April 5, 1994

Docket No. 50-348

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

Dear Mr. Morey:

SUBJECT: ISSUANCE OF AMENDMENT NO. 106 TO FACILITY OPERATING LICENSE NO. NPF-2 REGARDING INTERIM PLUGGING CRITERIA OF THE STEAM GENERATOR TUBE DEFECTS WITHIN THE BOUNDARY OF THE SUPPORT PLATE - JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1 (TAC NO. M88375)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 106 to Facility Operating License NPF-2 for the Joseph M. Farley Nuclear Plant, Unit 1. The amendment consists of changes to the Technical Specifications (TS) in response to your submittal dated December 9, 1993, as supplemented February 23, and April 1, 1994.

The amendment modifies TS 3/4.4.6, Steam Generators, and TS 3/4.4.9, Specific Activity, and their associated bases. The steam generator plugging/repair limit is being modified in the TS to incorporate a 2.0 volt steam generator tube support plate interim plugging criteria for Cycle 13 only. In addition, the TS limit for specific activity of dose equivalent I^{131} and its transient dose equivalent I^{131} reactor coolant specific activity will be reduced by a factor of 4 in order to increase the allowable leakage in the event of a steam line break for Cycle 13 only.

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

Sincerely,

ORIGINAL SIGNED BY WILLIAM H. BATEMAN FOR:

Byron L. Siegel, Senior Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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

cc w/enclosures:
See next page

*See Previous Concurrence

OFFICE	LA:PD21:DRPE	PM:PD21:DRPE	PRPB*
NAME	PDAnderson	BLSiegel	LJCunningham
DATE	4/5/94	4/5/94	04/04/94
OFFICE	OGC*	PD:PD21:DRPE	
NAME	MYoung	WHBateman	
DATE	04/04/94	4/5/94	

OFFICIAL RECORD COPY Document Name: G:\FARLEY\FAR88375.AMD

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

April 5, 1994

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Southern Nuclear Operating Co., Inc.
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Sincerely,

William H. Bateman for

Byron L. Siegel, Senior Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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

cc w/enclosures:
See next page

Mr. D. N. Morey
Southern Nuclear Operating
Company, Inc.

Joseph M. Farley Nuclear Plant

cc:

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

DOCKET NO. 50-348

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 106
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., dated December 9, 1993, as supplemented February 23, and April 1, 1994, 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. 106 , are hereby incorporated into the license. Southern Nuclear Operating Company, Inc., 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

William H. Bateman

William H. Bateman, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 5, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 106

FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages as indicated. The revised areas are indicated by marginal lines.

Remove Pages	Insert Pages
3/4 4-12	3/4 4-12
3/4 4-12a	3/4 4-12a
3/4 4-23	3/4 4-23
3/4 4-24	3/4 4-24
3/4 4-26	3/4 4-26
B3/4 4-3	B3/4 4-3
B3/4 4-3a	B3/4 4-3a
B3/4 4-5	B3/4 4-5

4.4.6.4 Acceptance Criteria

a. As used in this Specification:

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

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

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

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

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

6. Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be repaired (i.e., sleeved) or removed from service by plugging and is greater than or equal to 40% of the nominal tube wall thickness. For a tube that has been sleeved with a mechanical joint sleeve, through wall penetration of greater than or equal to 31% of sleeve nominal wall thickness in the sleeve requires the tube to be removed from service by plugging. For a tube that has been sleeved with a welded joint sleeve, through wall penetration greater than or equal to 37% of sleeve nominal wall thickness in the sleeve between the weld joints requires the tube to be removed from service by plugging. At tube support plate intersections, the repair limit for the Thirteenth Operating Cycle is based on maintaining steam generator tube serviceability as described below:

   a. An eddy current examination using a bobbin probe of 100% of the hot and cold leg steam generator tube support plate intersections will be performed for tubes in service.

   b. 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.

SURVEILLANCE REQUIREMENTS (Continued)

c. 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.6.d below.

d. 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 3.6 volts may remain inservice if a rotating pancake coil probe (RPC) inspection does not detect degradation. Indications of outside diameter stress corrosion cracking degradation with a bobbin voltage greater than 3.6 volts will be plugged or repaired.

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

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

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

REACTOR COOLANT SYSTEM

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 0.25 microCurie per gram DOSE EQUIVALENT I-131 for the Thirteenth Operating Cycle only;

b. Less than or equal to 1.0 microCurie per gram DOSE EQUIVALENT I-131 for subsequent cycles;

c. 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. For the Thirteenth Operating Cycle only, with the specific activity of the primary coolant greater than 0.25 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

b. For subsequent cycles, with the specific activity of the primary coolant greater than 1.0 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

c. With the specific activity of the primary coolant greater than $100/\bar{E}$ microCurie per gram, be in at least HOT STANDBY with T_{avg} 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. For the Thirteenth Operating Cycle only, with the specific activity of the primary coolant greater than 0.25 microCurie per gram DOSE EQUIVALENT I-131 or greater than 100/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.

b. For subsequent cycles, with the specific activity of the primary coolant greater than 1.0 microCurie per gram DOSE EQUIVALENT I-131 or greater than $\overline{100}$/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.

