, plug or repair defective tubes and inspect 20% of the repaired tubes in each steam generator Notification to NRC pursuant to 10 CFR 50.72(b)(2).	All other steam generators are C-1.	None
			Some steam generators C-2 but no additional steam generators are C-3.	Perform action for C-2 result of first sample.
			Additional steam generator is C-3.	Inspect all repaired tubes in each steam generator and plug or repair defective tubes. Notification to NRC pursuant to 10 CFR 50.72(b)(2).

(1) Each repair method is considered a separate population for determination of scope expansion.
(2) The inspection of repaired tubes may be performed on tubes from 1 to 3 steam generators based on outage plans.

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 24% for the laser welded sleeve of sleeve nominal wall thickness in the sleeve, it must be plugged. The 31% and 24% 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:

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. 119
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 January 10, 1997, as supplemented by letter dated February 24, 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. 119, 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

Kahtan 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. 119

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	Insert
3/4 4-9	3/4 4-9
3/4 4-10	3/4 4-10
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
3/4 4-13b	3/4 4-13b
---	3/4 4-15a
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	B 3/4 4-3b
B 3/4 4-3c	---

REACTOR COOLANT SYSTEM

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 Tables 4.4-2 and 4.4-3. 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* 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.

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

3. 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.

4. 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 Tables 4.4-2 and 4.4-3) 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:

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* Tube Inspection - In addition to the minimum sample size as determined by Specification 4.4.6.2.1, all F* tubes will be inspected within the tubesheet region. The results of this inspection will not be a cause for additional inspections per Tables 4.4-2 and 4.4-3.

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 Tables 4.4-2 and 4.4-3 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 Tables 4.4-2 and 4.4-3 during the shutdown subsequent to any of the following conditions:

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* 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 24% 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.16 for the repair limit applicable to these intersections. For a tube with an imperfection or flaw in the tubesheet below the lower joint of an installed elevated laser welded sleeve, no repair or plugging is required provided the installed sleeve meets all sleeved tube inspection requirements.

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 with a tube sheet sleeve installed, the point of entry is the bottom of the tube sheet sleeve below the lower sleeve joint. 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 reports WCAP-13088, Revision 4, and WCAP-14740 dated January 1997, 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 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 1.54 inches plus allowance for eddy current uncertainty measurement and is measured down from the top of the tube sheet or the bottom of the roll transition, whichever is lower in elevation. The allowance for eddy current uncertainty is documented in the steam generator eddy current inspection procedure.

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.

13. 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. Tube expansion also refers to that portion of a sleeve which has been increased in diameter by a rolling process such that no crevice exists between the outside diameter of the sleeve and the parent steam generator tube.

14. Tube Support Plate Repair Limit is used for the disposition of an alloy 600 steam generator tube for continued service that is experiencing predominately 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 voltages 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.14.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.14.a, 4.4.6.4.a.14.b, and 4.4.6.4.a.14.c. The mid-cycle repair limits are determined from the following equations:

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

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

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

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.14.a, 4.4.6.4.a.14.b, and 4.4.6.4.a.14.c.

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 Tables 4.4-2 and 4.4-3.

4.4.6.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged, repaired or designated F* 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 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.

TABLE 4.4-3

STEAM GENERATOR REPAIRED TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION	
Sample Size	Result	Actions Required	Result	Action Required
A minimum of 20% of repaired tubes (1)(2)	C-1	None	NA	NA
	C-2	Plug or repair defective repaired tubes and inspect 100% of the repaired tubes in this steam generator.	C-1	None
			C-2	Plug or repair defective repaired tubes.
			C-3	Perform action for C-3 result of first sample.
	C-3	Inspect all repaired tubes in this steam generator, plug or repair defective tubes and inspect 20% of the repaired tubes in each steam generator Notification to NRC pursuant to 10 CFR 50.72(b)(2).	All other steam generators are C-1.	None
			Some steam generators C-2 but no additional steam generators are C-3.	Perform action for C-2 result of first sample.
			Additional steam generator is C-3.	Inspect all repaired tubes in each steam generator and plug or repair defective tubes. Notification to NRC pursuant to 10 CFR 50.72(b)(2).

