April 7, 1995

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

SUBJECT: ISSUANCE OF AMENDMENT NO. 106 TO FACILITY OPERATING LICENSE NO. NPF-8 REGARDING VOLTAGE-BASED STEAM GENERATOR TUBE REPAIR CRITERIA - JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2 (TAC NO. M91049)

Dear Mr. Morey:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. to Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant, Unit 2. The amendment changes the Technical Specifications (TSs) in response to your submittal dated December 7, 1994, as supplemented February 14 and March 20, 1995. The December 7, 1994, submittal requested a permanent change to the TSs for both units related to steam generator tube support plate voltage-based repair criteria in accordance with the draft Generic Letter on this issue. Because the staff is not prepared to grant permanent TSs on this subject until the draft Generic Letter has been issued, your two supplements to the original submittal provided plant-specific information to support one additional cycle of operation for Unit 2. You also stated that the no significant hazards consideration determination originally submitted remains valid for Unit 2 for the one operating cycle.

The original submittal also included TS changes for Unit 1 (TAC No. M91048). This TAC will remain open until the pending plant-specific supplemental information is submitted later this year.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's bi-weekly Federal Register 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

Docket No. 50-364

Enclosures:
1. Amendment No. 106 to NPF-8
2. Safety Evaluation

cc w/enclosures:
See next page

DOCUMENT NAME: G:\FARLEY\FA91049.AMD

***SEE PREVIOUS CONCURRENCE**

OFFICE	LA:PDII-2	PM:PDII-2	D:PDII-2	EMCB*	OGC*
NAME	LBerry	BSiegel	HBerkow	JStrosnider	
DATE	04/6/95	04/4/95	04/4/95	03/31/95	04/05/95
COPY	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No

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

April 7, 1995

Mr. D. N. Morey, Vice President
Southern Nuclear Operating Co., Inc.
Post Office Box 1295
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SUBJECT: ISSUANCE OF AMENDMENT NO. 106 TO FACILITY OPERATING LICENSE NO. NPF-8 REGARDING VOLTAGE-BASED STEAM GENERATOR TUBE REPAIR CRITERIA - JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2 (TAC NO. M91049)

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Docket No. 50-364

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1. Amendment No. 106 to NPF-8
2. Safety Evaluation

cc w/enclosures: See next page

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AMENDMENT NO. 106 TO FACILITY OPERATING LICENSE NO. NPF-8 - FARLEY, UNIT 2

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

DOCKET NO. 50-364

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 106
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 December 7, 1994, as supplemented February 14, 1995, and March 20, 1995, 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. 106, 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

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: April 7, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 106

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 4-10	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 4-12b
3/4 4-13	3/4 4-13
3/4 4-13a	3/4 4-13a
3/4 4-17	3/4 4-17
3/4 4-23	3/4 4-23
3/4 4-24	3/4 4-24
3/4 4-25	3/4 4-25
3/4 4-26	3/4 4-26
B 3/4 4-3	B 3/4 4-3
B 3/4 4-4	B 3/4 4-4
B 3/4 4-5	B 3/4 4-5

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

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

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

5. Tube support plate indications left in service as a result of application of the tube support plate plugging criteria shall be inspected by bobbin coil probe during the following refueling outages.

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

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

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

d. Implementation of the steam generator tube/tube support plate plugging criteria requires 100 percent bobbin coil inspection for hot-leg tube support plate intersections and cold-leg intersections down to the lowest cold-leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of 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 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:

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. <u>Plugging or Repair Limit</u> 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 to the area of the tubesheet region below the F* distance in the F* tubes. 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. <u>Unserviceable</u> 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. <u>Tube Inspection</u> 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. <u>Tube Repair</u> refers to mechanical sleeving, as described by Westinghouse report WCAP-11178, Rev. 1, or laser welded sleeving as described by Westinghouse report WCAP-12672, which is used to maintain a tube in service or return a tube to service. This includes the removal of plugs that were installed as a corrective or preventive measure.

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

11. 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.

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.

14. 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 Operating 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 a 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* 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.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.7.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. Primary-to-secondary leakage through all steam generators shall be limited to 450 gallons per day and 150 gallons per day through any one steam generator.