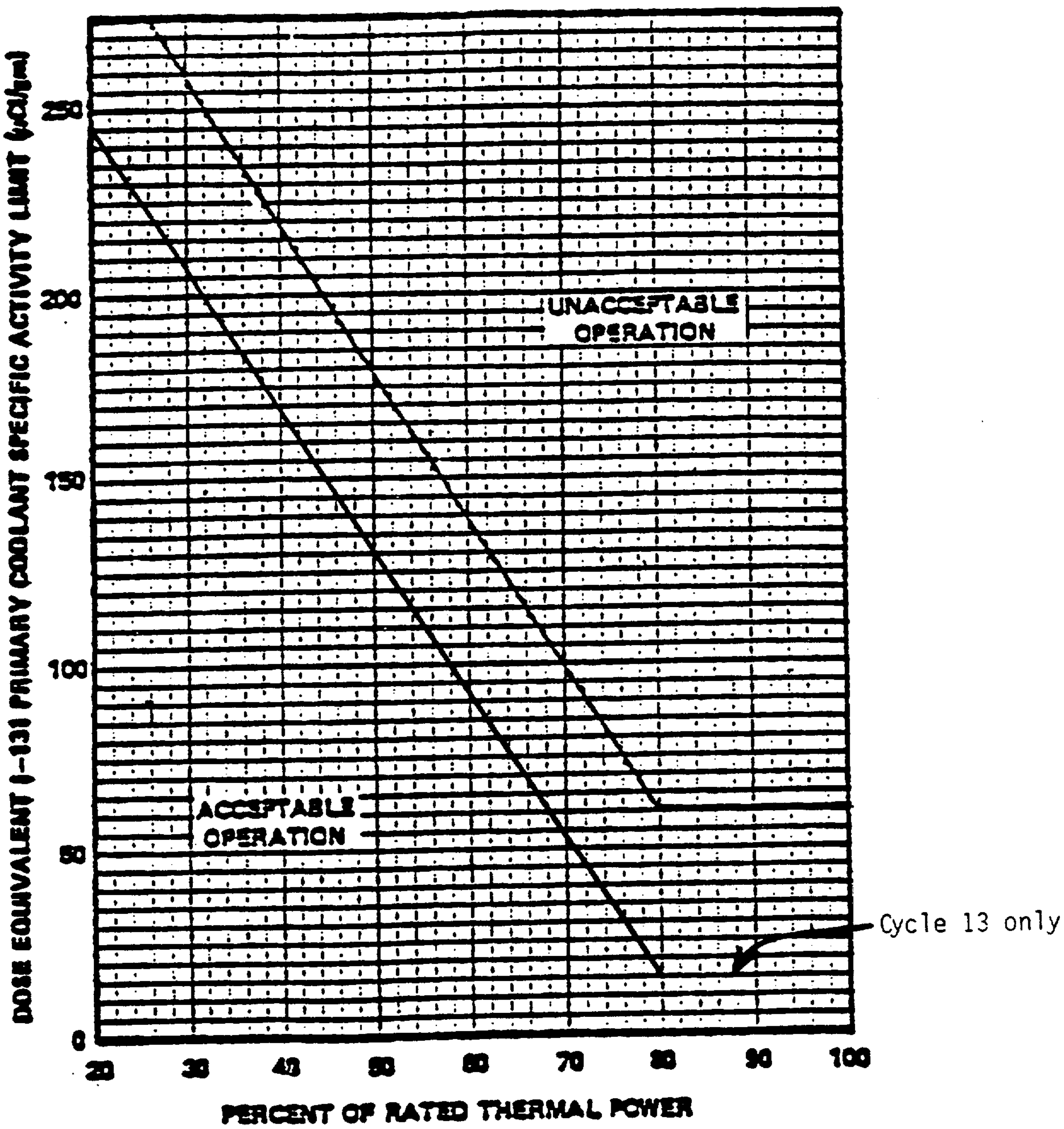

FIGURE 3.4-1

DOSE EQUIVALENT I–131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 1.0 μCi/gram Dose Equivalent I-131

(Activity > .25 μCi/gram Dose Equivalent I-131 for Cycle 13 only.)

3/4.4.6 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 140 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operational leakage of this magnitude can be readily detected by existing Farley Unit 1 radiation monitors. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged or repaired.

For the Thirteenth Operating Cycle only, the repair limit for tubes with flaw indications contained within the bounds of a tube support plate has been provided to the NRC in Southern Nuclear Operating Company letter dated December 09, 1993. The repair limit 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:

a. Mechanical

1. Indications of degradation in the entire length of the sleeve must be evaluated against the sleeve plugging limit.

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

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

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

b. Laser Welded

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

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

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

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

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

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

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

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 contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having 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 contaminant 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 following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Farley site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

For the Thirteenth Operating Cycle only, the limitations on the specific activity of the primary coolant have been reduced. The reduction in specific activity limits continues to 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 1.0 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 106 TO FACILITY OPERATING LICENSE NO. NPF-2

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-348

1.0 INTRODUCTION

By letter dated December 9, 1993, as supplemented by letters dated February 23, and April 1, 1994, Southern Nuclear Operating Company, Inc., the licensee, submitted a request to change the Technical Specifications (TS) for Joseph M. Farley Nuclear Plant, Unit 1 (Farley, Unit 1). The requested amendment would revise (1) TS 4.4.6.4 and Bases 3/4.4.6 to allow the continuance of a voltage-based steam generator tube plugging criteria for defects located at the tube support plate elevations, and (2) TS Figure 3.4-1, TS 3.4.9, and Bases 3/4.4.9 to allow for reduced Dose Equivalent I-131. All of the proposed changes are applicable to the Cycle 13 only.

The February 23, 1994, letter provided supplemental information and deleted the requested TS upper limit bobbin voltage of 5.7 volts for tube plugging that was requested in the December 9, 1993, letter, and retained the current value of 3.6 volts. The April 1, 1994, letter revises Bases Section 3/4.4.6 to reference the licensee's February 23, 1994, letter. The February 23, and April 1, 1994, supplements did not change the original no significant hazards consideration finding.

The proposed voltage criteria pertains specifically to outside diameter stress corrosion cracking (ODSCC) flaws. The proposed criterion (1) permits flaws within the bounds of the tube support plate elevations with bobbin voltages less than or equal to 2.0 volts to remain in service, (2) permits flaws within the bounds of the tube support plate with bobbin voltages greater than 2.0 volts but less than 3.6 volts to remain in service if a rotating pancake coil (RPC) probe does not detect degradation, and (3) requires flaw indications at the tube support plate elevations with bobbin voltages greater than 3.6 volts to be plugged or repaired.

The staff is currently developing a generic interim position on voltage-based limits for ODSCC at tube support plate elevations. The staff has published several tentative conclusions regarding voltage-based plugging criteria in draft NUREG-1477; however, the staff is continuing to evaluate an acceptable generic position which takes into consideration public comments received on draft NUREG-1477, domestic operating experience under the voltage-based repair criteria, and additional data which has been made available from European nuclear power plants. The staff currently plans to document its final position in a generic letter with the disposition of public comments being documented in the final version of NUREG-1477. In the meantime, pending completion and issuance of the staff's generic position on the voltage-based interim plugging criteria (IPC), the staff is continuing to evaluate IPC proposals on a case-specific basis, as necessary, to ensure that there is adequate assurance of public health and safety.