(1) Each repair method is considered a separate population for determination of scope expansion.
(2) The inspection of repaired tubes may be performed on tubes from 1 to 3 steam generators based on outage plans.

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 = 150 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 2 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 4.4.6.4.a.14 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 4.4.6.4.a.14 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 bound 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 are discussed in GL 95-05.

The mid-cycle equation in 4.4.6.4.a.14.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 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 24% for the laser welded sleeve of sleeve nominal wall thickness in the sleeve, it must be plugged. The 31% and 24% 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.

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 1.54 inches plus allowance for eddy current uncertainty measurement and is measured down from the top of the tubesheet or the bottom of the roll transition, whichever is lower in elevation.

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. 125 TO FACILITY OPERATING LICENSE NO. NPF-2

AND AMENDMENT NO. 119 TO FACILITY OPERATING LICENSE NO. NPF-8

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-348 AND 50-364

1.0 INTRODUCTION

By letter dated January 10, 1997, as supplemented by letter dated February 24, 1997, the Southern Nuclear Operating Company, Inc., et al. (the licensee), submitted a request for changes to the Joseph M. Farley Nuclear Plant, Units 1 and 2, Technical Specifications (TS). The requested changes would incorporate the latest revised topical reports governing the installation of laser welded steam generator tube sleeves designed by Westinghouse (W). The licensee is presently licensed to use Westinghouse sleeves for repairing steam generator tubes as an alternative to plugging. The proposed changes update the existing Farley plant Technical Specifications to incorporate the latest advances in Westinghouse sleeve installation technology and add certain current industry recommended practices for sleeved tube reinspections. In addition, the reference to a one-cycle implementation of L*, which expired at the last Unit 2 outage would be deleted from the Unit 2 TSs. The February 24, 1997, letter provided clarifying information that did not change the original application and the initial proposed no significant hazards consideration determination published in the Federal Register on January 29, 1997 (61 FR 4355).

The revised topical reports detail the latest design and installation features of three types of steam generator tube sleeves: a full depth tubesheet sleeve, an elevated tubesheet sleeve, and a tube support plate sleeve. The updated design for Westinghouse sleeves is to laser weld and heat treat the freespan joint(s) and mechanically roll the tubesheet joints. As previously licensed, the Westinghouse sleeves for use at the Farley plant specified a tubesheet joint incorporating both a rolled and welded joint, and a freespan welded joint with an optional heat treatment.

Additionally, due to certain construction details applicable to the Farley units, the proposed change also modifies the normal rolled joint to incorporate a double roll.

The proposed changes are based upon the experiences gained and improvements incorporated by Westinghouse during the last 2 years during large sleeving campaigns at other facilities.

Extensive analyses and testing were performed on the design and process modifications to demonstrate that Regulatory and Code design criteria for sleeved tubes were satisfied under normal operating and postulated accident conditions. The details of the sleeve qualifications for Farley are discussed in Westinghouse reports WCAP-13088, "Westinghouse Series 44 and 51 Steam Generator Generic Sleeving Report, Laser Welded Sleeves" (proprietary), dated January 1997, and WCAP-14740, "Specific Application of Laser Welded Sleeves for the Farley Units 1 and 2 Steam Generators" (proprietary), dated January 1997.

The staff has previously reviewed similar W documents supporting TS amendments for sleeve installations at other plants. The bulk of the technical and regulatory issues for the present request are identical to those reviewed in previous Safety Evaluations (SEs) concerning the use of W laser welded sleeves. This SE summarizes the principal issues discussed in previous reviews and adds a discussion of those warranting revision, amplification, or inclusion based upon current experience. Details of the prior staff evaluation of W sleeves may be found in SEs for Calvert Cliffs Nuclear Power Plant Units 1 and 2, Docket Nos. 50-317 and 50-318, dated March 22, 1996; DC Cook Nuclear Power Plant Unit 1, Docket No. 50-315, dated January 4, 1996; and Maine Yankee Nuclear Power Plant, Docket No. 50-309, dated May 22, 1995. Additionally, prior evaluations of Westinghouse sleeves have been performed for the Farley plant. The relevant TS amendments were dated September 18, 1987, October 22, 1990, and November 20, 1996.