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 31 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig.

f. The maximum allowable leakage of any Reactor Coolant System Pressure Isolation Valve shall be as specified in Table 3.4-1 at a pressure of 2235 ± 20 psig.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the limit specified in Table 3.4-1, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3/4.4.9 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.9 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 0.5 microCurie per gram DOSE EQUIVALENT I-131;

b. Less than or equal to $100/\bar{E}$ microCurie per gram.

APPLICABILITY: MODES 1, 2, 3, 4 and 5

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the primary coolant greater than 0.5 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

b. With the specific activity of the primary coolant greater than $100/\bar{E}$ microCurie per gram, be in at least HOT STANDBY with Tavg less than 500°F within 6 hours.

* With T_{avg} greater than or equal to 500°F.

ACTION: (Continued)

MODES 1, 2, 3, 4 and 5:

a. With the specific activity of the primary coolant greater than 0.5 microCurie per gram DOSE EQUIVALENT I-131 or greater than $100/\overline{E}$ microCuries per gram, perform the sampling and analysis requirements of item 4a of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

4.4.9 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Activity Determination	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Radiochemical for $\bar{E}$ Determination	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, 1-133, and 1-135	a) Once per 4 hours, whenever the specific activity exceeds 0.5 µCi/gram DOSE EQUIVALENT 1-131 or $100/\bar{E}$ µCi/gram, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.	1, 2, 3

Until the specific activity of the primary coolant system is restored within its limits.
* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