2.0 <u>BACKGROUND</u>

By letter dated February 26, 1991, Alabama Power Company submitted a steam generator tube support plate alternate plugging criteria (APC). As a result of technical issues raised during the review of this and other similar submittals, the full APC repair limit was not approved by the NRC staff; however, a reduced IPC repair limit was approved on a one-cycle basis. The modifications to the tube repair limits as a result of this IPC approval were documented in Amendment No. 95, dated October 8, 1992. The tube repair limits documented this amendment included a 1.0 volt repair criterion for axially oriented ODSCC flaws confined to within the thickness of the tube support plate in lieu of the depth-based limit of 40 percent. In addition, in the amendment the staff allowed bobbin indications between 1.0 and 3.6 volts to remain in service provided RPC inspection of these indications did not confirm the degradation to be present. The staff concluded in Amendment No. 95 that the proposed interim tube repair limits and leakage limits would ensure adequate structural and leakage integrity of the steam generator tubing at Farley, Unit 1, consistent with applicable regulatory requirements for Cycle 12. The licensee's current proposal is applicable to Cycle 13 and is similar to the licensee's previous proposal, except as noted below.

The licensee's current IPC proposal differs from the previously approved case-specific IPC for Farley, Unit 1, in several areas including:

1. <u>The determination of the tube structural limit.</u> Calculation of the tube structural limit has been based on maintaining a margin of safety of 1.43 against tube failure under postulated accident conditions vice maintaining a margin of safety of 3 against burst during normal operation.

2. <u>The IPC voltage limit.</u> A 2.0 volt limit vice a one volt limit has been proposed.

3. <u>The threshold for performing RPC examinations.</u> All flaw indications with bobbin voltages greater than 1.5 volts and less than 2.0 volts will be inspected by an RPC probe.

4. <u>The methodology for calculating postulated main steamline break (MSLB) leakage.</u> Steam generator tube leakage during a postulated MSLB will be calculated in accordance with the methods described in draft NUREG-1477.

5. <u>The diameter of the bobbin coil probe to be used in inspecting certain tubes.</u> A 0.640" bobbin coil probe vice a 0.720" probe has been proposed for use in inspecting intersections which can not be

inspected with the 0.720" probe (e.g., intersections between sleeves).

To evaluate the 2.0 volt IPC proposal for Farley, Unit 1, the staff considered not only the licensee's submittals but also operating experience from Farley Unit 2 (Southern Nuclear Operating Company, Inc., letter dated January 19, 1994), foreign operating experience, and public comments received on draft NUREG-1477. The inservice inspection results from Farley, Unit 2 (Fall 1993) were used to assess the IPC methodology, since Farley, Unit 2, was the first plant to operate a full cycle with a voltage-based IPC.

3.0 PROPOSED INTERIM PLUGGING CRITERIA

Farley, Unit 1, TS 4.4.6.4.a.6, "Plugging or Repair Limit" and Bases 3/4.4.6, "Steam Generators," are revised to specify the tube repair and leakage criteria for ODSCC at the tube support plate elevations for Cycle 13. The tube repair and leakage criteria are based on (1) the analysis in WCAP-12871, Revision 2, "J.M. Farley Units 1 and 2 SG Tube Plugging Criteria for ODSCC at Tube Support Plates," (2) documentation contained in the Electric Power Research Institute (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", and (3) analyses contained in the licensee's previously mentioned submittals. The changes to the tube repair and leakage criteria for Cycle 13 are described below:

1. An eddy current examination using a bobbin probe of 100 percent of the hot and cold leg steam generator tube support plate intersections will be performed for tubes in service.

2. Degradation attributed to ODSCC within the bounds of the tube support plate with a bobbin voltage less than or equal to 2.0 volts will be allowed to remain in service.

3. Degradation attributed to ODSCC 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) below.

4. 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 3.6 volts may remain in service if a RPC probe inspection does not detect degradation. Indications of ODSCC degradation with a bobbin voltage greater than 3.6 volts will be plugged or repaired.

In addition to the above TS changes, the licensee also made the following proposals/commitments for implementing the IPC:

1. All flaw indications with bobbin voltages greater than 1.5 volts and less than 2.0 volts will be inspected using an RPC probe. In addition, the licensee has stated that all flaw indications with bobbin voltages greater than 1.5 volts will be inspected using an RPC probe.

2. A sample RPC inspections of a minimum of 100 tube support plate intersections will be performed. This sample RPC inspection will include intersections with a bobbin dent voltage exceeding 5.0 volts. Inclusion of other intersections in the sample population will be based on inspecting intersections with artifact indications and intersections with unusual phase angles. Expansion of the sample plan, if required, will be based on the nature and number of the flaws discovered. In addition, the licensee has stated that all intersections with bobbin dent voltages exceeding 5.0 volts will be inspected with an RPC probe.

3. RPC flaw indications not found by the bobbin due to masking effects (due to denting, artifact indications, noise) will be plugged or repaired.

4. The NRC will be informed, prior to plant restart from the refueling outage, of any unexpected inspection findings relative to the assumed characteristics of the flaws at the tube support plate elevations. This includes any detectable circumferential indications or detectable indications outside the tube support plate.

5. The predicted MSLB leakage will be reported to the NRC prior to restart from the refueling outage.

6. The probability of tube burst, given a MSLB, will be reported to the NRC following completion of the refueling outage.

7. An assessment of the effectiveness of the IPC methodology will be provided to the NRC following completion of the refueling outage.

4.0 EVALUATION

4.1 Tube Integrity Issues

The purpose of the TS tube repair limits is to ensure that tubes accepted for continued service will retain adequate structural and leakage integrity during normal operating, transient, and postulated accident conditions, consistent with GDC 14, 15, 31 and 32 of 10 CFR Part 50, Appendix A. Structural integrity refers to maintaining adequate margins against gross failure, rupture, and collapse of the steam generator tubing. Leakage integrity refers to limiting primary-to-secondary leakage to within acceptable limits. The traditional strategy for accomplishing these objectives has been to establish a minimum wall thickness requirement in accordance with the structural criteria of Regulatory Guide 1.121, "Basis for Plugging Degraded PWR Steam Generator Tubes." Allowance for eddy current measurement error and flaw growth between inspections has been added to the minimum wall thickness requirements (consistent with the Regulatory Guide) to arrive at a depth-based repair limit. Enforcement of a minimum wall thickness requirement would implicitly serve to ensure leakage integrity (during normal operation and accidents), as well as structural integrity. It has been recognized, however, that defects, especially cracks, will occasionally grow entirely through-wall and develop small leaks. For this reason, limits on allowable primary-to-secondary leakage have been established in the TS to ensure timely plant shutdown before adequate structural and leakage integrity of the affected tube is impaired.

The proposed interim tube repair limits for Farley, Unit 1, consist of voltage amplitude criteria rather than the traditional depth-based criteria. Thus, the repair criterion represents a departure from the past practice of explicitly enforcing a minimum wall thickness requirement.