2.0 SUMMARY OF PREVIOUS REVIEWS

Previous staff evaluations of W sleeves addressed the technical adequacy of the sleeves in the principal areas of pressure retaining component design: structural requirements, material of construction, welding and post weld heat treatment effects, and nondestructive examination. Along with these design evaluations, the staff has included evaluations of sleeve design changes based upon operating experiences with previous sleeving installations. The staff position and findings regarding sleeving methods are summarized below:

2.1 Structural Requirements

The function of sleeves is to restore the structural integrity of the tube pressure boundary. Consequently, structural analyses were performed for a variety of loadings including design pressure, operating transients, and other parameters selected to envelop loads imposed during normal operating, upset, and accident conditions. Stress analyses of sleeved tube assemblies were performed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III. These analyses, along with the results of qualification testing and previous plant operating experience, were cited to demonstrate the sleeved tube assembly is capable of restoring steam generator tube structural integrity.

2.2 Material of Construction

The sleeves are fabricated from thermally treated alloy 690, a Code-approved material (ASME SB-163) covered by ASME Code Case N-20. The staff found the use of alloy 690 is an improvement over the alloy 600 material used in the original SG tubing. Corrosion tests conducted under Electric Power Research Institute (EPRI) sponsorship confirmed test results regarding the improved corrosion resistance of alloy 690 over that of alloy 600. Accelerated stress corrosion tests in caustic and aqueous chloride solutions also indicated alloy 690 resists general corrosion in aggressive environments. Isothermal tests in high purity water have shown that, at normal stress levels, alloy 690 has high resistance to intergranular stress corrosion cracking (IGSCC) in extended high temperature exposure. The NRC staff concluded, as a result of these laboratory corrosion tests, that alloy 690 is acceptable for use in nuclear power plants. The NRC endorsed the use of Code Case N-20 in Regulatory Guide 1.85, "Materials Code Case Acceptability, ASME Section III, Division 1." The NRC staff has approved the use of alloy 690 tubing in replacement steam generators as well as sleeving applications.

2.3 Welding and Post Weld Heat Treatment

Automatic autogenous laser welding is employed to join the sleeve to the parent tube in the free span regions. The application of this process to the W sleeve design was specifically qualified and demonstrated during laboratory tests employing full scale sleeve/tube mockups. Qualification of the welding procedures and welding equipment operators was performed in accordance with the requirements of the ASME Code, Section IX.

Accelerated corrosion tests have confirmed that a post weld heat treatment (PWHT) significantly improves the IGSCC resistance of the alloy 600 parent tube material in the weld zone. A PWHT reduces the residual stresses resulting from welding. Residual stresses from forming operations (such as bending, welding, etc.) are known to be a principal contributor to IGSCC in alloy 600. Performance of a PWHT greatly reduces the residual stresses from welding thereby enhancing the IGSCC resistance of the alloy 600 portion of the weld zone. The alloy 690 sleeve material is highly resistant to IGSCC either with or without PWHT. All free span laser welded joints will be heat treated in accordance with the W generic sleeving report (WCAP-13088) and the NRR staff conclusion that PWHT enhances weld joint resistance to IGSCC.

The rolled joint used to join the sleeve to the tube within the tubesheet effectively isolates the alloy 600 of the parent tube from the environment and, thus, is not susceptible to IGSCC. Stress relief of these joints is unwarranted. PWHT of lower joint seal welds (where used) is undesirable due to potentially deleterious effects upon the tubesheet material and the integrity of the rolled joint.

2.4 PWHT and Tube Lockup

Field experience with the installation of welded sleeves with PWHT indicated that steam generator tubes may be constrained in their tube supports ("tube lockup"). The principal result of such tube locking is a residual stress ("far-field" stress) remaining after the heat treatment is completed. Full scale laboratory mock-ups of sleeved tube assemblies allowed study of the contributing factors to the far-field stress. Based upon these laboratory tests, the sleeve installation procedure was modified to minimize the far-field stress. Strain gage measurements of the remaining residual stress have shown it to be moderate compared to the stress resulting from welding without subsequent PWHT.