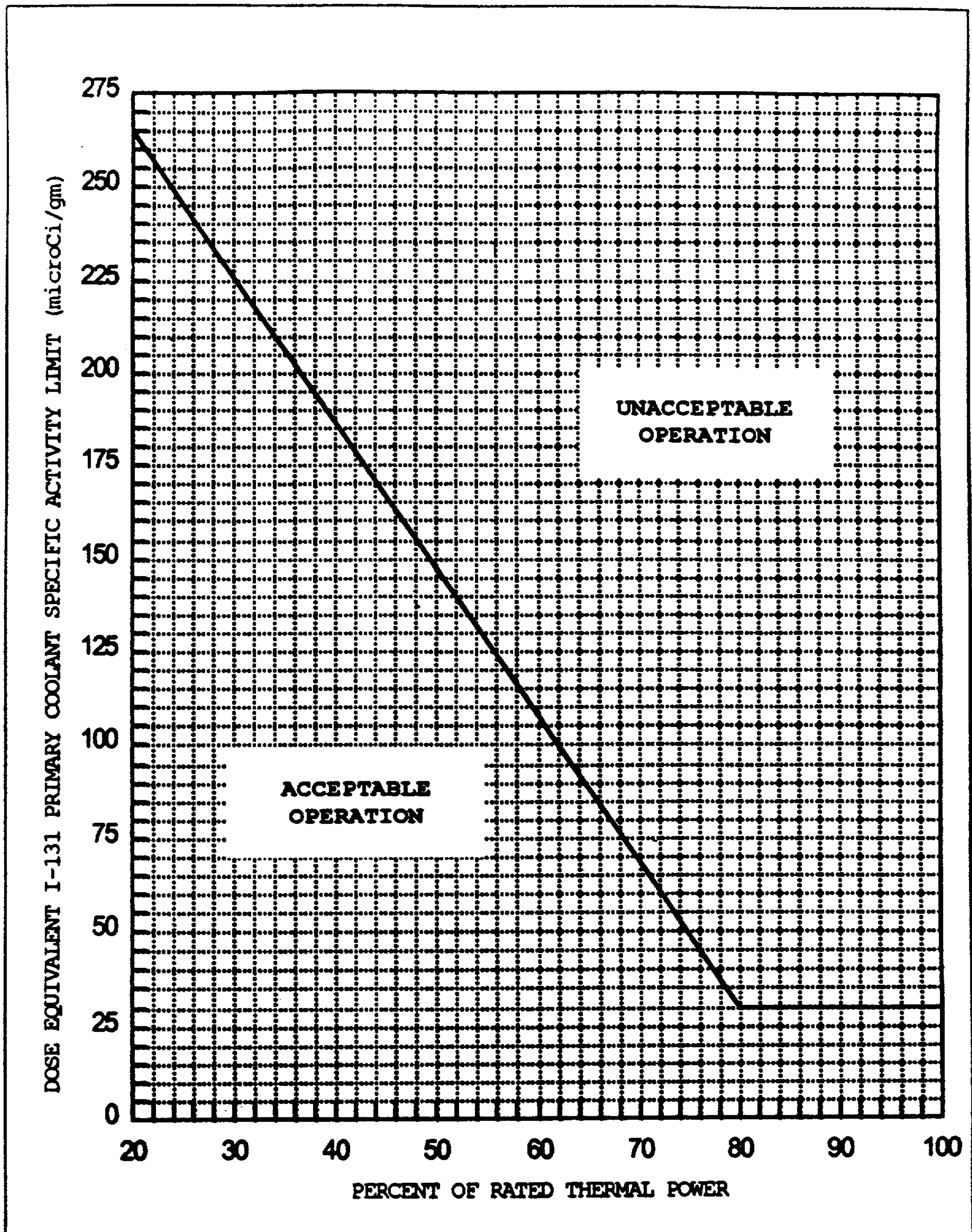

FIGURE 3.4-1

DOSE EQUIVALENT I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 0.5 µCi/gram Dose Equivalent I-131

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 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 repair limit for ODSCC at tube support plate intersections is based on the analysis contained in WCAP-12871, Revision 2, "J. M. Farley Units 1 and 2 SG Tube Plugging Criteria for ODSCC at Tube Support Plates," and documentation contained in EPRI Report TR-100407, Revision 1, "PWR Steam Generator Tube Repair Limits - Technical Support Document for Outside Diameter Stress Corrosion Cracking at Tube Support Plates." The application of this criteria is based on limiting primary-to-secondary leakage during a steam line break to ensure the applicable Part 100 limits are not exceeded.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or repair will be required for all tubes with imperfections exceeding 40% of the tube nominal wall thickness. If a sleeved tube is found to have through wall penetration of greater than or equal to 31% for the mechanical sleeve and 37% for the laser welded sleeve of sleeve nominal wall thickness in the sleeve, it must be plugged. The 31% and 37% limits are derived from R. G. 1.121 calculations with 20% added for conservatism. The portion of the tube and the sleeve for which indications of wall degradation must be evaluated can be summarized as follows:

3/4.4.7 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.7.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.7.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 31 GPM with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the accident analyses.

The surveillance requirements for RCS Pressure Isolation Valves provide added assurance of valve integrity, thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS Pressure Isolation valves is IDENTIFIED LEAKAGE and will be considered a portion of the allowed limit.

The total steam generator tube leakage limit of 450 gallons per day for all steam generators and 150 gallons per day for any one steam generator ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The limits are consistent with the assumptions used in the analysis of these accidents. The 150 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

3/4.4.8 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with containment concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the containment concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.9 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits in the event of primary-to-secondary leakage as a result of a steam line break.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.5 microCuries/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 106 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 December 7, 1994, as supplemented February 14 and March 20, 1995, Southern Nuclear Operating Company (the licensee) submitted a request for changing the Joseph M. Farley Nuclear Plant, Unit 2, (Farley Unit 2) Technical Specifications (TS). The requested amendment revises, in part, TS 4.4.6.2, 4.4.6.4, 4.4.6.5, and 3.4.7.2 for the Farley Unit 2, Cycle 11 operation to permit the use of a voltage-based steam generator tube repair criteria for defects confined within the thickness of the tube support plate. The February 14 and March 20, 1995, letters provided clarifying information that did not change the December 7, 1994, application and the proposed no significant hazards consideration determination or expand the scope of the original Federal Register notice.

2.0 BACKGROUND

The staff previously approved similar requests from the licensee to apply the voltage-based tube repair criteria at Farley Unit 2. Implementation of the voltage-based tube repair criteria for the ninth operating cycle was approved as documented in Amendment No. 87 to Facility Operating License No. NPF-8 issued April 1, 1992; as corrected by letter dated April 22, 1992.

Similarly, implementation of the voltage-based tube repair criteria for the tenth operating cycle was approved by Amendment No. 94 dated October 20, 1993. The staff concluded that the tube repair limits and leakage limits would ensure adequate structural and leakage integrity for indications accepted for continued service under the voltage-based repair criteria at Farley Unit 2 consistent with applicable regulatory requirements, for the ninth and tenth operating cycles.