The industry-wide database from the pulled tube examinations show that for bobbin indications at or near 2.0 volts (i.e., the proposed IPC repair limit) maximum crack depths range between 50 percent and 100 percent through-wall. The likelihood of through-wall or near through-wall crack penetrations appears to increase with increasing voltage amplitude. For indications at or near 3.6 volts, the maximum crack depths have been found to generally range between 90 percent and 100 percent through-wall. Clearly, many of the tubes which will be allowed to remain in service under the proposed IPC may have or may develop through-wall or near through-wall crack penetrations during the upcoming cycle, thus creating the potential for leakage during normal operation and postulated MSLB accidents. The staff's evaluation of the proposed repair criteria from a structural and leakage integrity standpoint is provided in Sections 4.2 and 4.3, respectively. Section 4.4 contains the staff's evaluation of several inspection issues, and Section 4.5 addresses the importance of assessing the overall IPC methodology.

4.2 Structural Integrity

In support of the 1.0 volt repair limit approved in Amendment No. 95 for the Cycle 12, the licensee developed a burst pressure/bobbin voltage correlation to demonstrate that bobbin indications satisfying the 1.0 volt interim repair criterion would retain adequate structural margins during Cycle 12, consistent with the criteria of Regulatory Guide 1.121. The correlation was developed from both pulled tube data (using pre-pull bobbin voltages) and laboratory tube specimens containing ODSCC flaws. The bobbin voltage data used to construct the burst pressure/bobbin voltage correlation were normalized to be consistent with the calibration standard voltage set-ups and voltage measurement procedures described in WCAP-12871, Revision 2, and most recently in the guidelines contained in the licensee's February 23, 1994, submittal. The normalization was performed to ensure consistency among the voltage data in the burst pressure/bobbin voltage correlation and consistency between the voltage data in the correlation and the field voltage measurements at Farley, Unit 1.

For Farley, Unit 1, the most limiting burst pressure criterion of Regulatory Guide 1.121 is that degraded tubes shall retain a margin of three against burst under normal operating differential pressures across the tube. For Farley, Unit 1, this translates to a limiting burst pressure of 4380 psi. From the burst pressure/bobbin voltage correlation presented in EPRI report TR-100407, Revision 1, the maximum voltage which will satisfy this burst pressure criterion at a 95 percent prediction interval is 4.5 volts. Since during normal operation the support plates provide constraint against tube rupture, the margin of safety of three against rupture during normal operation is inherently satisfied for flaws contained within the bounds of the tube support plates. Therefore, for ODSCC within the bounds of the tube support plates, the licensee has proposed that the tube structural limit should be based on maintaining a margin of safety of 1.43, consistent with Regulatory Guide 1.121, against tube failure under postulated accident conditions (e.g., MSLB) vice a factor of safety of three against burst during normal operation. For Farley, Unit 1, this translates to a limiting burst pressure of 3660 psi. From the burst pressure/bobbin voltage correlation presented in EPRI report TR-100407, Revision 1, the maximum voltage which will satisfy this burst pressure criterion at a 95 percent prediction interval is 9.6 volts.

In the licensee's December 9, 1993, submittal, the 9.6 volt structural limit was adjusted to include an allowance of 20 percent for non-destructive examination (NDE) measurement uncertainty and an allowance of 50 percent for voltage growth over the next operating cycle to arrive at a 5.6 volt APC repair limit. The NDE measurement uncertainty estimate considers measurement uncertainties stemming from bobbin coil probe design characteristics, including wear characteristics and variability in the analysts' interpretation of the bobbin coil voltage. Potential flaw growth between inspections has been evaluated based on observed voltage amplitude changes during prior cycles at Farley, Unit 1. Over the last five cycles, the average percent growth of all indications has been 45 percent (1985 to 1986), 59 percent (1986 to 1988), 36 percent (1988 to 1989), 33 percent (1989-1991), and 26 percent (1991 to 1992). The 50 percent average voltage growth allowance used to support the 5.6 volt APC repair limit is intended to provide margins for variation in future growth rates at Farley, Unit 1.

For any specific individual tube, voltage measurement uncertainty and/or voltage growth may exceed the value assumed in the previously mentioned Regulatory Guide 1.121 deterministic analysis because the deterministic analysis does not consider the full tails of the voltage measurement uncertainty and voltage growth distributions. Similarly, the burst pressure for some tubes may be less than the 95 percent lower prediction interval values in the burst pressure/bobbin voltage correlation. These distribution tails may involve sizable numbers of tubes in instances where a large number of tubes with indications are being accepted for continued service. To directly account for these uncertainties, Monte Carlo methods will be used to demonstrate that the probability of burst during a postulated MSLB accident is acceptably low for the distribution of voltage indications being left in service. Under this approach, the beginning-of-cycle (BOC) indications left in service are projected to the end-of-cycle (EOC) by randomly sampling the NDE uncertainty probability distribution and the voltage growth per cycle probability distribution. The EOC voltage distribution, the distribution of burst pressures, and a distribution of material tensile properties are then randomly sampled many times (e.g., 1,000,000) in order to determine the probability of burst during a postulated MSLB. In the probability of burst calculation, the material tensile properties distribution is sampled to adjust the burst pressure correlation which is based on a flow stress of 75 ksi. This probabilistic analysis allows for the possibility of burst pressures below those that were used to construct the burst pressure versus bobbin voltage correlation.

The licensee's current submittal permits bobbin indications greater than 2.0 volts but less than 3.6 volts to remain in service if an RPC probe inspection does not detect a flaw, and it requires flaw indications with a bobbin voltage greater than 3.6 volts to be plugged or repaired. The staff notes that the 3.6 volts reflects an alternate plugging criteria (APC) repair limit that was derived in WCAP-12871 Revision 2. In WCAP-12871 Revision 2, the APC repair limit was based on a structural limit of three times the normal operating pressure differential. The maximum voltage which would satisfy this burst pressure criterion at a 95 percent prediction interval was 6.2 volts based on the data available at that time. A 3.6 volt APC repair limit was calculated from the 6.2 volt structural limit by including an allowance of 20 percent for NDE measurement uncertainty and a 50 percent allowance for voltage growth over the next operating cycle. Since the issuance of WCAP-12871 Revision 2 in February 1992, additional data has been added to the burst pressure database used in the development of this APC voltage limit and several of the existing data points in the database have been updated as a result of additional analysis. In addition, it has been proposed that the voltage limit should be derived from a structural limit of 1.43xMSLB pressure vice three times the normal operating differential pressure as a result of the constraint provided by the tube support plate during normal operation. This has resulted in a new APC repair limit of 5.6 volts for Farley, Unit 1.