The effectiveness of the PWHT with modified installation procedure for minimizing the far-field stress has been further verified with accelerated corrosion tests. W has developed a proprietary accelerated corrosion test that is used to rank the relative IGSCC resistance of tubes, welds or other components. Welded tube/sleeve samples with and without PWHT and subjected to operating pressure plus far-field stresses have been tested for susceptibility to IGSCC. These test results have consistently and clearly demonstrated the superiority of the specimens with PWHT over all others.

2.5 Service Life Predictions for Sleeved Steam Generator Tubes

The staff position on sleeving considers the method unable to assure an unlimited service life for a repaired tube. The conservative view is that sleeving creates new locations in the parent tube, which may be susceptible to IGSCC after new incubation times are expended. Incubation times are not quantified. They are observed to vary between individual steam generators and the various tubes within, based upon prior experiences with U-bend and roll transition cracking.

This staff position that sleeving has limited service life is based upon the circumstances of the sleeving processes. Sleeve installation methods can enhance one or two of the conditions necessary for IGSCC. The primary contributor is the residual stress resulting from the various joining methods. In addition, the local environment of the tube may be altered as a result of the formation of a wetted crevice between the tube and sleeve. Remediation of these contributors would benefit sleeved tube life. Of the two, stress relieving may be the most beneficial given the underlying causes of IGSCC and present sleeve designs. As discussed earlier, the sleeve installation procedure includes a PWHT of the weld joints to increase the resistance to IGSCC.

2.6 Nondestructive Examination

The sleeve assemblies can be inspected by nondestructive techniques in accordance with the recommendations of Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes."

Nondestructive examination of sleeved tubes is conducted in two primary ways. Ultrasonic testing (UT) is performed after welding to confirm the laser welds are consistent with critical process dimensions and are of acceptable weld quality. W presented data on a UT system demonstrating that post weld examinations of the sleeve/tube assembly will be adequate. Standards that included undersized welds were used in the qualification of the UT technique. The results of the qualification tests demonstrate the system can confirm that there is a continuous metallurgical bond between the sleeve and tube and the weld size (width) meets minimum acceptable dimensions.

Eddy current testing (ECT) is then used to establish baseline inspection data for every installed sleeve/tube. This data is compared with subsequent ECT inspections to aid in identifying any possible service induced degradation, should it occur. In performing these inspections, the licensee will use Electric Power Research Institute (EPRI) "PWR Steam Generator Tube Examination Guidelines" Appendix G qualified personnel and Appendix H qualified ECT techniques. For future sleeve/tube inspections, the licensee committed to following the most current revision of the EPRI guidelines in terms of inspection scope and expansion criteria as reflected in adoption of TS Table 4.4-3, "Steam Generator Repaired Tube Inspection."

3.0 DISCUSSION

The previous section addressed generic topics applicable to steam generator tube sleeve installations using W laser welded sleeves. For the Farley amendment request, plant specific modifications were also proposed.

3.1 Modified Sleeve Rolling Procedure for Tubesheet Joints

Due to certain original construction details peculiar to Farley Unit 1, the licensee requested a modification of the standard W lower rolled joint for use at the Farley facility. The construction difference between the two units' steam generators involves the WEXTEX expansion method that was employed on Unit 1. Unit 2's steam generators were constructed with full depth rolling. In the interests of commonality, the licensee sought a lower joint rolling method that would be applicable to both units. Consequently, the licensee engaged Westinghouse to develop a modified rolling procedure (called a two-roll pass lower joint) for use at Farley.

Since the modified rolling procedure was a departure from that previously approved for other installations, a new series of qualification tests were performed as detailed in WCAP-14740. The principal tests concern measured leak rate (if any) and structural integrity for all design conditions. Mock-ups of the modified rolled joint were produced and laboratory tested for conformance with the requirements for leak rate and structural capability.