This evaluation addresses comparable tube repair criteria for operating Cycle 11; however, in this amendment, the licensee has proposed to increase the voltage limits from 1.0/3.6 volts to 2.0/5.6 volts. Voltage limits of 2.0/3.6 volts were approved for Farley Unit 1 in Amendment No. 106 dated April 5, 1994.

The staff is currently developing a generic interim position on voltage-based limits for outside diameter stress corrosion cracking (ODSCC) confined to the thickness within the tube support plates. The NRC staff has published several conclusions regarding voltage-based repair criteria in draft NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes" and in a draft generic letter titled "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes." The latter document was published for public comment in the Federal Register on August 12, 1994 (59 FR 41520). However, the staff is continuing to evaluate an acceptable generic position that will take into consideration public comments on the draft generic letter cited above, domestic operating experience under the voltage-based repair criteria, and additional data which have been made available from European nuclear power plants. The NRC staff currently plans to document its final position on this matter in a generic letter. Pending completion and issuance of the staff's final generic position on the voltage-based tube repair criteria, the staff is continuing to evaluate voltage-based repair criteria proposals on a case-specific basis. Each of the case-specific evaluations of the voltage-based repair criteria are limited to one cycle of operation.

The licensee's current proposal is applicable to Cycle 11 operation and is similar to the licensee's previous proposals that were approved. Furthermore, the licensee's submittal is, for the most part, consistent with the draft generic letter issued for public comment on August 12, 1994, except as noted below.

3.0 PROPOSED INTERIM TUBE REPAIR CRITERIA

The Joseph M. Farley Nuclear Plant, Unit 2, TS 4.4.6.2, 4.4.6.4, 4.4.6.5, and 3.4.7.2 and Bases 3/4.4.6 and 3/4.4.7 are revised by this amendment request to specify the voltage-based tube repair criteria for ODSCC confined to within the thickness of the tube support plate. Modifications have been made to the previously approved (Cycle 9 and 10) TS pertaining to the implementation of the voltage-based tube repair criteria to make the currently proposed TS similar to those proposed in the draft generic letter. The changes in the TS for Cycle 11 implementation of the voltage-based tube repair criteria include, in part:

a. Specifying that tube support plate indications left in service as a result of application of the tube support plate plugging criteria shall be inspected by bobbin coil probe during the following refueling outages.

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

c. Changing the Cycle 10 repair limits for tube support plate intersections with indications of ODSCC from 1.0 and 3.6 volts to the following for Cycle 11:

1. Degradation attributed to ODSCC 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.

2. Degradation attributed to ODSCC 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 c.3 below.

3. Indications of potential degradation attributed to ODSCC 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 ODSCC degradation with a bobbin voltage greater than 5.6 volts will be plugged or repaired.

d. Adding the following reporting requirements:

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 the estimated leakage based on the actual measured 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 the 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.

e. Permanently reducing the limits on primary-to-secondary leakage through all steam generators to 450 gallons per day and 150 gallons per day through any one steam generator.

In addition to the above TS changes, the licensee has also made the following commitments for implementing the voltage-based tube repair criteria:

1. The requested actions of the draft generic letter will be followed with a few exceptions. Exceptions to the draft generic letter include the following items: (1) calibration of the bobbin coil, (2) use of the probe wear standard, (3) limiting new probe variability, (4) removing specimens for destructive examination and reporting of the results, and (5) the application of data exclusion criteria. These exceptions are discussed in Sections 4.1, 4.2, and 4.3 of this evaluation. In addition, the licensee has proposed not to include the mid-cycle

equation for determining the voltage limits in the event of a forced outage not attributable to ODSCC at the tube support plates pending issuance of the final generic letter.

2. Calculation of the conditional probability of burst and total leak rate during a main steam line break (MSLB) will follow the methodology described in WCAP-14277, "SLB Leak Rate and Tube Burst Probability Analysis Methods for ODSCC at TSP Intersections," dated January 1995. As discussed in WCAP-14277, these methods are intended to be in accord with the draft generic letter on voltage-based tube repair criteria.