To confirm the nature of the degradation occurring at the tube support plate elevations, the licensee has pulled several tubes from the steam generators at Farley, Units 1 and 2, during past outages. Tube pulls not only confirm the nature of the degradation but 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). Metallurgical examination performed on the tubes removed from Farley, Units 1 and 2, confirmed that the dominant flaw feature affecting tube integrity at Farley is ODSCC. These examinations also revealed the presence of a mixture of short axial and obliquely oriented cracks which formed a cell-like structure. The examination results demonstrated, however, that the dominant flaw feature affecting tube integrity was axial ODSCC. The maximum voltage indication removed during the 1992 Farley, Unit 1, outage was 3.3 volts, and the burst pressure for this specimen was approximately 5800 psi. An additional indication with a bobbin voltage of 3.2 volts exhibited a burst pressure of approximately 7000 psi. The staff believes that no additional pulled tube data is required to support implementation of the 2.0 volt IPC during the Refueling Outage 12 provided no unusual inspection findings (described previously) are found during the inspection.

The staff concludes that the proposed 2.0 volt interim criterion will provide adequate assurance that most tubes with indications which are accepted for continued service during Cycle 13 operation will meet the burst pressure criteria of Regulatory Guide 1.121 at the end of Cycle 13. The staff notes that the bounding value of voltage growth per cycle at Farley, Unit 1, during the last outage was 1.9 volts. The staff estimates the 1.9 volts to represent a bounding value, assuming no increase in corrosion rates over what has been observed previously at Farley, Unit 1. Assuming a 20 percent voltage measurement uncertainty for a 2.0 volt indication left in service, the EOC voltage is expected by the staff to be bounded by 4.3 volts. This is below the 9.6 volt structural limit evaluated by the licensee as the lower 95 percent confidence limit for meeting the burst pressure criterion of 1.43xMSLB pressure using the burst pressure correlation in EPRI Report TR-100407, Revision 1. The staff also concludes that for axially oriented ODSCC within the bounds of the tube support plates, a structural limit based on maintaining a margin of safety of 1.43 against tube failure under postulated accident conditions (e.g., MSLB) is acceptable since tube support plate constraint during normal operation inherently satisfies the margin of safety of three during normal operation.

The staff also concludes that the proposal to allow bobbin indications between 2.0 and 3.6 volts to remain in service provided that the RPC probe inspection does not confirm the degradation observed with the bobbin coil probe to be acceptable. The staff notes that short and/or relatively shallow cracks detected by the bobbin coil may sometimes not be detectable by the RPC probe, although the RPC probe is considered by the staff to be more sensitive to longer, deeper flaws which are of structural significance. The staff further notes that burst strength is not a unique function of voltage, rather for a given voltage there is a statistical distribution of possible burst strengths, as indicated in the burst pressure/bobbin voltage correlation. The staff believes that the burst pressure for bobbin indications which were not confirmed to be flaw-like by the RPC probe will tend to be at the upper end of the burst pressure distribution (i.e., exhibit a higher burst pressure). The 3.6 volt cutoff, such that all bobbin indications would be plugged or repaired (with or without confirming RPC indications), provides assurance that all excessively degraded tubes will be removed from service. The staff notes that even if a 3.6 volt indication were left in service, that assuming an allowance of 20 percent for measurement uncertainty and a growth rate of 3.8 volts (i.e., twice the maximum growth rate observed during the previous cycle), the EOC voltage would be 8.1 volts. This 8.1 volts provides significant margin relative to the 9.6 volt structural limit. The staff further notes that the projected leakage from these tubes (i.e., tubes with bobbin voltages between 2.0 and 3.6 volts which exhibited no detectable degradation during the RPC inspection) will be considered in the leak rate assessment performed by the licensee prior to plant restart. Thus, the staff finds the proposed exception to the 2.0 volt criterion to be acceptable.

Furthermore, the staff concludes that the methodology for calculating the conditional probability of burst given a MSLB, referenced above, should use a BOC distribution that includes (1) all indications including those that were not confirmed by the RPC probe to be degraded, and (2) non-detected ODSCC indications. In addition, the burst pressure correlation used in these calculations should include all data unless a specific error in either the burst pressure test or voltage measurement occurred. The licensee has committed to perform such an analysis following the inspection at Farley, Unit 1, to confirm an acceptably low probability of burst given a MSLB. The results of this analysis (which should consider the most recent burst pressure/bobbin voltage correlation and the most recent growth rate data) should be reported to the staff following completion of the refueling outage. The staff notes that such a calculation was performed following implementation of a 1.0 volt IPC at Farley, Unit 2, which indicated that implementation of a 1.0 volt repair criterion at that time would have yielded a conditional probability of burst given a MSLB of less than 10^{-4}. This value indicates a low probability of burst given a MSLB, approximately two orders of magnitude less than the value considered in the staff's generic risk assessment for steam generators contained in NUREG-0844.

4.2.2 Combined Accident Loadings

The licensee has evaluated the effects of combined safe shutdown earthquake (SSE) and loss-of-coolant accident (LOCA) loads and SSE plus MSLB loads on tube integrity, consistent with GDC 2 of 10 CFR Part 50, Appendix A. A combined LOCA plus SSE must be evaluated for potential yielding of the tube support plates which could result in subsequent deformation of the tubes. If significant tube deformation should occur, primary flow area could be reduced and postulated cracks in tubes could open up which might create the potential for in-leakage (i.e., secondary-to-primary leakage) under LOCA conditions. In-leakage during a LOCA would pose a potential concern since it may cause an increase in the core peak clad temperature (PCT).

The most limiting accident conditions for tube deformation considerations result for the combination of SSE and LOCA loads. The seismic excitation defined for steam generators is in the form of acceleration response spectra at the steam generator supports. In the seismic analysis, the licensee has used generic response spectra which envelop the Farley-specific response spectra. A finite element model of the Series 51 steam generator was developed and the analysis was performed using the WECAN computer program. The mathematical model consisted of three dimensional lumped mass, beam, and pipe elements as well as general matrix input to represent the piping and support stiffness. Interactions at the tube support plate shell and wrapper/shell connections were represented by concentric spring-gap dynamic elements. Impact damping was used to account for energy dissipation at these locations.