Leak test specimens subjected to a range of pressures (reflecting primary-to-secondary pressure differentials) showed no test samples with leak rates beyond a tiny fraction of a drop per minute. Using the average measured leak rate for all the tested samples and assuming a worst case accident condition (steam line break) with 1000 sleeves per generator would yield a resulting leak rate measurable in the drops per minute range. This is an insignificant fraction of the allowable primary-to-secondary accident-induced leakage limit. For Farley, this limit is 11.4 gpm. (An accident-induced primary-to-secondary leakage limit is determined to ensure that the 10 CFR Part 100 and General Design Criterion 19 radiation limits are met). Therefore, it can be concluded that primary-to-secondary leakage under these worst case conditions would be insignificant or zero for plant normal and postulated faulted event pressure conditions for the two-step rolled joint configuration.

The mechanical strength of the two-step rolled joint was tested by loading mock-up joints to failure and noting the load (pull-out test). In every case, the samples had pull-out strengths that easily exceeded the most stringent requirement of Regulatory Guide 1.121 (which specifies a minimum load capability of 3 times the normal operating value).

The staff finds that the leakage and structural capability tests are consistent with previous tests of rolled joints and that they meet all applicable design and regulatory requirements.

3.2 Deletion of Seal Weld from Tubesheet Joint

The existing Farley TS section for laser welded sleeves specifies a lower rolled joint, which included a seal weld. As demonstrated through the tests discussed in WCAP-13088, a rolled joint is adequately leak limiting and, thus, use of a seal weld in conjunction with the lower rolled joint is now regarded as unnecessary.

The seal weld practice originated from the desire to preclude any leakage through the lower rolled joint. However, numerous tests performed by Westinghouse have consistently demonstrated that any rolled joint leakage, should it occur, is a minuscule fraction of the 10 CFR Part 100 limits. Additionally, extensive operating experience with thousands of sleeves, installed only with rolled joints, has demonstrated actual performance to be essentially leak tight.

3.3 Sleeve Plugging Limits

The sleeve minimum acceptable wall thickness is determined using the criteria of Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes" and ASME Code Section III allowable stress values and pressure stress equations. According to Regulatory Guide 1.121 criteria, an allowance for nondestructive evaluation (NDE) uncertainty and postulated operational growth of tube wall degradation within the sleeve must be accounted for when using NDE to determine sleeve plugging limits. Therefore, a conservative tube wall combined allowance for postulated degradation growth and eddy current uncertainty of 20% through-wall per cycle was assumed for the purpose of determining the sleeve plugging limit. The sleeve plugging limit, which was calculated based on the most limiting of normal, upset, or faulted conditions for welded sleeves installed at the Farley plant, was determined to be 24% of the sleeve nominal wall thickness based on ASME Code minimum material properties in accordance with staff positions. This plugging limit of 24% provides assurance that pressure boundary integrity will be maintained.

3.4 Technical Specification Changes

The staff finds acceptable the following proposed technical and editorial changes to the plant TS SR 4.4.6.4 and associated bases.

1. Table 4.4-3, "Steam Generator Repair Tube Inspection" has been incorporated. This TS change increases the sample size requirements from 3% to 20%, which will ensure adequate detection of additional tube degradation. In addition, TS SRs have been modified to incorporate a reference to Table 4.4-3.

2. The definition of tube repair has been modified to indicate that laser welded sleeves will be installed as described in Westinghouse reports WCAP-13088, Revision 4 and WCAP-14740 (January 1997).

3. The definition of plugging or repair limit and the associated bases has been modified to incorporate the plugging limit of 24%. This change provides assurance that pressure boundary integrity will be maintained.

4. The reference to a one-cycle implementation of L*, which expired at the last Unit 2 outage has been deleted from Unit 2 TS 4.4.6.4 and associated bases. In addition, the definition of L* and associated bases have been deleted from the Unit 2 TSs.

The staff finds these TS changes consistent with the laser welded sleeve installation.

4.0 STAFF CONCLUSIONS

The staff concludes the proposed W laser welded sleeves, as described in the sleeve topical reports WCAP-13088, Revision 4, and WCAP-14740, will provide sleeved tubes of acceptable metallurgical properties, structural integrity, leak tightness, and corrosion resistance. The staff also finds the preservice and inservice inspection methods, minimum percentages, and expansion criteria for examining the welds and sleeved tubes are acceptable.

5.0 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.

6.0 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 (January 29, 1997, 62 FR 4355). The amendments also revise recordkeeping and reporting requirements. Accordingly, the amendments meet 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 amendments.

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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

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