3. The NRC will be notified prior to restart if any indications of primary water stress corrosion cracking (PWSCC) are detected at the tube support plate elevations. Furthermore, the data analysts will be briefed on the possibility that PWSCC can occur at tube support plate elevations.

4. A tube pull aimed at obtaining three (3) tube support plate intersections will be performed during this outage. The tube pull will be successful if at least two intersections are successfully removed.

5. No distribution cutoff will be applied to the voltage measurement variability distribution.

6. All intersections where copper signals interfere with the detection of flaws will be inspected with a motorized rotating pancake coil probe.

7. All intersections with large mixed residuals will be inspected with a rotating pancake coil probe.

8. All bobbin flaw indications with voltages greater than 1.5 volts will be inspected with a rotating pancake coil probe.

4.0 EVALUATION

4.1 Inspection Issues

The licensee intends to incorporate the inspection guidance of the draft generic letter into their inspection program with the exception of the bobbin coil calibration procedure, the implementation of limits on new probe variability, and the probe wear re-inspection requirements. For the calibration of the bobbin coil, the licensee intends to calibrate the bobbin coil on the 4-20 percent holes rather than the 4-100 percent holes recommended in the draft generic letter. For the limits on new probe variability, the licensee proposes to implement such limits when probes are available and certified to meet the limits in the draft generic letter. For the re-inspection of probes that do not meet the probe wear re-inspection requirements, the licensee proposes to use the same practices used during the last Farley Unit 1 steam generator inspections as discussed in a letter dated February 23, 1994.

The licensee has calibrated the bobbin coil on the 4-20 percent through-wall holes, since initial implementation of the voltage-based tube repair criteria in 1992. The staff has concluded that calibrating on the 4-20 percent through-wall holes rather than the 4-100 percent through-wall holes is acceptable based on (1) a review of the material provided in EPRI report NP-7480-L Volume 1 pertaining to assessing the use of 20 percent and 100 percent through-wall holes and 100 percent through-wall EDM slots, and (2) the results obtained by an independent contractor pertaining to the repeatability of voltage measurements between standards containing 20 percent through-wall holes, 100 percent through-wall holes, and 100 percent electromagnetic discharge method (EDM) notches. These two studies showed that the 20 percent through-wall holes were more reproducible and the voltage readings obtained on these holes were more repeatable. Although deeper defects are typically the ones of most concern and the 100 percent through-wall holes are more representative of these defects, the staff has concluded that the better reproducibility of the 20 percent holes and the better measurement repeatability provided with these holes, in conjunction with the limits on new probe variability on the other holes in the standard (i.e., the 40, 60, 80, and 100 percent through-wall holes), justifies calibrating the bobbin coil on the 4-20 percent through-wall holes. Although the new probe variability requirements may not be implemented during this outage at Farley Unit 2, the staff finds the licensee's proposal to calibrate the bobbin coil on the 4-20 percent through-wall holes to be acceptable for this one cycle. This is consistent with previous practice at Farley Unit 2.

With respect to implementing the limits on new probe variability discussed in the draft generic letter, the staff has concluded that pending finalization of the draft generic letter that the licensee's proposal on new probe variability is acceptable.

With respect to the use of alternate procedures (i.e., those which differ from the draft generic letter) for re-inspecting tubes that fail to meet the probe wear criterion, the staff has concluded that alternate probe wear methods may be used on a continuing basis provided an assessment is performed demonstrating that (1) they provide equivalent detection and sizing capability on a statistically significant basis when compared to the guidance in the draft generic letter and (2) they are consistent with current methods for determining the end-of-cycle (EOC) voltage distributions which are used in the tube integrity analyses. These assessments, along with the statistical criteria for demonstrating that the techniques are equivalent, should be provided to the NRC for review and approval. With respect to this cycle specific application, however, the NRC staff has concluded that the methods which have been previously employed for reinspecting tubes when a probe fails to meet the probe wear criterion are acceptable.

As a result of the potential for the possible development of primary water stress corrosion cracking (PWSCC) flaws at dented tube support plate intersections, the licensee will brief their eddy current analysts of the potential for PWSCC to occur at these locations. Furthermore, the licensee has agreed to notify the NRC prior to plant restart if any PWSCC indications are detected at the tube support plate elevations. The staff notes that PWSCC may be detected at tube support plate elevations. If this occurs, an evaluation may need to be performed to ensure that the voltage-based repair criteria is only applied to the ODSCC indications. In summary, the staff concludes that the inspection guidelines submitted by the licensee are acceptable since the proposed repair criteria is limited to one cycle, and the calibration, recording, and analysis requirements are consistent with the methodology used in the development of the databases and supporting evaluations.