Prior qualification of the Farley, Unit 1, primary piping for leak-before-break requirements resulted in the limiting LOCA event being the break of a minor branch line. The licensee, however, has used the loads for the primary piping break as a conservative approximation. The principal tube loading during a LOCA is caused by the rarefaction wave in the primary fluid. This wave initiates at the postulated break location and travels around the tube U-bends. A differential pressure is created across the two legs of the tube which causes an in-plane horizontal motion of the U-bends and induces significant lateral loads on the tubes. The pressure time histories needed for creating the differential pressure across the tube are obtained from transient thermal-hydraulic analyses using the MULTIFLEX computer code. For the rarefaction wave induced loadings, the predominant motion of the U-bends is in the plane of the U-bend. Thus, the individual tube motions are not coupled by the anti-vibration bars and the structural analysis is performed using single tube models limited to the U-bend and the straight leg region over the top two tube support plates.

In addition to the rarefaction wave loading discussed above, the tube bundle is subjected to bending loads during a LOCA. These loads are due to the shaking of the steam generator caused by the break hydraulics and reactor coolant loop motion. However, the resulting tube support plate loads from this motion are small compared to those due to the rarefaction wave induced motion.

To obtain the LOCA induced hydraulic forcing functions, a dynamic blowdown analysis is performed to obtain the system hydraulic forcing functions assuming an instantaneous (1.0 msec break opening time), double-ended guillotine break. The hydraulic forcing functions are then applied, along with the displacement time-history of the reactor pressure vessel (obtained from a separate reactor vessel blowdown analysis) to a system structural model that includes the steam generator, the reactor coolant pump, and the primary piping. This analysis yields the time-history displacements of the steam generator at its upper lateral and lower support nodes. These time-history displacements formulate the forcing functions for obtaining the tube stresses due to LOCA shaking of the steam generator.

In calculating a combined tube support plate load, the licensee combined the LOCA rarefaction and LOCA shaking loads directly, while the LOCA and SSE loads were combined using the square root of the sum of the squares. The staff found this combination methodology acceptable. The overall tube support plate load was transferred to the steam generator shell through wedge groups located at discrete locations around the plate circumference.

The radial loads due to combined LOCA and SSE could potentially result in yielding of the tube support plate at the wedge supports, causing some tubes in the vicinity of the wedge supports to be deformed. Utilizing results from recent tests and analysis programs, the licensee has shown that tubes will undergo permanent deformation if the change in diameter exceeds a minimum threshold value. This threshold for tube deformation is related to the concern for tubes with pre-existing tight cracks that could potentially open during a combined LOCA plus SSE event. For Farley, Unit 1, the LOCA plus SSE loads were determined to be of such magnitude that none of the tubes (which are assumed to contain pre-existing tight cracks) are predicted to exceed this deformation threshold value and, therefore, will not lead to significant tube leakage.

The licensee has assessed the effect of SSE bending stresses on the burst strength of tubes with axial cracks. Tensile stress in the tube wall would tend to close the cracks while compressive stress would tend to open the cracks. On the basis of previously performed tests, the licensee has concluded that the burst strength of tubes with through-wall cracking is not affected by an SSE event.

Based on a review of the information provided by the licensee, the staff concludes that at Farley, Unit 1, no significant tube deformation or leakage is likely to occur during an SSE plus LOCA event. In addition, the burst strength of tubing with through-wall cracks is not affected by an SSE event.

4.3 Leakage Integrity

A number of the indications satisfying the proposed interim 2.0 volt repair limit can be expected to have, or to develop, through-wall or near through-wall crack penetrations during the next cycle, thus creating the potential for primary-to-secondary leakage during normal operation, transients, or postulated accidents. The staff finds that adequate leakage integrity during normal operating conditions is assured by the TS limits on allowable primary-to-secondary leakage. Adequate leakage integrity during transients and postulated accidents is demonstrated by showing that for the most limiting accident, assumed to occur at the end of the next cycle, the resulting leakage will not exceed a rate that will result in offsite dose limits being exceeded.

4.3.1 Normal Operating Leakage

Implementation of the voltage-based IPC includes a reduction in the TS reactor coolant system leakage limits that would usually be applicable for the one operating cycle to which the IPC applies. Specifically, for the voltage-based IPC, the standard 500 gallons per day (gpd) limit for primary-to-secondary leakage through any one steam generator would be reduced to 150 gpd, and the limit on the total primary-to-secondary leakage through all steam generators would be reduced from 1.0 gpm (1440 gpd) to 450 gpd.

The standard 500 gpd limit per steam generator is intended to ensure that through-wall cracks which leak at rates up to this limit during normal operation will not propagate and result in tube rupture under postulated accident conditions consistent with the criteria of Regulatory Guide 1.121. Development of the 150 gpd per steam generator interim leakage limit has utilized the extensive industry database regarding burst pressure as a function of crack length and leakage during normal operation. Based on leakage evaluated at the lower 95 percent confidence interval for a given crack size, the 150 gpd limit would be exceeded before the crack length reaches the critical crack length for MSLB pressures. Based on nominal, best estimate leakage rates, the 150 gpd limit would be exceeded before the crack length reaches the critical crack length corresponding to a burst pressure of three times normal operating pressure.

The interim leakage limits are more restrictive than the standard operating leakage limits in order to provide a margin of safety against rupture. The interim leakage limits are also intended to provide an additional margin to accommodate a rogue crack which might grow at much greater than expected rates or unexpectedly extend outside the thickness of the tube support plate, and thus provide additional protection against exceeding MSLB leakage limits. However, by Amendment No. 94, reactor coolant system leakage limits for Farley, Unit 1, contained in TS 3.4.7.2 were permanently reduced to 140 gpd through any one steam generator and 420 gpd total through all of the steam generators. These leakage limits were imposed for reasons unrelated to the voltage-based IPC and are more restrictive than the required interim leakage limits. Therefore, the staff finds the existing normal operating leakage limits in TS 3.4.7.2 to be acceptable for implementation of the IPC.

4.3.2 Accident Leakage

As the basis for estimating the potential leakage during MSLB accidents, Westinghouse has correlated leakage test data obtained under simulated MSLB conditions with the corresponding bobbin voltage amplitudes. The correlation is based on a linear regression fit of the logarithms of the corresponding leak rates and bobbin voltages. The leak rate data exhibits considerable scatter relative to the mean correlation. Thus, prediction intervals for leak rate at a given voltage have been established to statistically define the range of potential leak rates. As part of the on-going review of the APC, the staff is continuing to review the correlation of the leak rate data to bobbin voltage. The staff tentatively concluded in draft NUREG-1477 that no proven relationship between leakage rate and voltage presently exists and that the proposed approach fails to account for non-detected ODSCC that remains in service. The staff has also noted that there are very few leakage data points in the zero-to-three volt range.