4.2 Structural Integrity

4.2.1 Deterministic Structural Integrity Assessment

The licensee's tube repair limits are based on a correlation between the burst pressure and the bobbin voltage of pulled tube and model boiler data. This correlation is similar to that used in approving the voltage limits in the licensee's previous submittals and those used in the draft generic letter. The staff finds the licensee's proposed voltage limits acceptable given the current burst pressure/bobbin voltage database, the licensee's growth rates, and the non-destructive examination uncertainty estimates.

To confirm the nature of the degradation occurring at the tube support plate elevations, tubes are periodically removed from the steam generators for destructive analysis. Tube pulls confirm that the nature of the degradation being observed at the tube support plate elevations is predominantly axially oriented ODSCC and also provide data for assessing the reliability of the inspection methods and for supplementing existing databases (e.g., burst pressure, probability of leakage, and leak rate). The draft generic letter contains guidance that states utilities should remove 6 intersections for destructive examination every other outage. To follow the draft generic letter guidance on tube pulls, the licensee would need to pull 6 intersections from their steam generators during this outage since their last tube pulls were in 1990. Pending finalization of the final generic letter position on tube pulls, the staff has concluded that the licensee should remove tubes for destructive examination at Farley Unit 2 during this outage. The staff has concluded that the licensee's commitment for obtaining additional pulled tube specimens with an objective of retrieving three intersections and obtaining a minimum of two intersections is acceptable. Furthermore, the staff has concluded that the licensee's commitment to provide the metallurgical results from these pulled tube specimens within 120 days is acceptable for this cycle specific application.

4.2.2 Probabilistic Structural Integrity Assessment

A probabilistic analysis for the potential for steam generator tube ruptures, given a MSLB, has been performed for the previous applications of this tube repair criteria. The draft generic letter contains additional guidance on this analysis. The licensee intends to perform this calculation per the guidance in the draft generic letter that will most likely result in a higher conditional probability of burst than would have been obtained using the previous methodology because it includes parametric uncertainty. The results of the probabilistic analysis will be compared to a threshold value of $1x10^{-2}$ per the guidance in the draft generic letter. This threshold value will provide assurance that the probability of burst is acceptable considering the assumptions of the calculation and the results of the staff's generic risk assessment for steam generators contained in NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity." Failure to meet the threshold value indicates that ODSCC confined to within the thickness of the tube support plate could contribute a significant fraction to the overall conditional probability of tube rupture from all forms of degradation that was assumed and evaluated as acceptable in NUREG-0844.

The licensee intends to calculate the conditional probability of burst per the guidance of the draft generic letter. The licensee referenced WCAP-14277, "SLB Leak Rate and Tube Burst Probability Analysis Methods for ODSCC at TSP Intersections," dated January 1995, as a document containing the details of the methodology for calculating the conditional probability of burst given a MSLB. The staff finds the licensee's proposal to perform the calculation per the guidance in the draft generic letter to be acceptable for this outage-specific application. As noted above, the NRC staff expects this calculation to result in a higher probability of burst than would have been calculated previously because it includes parametric uncertainty. The staff notes that all applicable data should be included in the burst pressure database when performing this calculation except as discussed below.

4.2.3 Data Exclusion from the Burst Pressure Database

During the performance of the pulled tube examinations, malfunctions in the test equipment or improper specimen preparation can occasionally occur which could result in erroneous readings. Data like this should not be included in a database because it could result in invalid results and/or conclusions. The staff, therefore, concluded in draft NUREG-1477 that eliminating data from the burst pressure database was appropriate provided that the data could be shown to be erroneous or the result of an invalid test. The staff provided additional guidance regarding the exclusion of data from the burst pressure database in a meeting with the industry on February 8, 1994. As a result of this guidance, the industry provided criteria (i.e., data exclusion criteria) for determining whether data may be removed from the burst pressure database in an April 22, 1994, letter from Electric Power Research Institute to NRC. This letter was referenced and discussed in the draft generic letter and in the licensee's submittal dated February 14, 1995.