However, until the issue of the leak rate versus voltage correlation is resolved, the staff has concluded that a voltage-based approach can be used if these non-conservatisms are accounted for and sufficient conservatisms are included in the analysis. Therefore, the licensee has committed to provide a calculation of potential MSLB leakage by a methodology designed to address the staff concerns. The methodology that the licensee will use to calculate the MSLB leakage is described in draft NUREG-1477. This methodology treats the leakage rate data as independent of voltage. The staff notes that the MSLB leakage analysis should be performed with the most recent leak rate data for 7/8-inch outside diameter tubing. In addition, the voltage growth distribution used in the leakage assessment should (1) consider the most recent voltage growth data (i.e., Cycle 12), and (2) be adjusted for the planned Cycle 13 duration. Evaluation of the acceptability of the estimated primary-to-secondary leakage rate for postulated accident conditions should be consistent with the current licensing basis of the plant.

The staff noted in draft NUREG-1477 that there was no theoretical basis for assuming a log logistic fit for the probability of leakage function. Furthermore, the staff noted that the form of fit could significantly affect the predicted leakage but that the results would vary depending on the EOC voltage distribution. The staff concludes, therefore, that for the 2.0 volt IPC at Farley, Unit 1, the most conservative (with respect to the overall leakage) of the six functional forms for the probability of leakage function (discussed in draft NUREG-1477) should be used in predicting the primary-to-secondary leakage during a postulated MSLB.

4.4 Inspection Issues

In support of the proposed interim repair limit, the licensee proposes to utilize the eddy current test guidelines provided in their February 23, 1994, submittal to ensure the field bobbin indication voltage measurements are obtained in a manner consistent with the development and analyses of the supporting databases. The proposed guidelines define, in part, the bobbin specifications, calibration requirements, specific acquisition and analyses criteria, and flaw recording guidelines to be used for the inspection of the steam generators.

The proposed inspection guidelines and other licensee commitments contain, in part, requirements to (1) record all indications regardless of voltage amplitude (required for assessing postulated MSLB leakage and probability of burst), (2) perform RPC inspections of 100 tubes, including tubes with bobbin dent voltages exceeding 5 volts and also including tube support plate intersections with artifact indications or indications with unusual phase angles (expansion of this sample, if required, will be based on the nature and number of the flaws discovered), (3) perform RPC examinations of all tubes with bobbin voltages in excess of 1.5 volts, and (4) inform the staff prior to Cycle 13 any unexpected RPC findings relative to the assumed characteristics of the flaws at the tube support plates (which includes any detectable circumferential indications or detectable indications extending outside the thickness of the tube support plate) and provide a safety evaluation, if applicable, to address these findings.

The staff notes that the proposed NDE guidelines contain modifications to the previously used guidelines. These modifications include, in part, a discussion on the adequacy of RPC probes (i.e., 1-, 2-, or 3-coil) at distinguishing crack characteristics and provisions to reduce the tube repair limit for tubes inspected with a probe where the probe wear limit was exceeded. The staff is reviewing these changes with respect to the generic implementation of a 2.0 volt repair criterion; however, the staff concludes that these modifications are acceptable for the 1994 refueling outage at Farley, Unit 1.

The staff notes that the original calibration procedure for the bobbin coil presented in earlier APC submittals, which requires setting the bobbin coil voltage amplitude from the 400/100 kHz differential channel from the four 100 percent through-wall holes, is preferred over the more recent guidelines which require calibration on the four 20 percent through-wall holes, as discussed in draft NUREG-1477. In addition, the staff notes that there are several outstanding technical issues pertaining to the inspection guidelines, as documented in draft NUREG-1477, that will require resolution prior to adopting generic voltage limits.

As part of this IPC proposal, the licensee has proposed to use smaller diameter bobbin probes to inspect intersections which can not be accessed using the standard 0.720" bobbin probe (i.e., intersections between sleeved locations). To support the use of a smaller diameter bobbin probe, the licensee provided results from two plants, where a limited number of tubes were inspected with both the standard 0.720" bobbin probe and a smaller diameter bobbin probe (i.e., 0.560", 0.580", and 0.640" bobbin probes). The results from these tests demonstrated that the voltages measured with the smaller diameter bobbin probe were equal to or greater than the voltages measured with the larger diameter bobbin probe for the majority of the indications. However, the analysis provided was limited and did not account for the potentially higher noise levels on the detectability of the flaws when using smaller diameter probes. The staff concludes, therefore, that the use of smaller diameter bobbin probes is acceptable only if the licensee performs a more rigorous statistical analysis to demonstrate the adequacy of the smaller diameter bobbin probes not only to size but also to detect the indications. The analysis methodology for performing such a demonstration should be submitted for NRC review and approval. As a result of staff concerns, the licensee proposed in a letter dated February 23, 1994, that:

1. Prior to using smaller diameter bobbin probes in the implementation of the IPC, additional evaluation of the smaller probes will be performed, and

2. The evaluation used to support the use of smaller diameter bobbin probes will be discussed with the NRC staff prior to its implementation.

The staff finds this proposal acceptable for Farley, Unit 1; however, the statistical analysis methodology must be approved by the NRC staff prior to implementing IPC repairs on tubes inspected with the smaller diameter bobbin probes. The staff is also evaluating the generic aspects of using smaller diameter bobbin probes.

4.5 Overall Assessment

Draft NUREG-1477, issued by the NRC in June 1993, provided the conclusion of an NRC task force with regard to a 1.0 volt tube repair criteria. In that report the staff noted that there is not a unique relationship between eddy current voltage amplitude and crack depth and length and that this lack of a unique relationship is reflected in the scatter of the tube burst pressure and leakage data when plotted as a function of voltage. In this regard, the task group concluded that a voltage-based approach can be used if appropriate conservatisms are included in the statistical analysis. The staff has considered this conclusion in its current evaluation and has determined that adequate margin exist with regard to assumed burst pressure behavior, degradation rates, NDE variability, and leakage calculation to support this plant-specific implementation of a 2.0/3.6 volt IPC. The staff is continuing its evaluation of the public comments received on draft NUREG-1477 and notes that resolution of several outstanding technical issues (e.g., handling of outliers, limited pulled tube database above 3.6 volts, NDE uncertainty model, voltage growth model, need for additional operating experience, etc.) will be necessary to support a generic position on voltage repair limits. Several of the staff positions are supported by the most recent operating experience data (e.g., probability of detection adjustment to account for new indications, performance demonstration to reduce analyst variability, etc.) from Farley, Unit 2. The staff has concluded, however, that the 2.0/3.6 volt IPC is acceptable (as documented above) to ensure tube structural integrity for this plant-specific application.