The staff concluded that the exclusion of the burst pressure data points cited in the April 22, 1994, letter, from the burst pressure database is appropriate. However, the staff is continuing to assess the appropriateness of excluding data points from the burst pressure database on a case-by-case basis pending further review of the generic data exclusion criteria presented in the April 22, 1994, letter.

4.3 Leakage Integrity

4.3.1 Normal Operational Leakage

Consistent with prior amendments approving the use of the voltage-based repair criteria at Farley Unit 2, the licensee will continue to limit the amount of operating leakage through any one steam generator to 150 gallons per day (gpd) and will limit the amount of operating leakage through all steam generators to 450 gpd. This requirement will be made permanent with this amendment.

4.3.2 Accident Leakage

The licensee has proposed a model for calculating the steam generator tube leakage from the faulted steam generator during a postulated MSLB which consists of two major components: (1) a model predicting the probability that a given indication will leak as a function of voltage (i.e., the probability of leakage model); and (2) a model predicting leak rate as a function of voltage, given that leakage occurs (i.e., the conditional leak rate model).

The calculational methodology being proposed by the licensee for Farley, Unit 2 for determining the amount of primary-to-secondary leakage under postulated accident conditions has previously been reviewed and approved by the NRC staff in the Amendment No. 54 Safety Evaluation Related To Operating License NPF-72, Commonwealth Edison Company, Braidwood Station, Unit 1, Docket No. STN 50-456 dated August 18, 1994. The staff finds this methodology acceptable for Farley Unit 2. The staff notes that all applicable data should be included in the probability of leakage and conditional leak rate databases when performing this calculation except as discussed below. The staff notes that some minor variations in the details of the modeling may be necessary for the case where the p-value test is invalid at the 5 percent level. The staff, however, finds the licensee's proposal to perform the calculation using a methodology intended to follow the guidance of the draft generic letter to be acceptable.

The licensee has calculated the allowable steam generator leak rate in the faulted steam generator as discussed in Section 5.0. This value is intended to be consistent with maintaining the radiological consequences of a release outside containment to within a small fraction of the guideline values in 10 CFR Part 100. As a result, if the primary-to-secondary leakage during a postulated MSLB is less than this allowable limit, the steam generator tubing will maintain adequate leakage integrity under these conditions.

4.3.3 Data Exclusion from the Leakage Databases

During the performance of the pulled tube examinations, malfunctions in the test equipment or improper specimen preparation can occasionally occur which could result in erroneous readings. Data such as this should not be included in a database since it could result in invalid results and/or conclusions. The staff, therefore, concluded in draft NUREG-1477 that eliminating data from the steam generator leakage databases (i.e., the probability of leakage and the conditional leak rate databases) was appropriate provided that the data could be shown to be erroneous or the result of an invalid test. The staff provided additional guidance regarding the exclusion of data from the steam generator leakage databases in a meeting with the industry on February 8, 1994. As a result of this guidance, the industry provided criteria (i.e., data exclusion criteria) for determining whether data may be removed from the leakage databases in an April 22, 1994, letter from EPRI to the NRC. This letter was referenced and discussed in the draft generic letter and in the licensee's submittal dated February 14, 1995.

The staff has concluded that the exclusion of the probability of leakage data points cited in the April 22, 1994 letter, from the probability of leakage database is appropriate. Furthermore, the staff has concluded that exclusion of the conditional leak rate data points cited in the April 22, 1994, letter from the 7/8-inch conditional leak rate database, with the exception of model \boiler specimen 542-4 and pulled tube specimen J1-R8C74, is appropriate. However, pending further review of the generic data exclusion criteria presented in the April 22, 1994, letter, the staff is continuing to assess the appropriateness of excluding data points from the leakage databases on a case-by-case basis.

5.0 ASSESSMENT OF RADIOLOGICAL CONSEQUENCES

In support of the amendment request to apply a voltage-based repair limit for the Farley Unit 2 steam generator tube support plate intersections experiencing outside diameter stress corrosion cracking, the licensee stated that their assessment of the radiological dose consequences of a main steam line break accident was based upon an 11.4 gpm primary to secondary leak initiated by the accident. The licensee's conclusion as to the acceptability of the radiological doses also assumed an allowable activity level of dose equivalent ^{131}I of 0.5 μCi/g in the primary coolant and 0.1 μCi/g in the secondary coolant.

The staff has independently calculated the doses resulting from a main steamline break accident using the methodology associated with SRP 15.1.5, Appendix A. Two assessments were performed. One was based upon a pre-existing iodine spike activity level of 30 μCi/g of dose equivalent ^{131}I and the other was based upon an accident initiated iodine spike. The staff calculated doses for individuals located at the Exclusion Area Boundary (EAB) and at the Low-Population Zone (LPZ). The control room operator's thyroid dose was also calculated. The parameters which were utilized in the staff's assessment are presented in Table 1. The staff's calculations showed that the thyroid doses for the EAB and LPZ would be less than the limits established by SRP 15.1.5, Appendix A. The control room operator thyroid dose would be less than the limits of SRP 6.4 of NUREG-0800. Therefore, the staff concluded that, based upon a limit of 300 rem thyroid at the EAB for the pre-existing spike case and a limit of 30 rem thyroid for the accident initiated spike case and for all control room operator dose assessments, a leak rate of 11.4 gpm is an acceptable limit for the maximum primary to secondary leakage initiated by the steam line break accident.

6.0 SUMMARY OF EVALUATION

The licensee intends to follow the guidance of the draft generic letter on voltage-based tube repair criteria, except as noted above, for this cycle specific application. As a result, the staff concludes that adequate structural and leakage integrity can be ensured, consistent with applicable regulatory requirements, for indications to which the voltage-based repair criteria will be applied during Cycle 11 at Farley Nuclear Plant Unit 2. The staff's approval of the proposed voltage-based repair criteria is based, in part, on the licensee being able to demonstrate that the conditional probability of burst and the primary-to-secondary leakage during a postulated MSLB will be acceptable.

7.0 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.

8.0 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 requirement. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (60 FR 8754). 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 the amendment.

9.0 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.

Attachment:
Table 1

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John Hayes

Date: April 7, 1995

TABLE 1

INPUT PARAMETERS FOR FARLEY EVALUATION
OF MAIN STEAMLINE BREAK ACCIDENT

1. Primary coolant concentration of 30 μCi/g of dose equivalent ^{131}I.

Pre-existing Spike Value (μCi/g)		
^{131}I	=	23.1
^{132}I	=	8.3
^{133}I	=	37.0
^{134}I	=	5.6
^{135}I	=	20.4

2. Volume of primary coolant and secondary coolant.

Primary Coolant Volume (ft^3)	9146
Primary Coolant Temperature (°F)	578
Secondary Coolant Steam Volume (ft^3)	3742
Secondary Coolant Liquid Volume (ft^3)	2016
Secondary Coolant Steam Temperature (°F)	518.3
Secondary Coolant Feedwater Temperature (°F)	437.3

3. TS limits for DE ^{131}I in the primary and secondary coolant.

Primary Coolant DE ^{131}I concentration (μCi/g)	0.5
Secondary Coolant DE ^{131}I concentration (μCi/g)	0.1

4. TS value for the primary to secondary leak rate.

Primary to secondary leak rate, maximum any SG (gpd)	150
Primary to secondary leak rate, total all SGs (gpd)	450

5. Maximum primary/secondary leak rate to the faulted and intact SGs.

Faulted SG (gpm)	11.4
Intact SGs (gpm/SG)	0.1

6. Iodine Partition Factor

Faulted SG	1.0
Intact SG	0.1
Primary to Secondary Leakage	1.0

7. Steam Released to the environment
   Faulted SG (lbs/2 hours)... 91,000 plus primary/secondary leakage
   Intact SGs (lbs/2 hours)...479,000 plus primary/secondary leakage

8. Letdown Flow Rate (gpm) 60

9. Release Rate for 0.5 μCi/g of Dose Equivalent ^{131}I

		Ci/hr
^{131}I	=	4
^{132}I	=	9
^{133}I	=	9.7
^{134}I	=	14
^{135}I	=	9.7

10. Atmospheric Dispersion Factors

EAB (0-2 hours)	6.4×10^{-4}
LPZ (0-8 hours)	1.0×10^{-4}