The licensee has committed to perform an assessment following completion of the refueling outage of the effectiveness of the IPC methodology similar to that provided following the Unit 2 ninth refueling outage provided by letter dated January 19, 1994. The staff finds this proposal acceptable. Consistent with the assessment provided for Farley, Unit 2, this assessment should address any discrepancies between the predicted and actual values, and the following information should be included in this assessment in both tabular and graphical form:

1. EOC 11 voltage distribution - all indications found during the inspection regardless of RPC confirmation

2. Cycle 11 growth rate (i.e., from BOC 11 to EOC 11)

3. EOC 11 repaired indications voltage distribution - distribution of indications presented in (1) above that were repaired (i.e., plugged or sleeved)

4. Voltage distribution for indications left in service at the BOC 12 regardless of RPC confirmation - obtained from (1) and (2) above

5. Voltage distribution for indications left in service at the BOC 12 that were confirmed by RPC to be crack-like or not RPC inspected

6. Non-destructive examination uncertainty distribution used in predicting the EOC 12 voltage distribution

7. Projected EOC 12 voltage distribution

8. Actual EOC 12 voltage distribution - all indications found during the inspection regardless of RPC confirmation

9. Cycle 12 growth rate (i.e., from BOC 12 to EOC 12)

10. EOC 12 repaired indications voltage distribution - distribution of indications presented in (8) above that were repaired (i.e., plugged or sleeved)

11. Voltage distribution for indications left in service at the BOC 13 regardless of RPC confirmation - obtained from (8) and (10) above

12. Voltage distribution for indications left in service at the BOC 13 that were confirmed by RPC to be crack-like or not RPC inspected

13. Non-destructive examination uncertainty distribution used in predicting the EOC 13 voltage distribution

14. Projected EOC 13 voltage distribution

4.6 Radiological Consequences

As part of the Farley IPC TS request, Southern Nuclear Operating Company, Inc., proposed that the allowable limits for specific activity of reactor coolant contained in TS be reduced by a factor of four (from 1.0 to 0.25 microCuries per gram Dose Equivalent I-131) to enable a factor of four increase (5.7 gpm to 22.8gpm) in allowable post-MSLB primary-to-secondary leakage. The licensee concluded that the increased leakage estimates would be offset by the reduced TS limits on allowable reactor coolant activity.

The base analysis was provided in the licensee's June 4, 1992, letter (the licensee's response to the May 20, 1992, staff's Request for Additional Information). This analysis determined the maximum permissible steam generator primary-to-secondary leak rate during a main steamline break (MSLB) for both Farley units considering both the pre-accident and event-generated iodine spike cases. The licensee, in performing its analyses, considered the acceptance criteria of Standard Review Plan (SRP), Section 15.1.5, Appendix A. As a result of the June 4, 1992, analysis, the licensee concluded that the limiting primary-to-secondary steamline break leakage would be governed by the event-generated spike case and should be limited to 5.7 gallons per minute (gpm) so that accident consequences remain within SRP acceptance criteria.

The present request reduces the allowable limits for reactor coolant system specific activity by a factor of four, in order to allow an increase in steam generator leakage during a postulated MSLB (calculated per Section 4.3 above) by a similar factor of four (above the June 4, 1992 leakage limit of 5.7 gpm) and still meet SRP limits. The staff concludes that no increased radiological consequences would result from the increased projected leakage since the allowable TS specific activity limits are being reduced accordingly. The calculated MSLB leakage as determined by the methodology discussed in Section 4.3 of this Safety Evaluation must be below the proposed leakage limit (or additional tubes must be repaired until the leakage is within limits). Based on the above, the staff finds the proposed changes to TS 3.4.9, TS Figure 3.4-1 and Bases Section 3/4.4.9, to be acceptable.

4.7 Severe Accident Impact

Draft NUREG 1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes," (Section 4.4) addressed severe accident analysis with respect to steam generator tube IPC. The staff accepted IPC intending to maintain the current level of steam generator tube integrity, consistent with Regulatory Guide 1.121. This approach was considered credible since the degradation mechanism addressed is confined to regions within the tube support plate. The staff judged that expected tube performance would not be significantly impacted, so that high pressure severe accident analyses would not be affected.

The application proposing a revised IPC addresses analyses to demonstrate adequate tube structural and leakage integrity. These analyses ensure that the tube structural integrity guidelines of Regulatory Guide 1.121 are met. As detailed elsewhere in this evaluation, extending interim plugging criteria to include higher voltage indications does not significantly alter accepted tube integrity. Therefore, the tube behavior for normal operation or transients is not expected to be markedly degraded. It is the staff's judgement that the effect on high pressure severe accident response of this change is within the uncertainties associated with severe accident analysis capabilities. Therefore, the basis for the staff conclusion reported in NUREG 1477 regarding severe accident impact is unchanged. That is, the staff judges that under a higher voltage IPC, expected tube performance would not be impacted sufficiently to alter high pressure severe accident analyses.

5.0 SUMMARY

Based on the above evaluation, it can be concluded that adequate structural integrity of the steam generator tubing can be ensured for Cycle 13 at Farley, Unit 1, consistent with applicable regulatory requirements. In addition, the staff concludes that the methodology for determining the expected primary-to-secondary leakage during a postulated MSLB at the end of fuel Cycle 13 for

Farley, Unit 1, is acceptable. The staff's approval of the proposed interim repair limit is based on the licensee being able to demonstrate that the primary-to-secondary leakage during a postulated MSLB will be acceptable. The licensee has agreed to report, prior to restart from the Refueling Outage 12, the results of the MSLB leakage analysis. The licensee has also agreed to inform the NRC prior to plant restart from the refueling outage of any unexpected inspection findings relative to the assumed characteristics of the flaws at the tube support plates. This includes any detectable circumferential indications or detectable indications outside the tube support plate thickness. The licensee's proposed changes, to revise (1) TS 4.4.6.4 and Bases 3/4.4.6 to allow the continuance of a voltage-based steam generator tube plugging criteria for defects located at the tube support plate elevations, and (2) TS Figure 3.4-1, TS 3.4.9, and Bases 3/4.4.9 to allow for reduced Dose Equivalent I-131. All of the proposed changes are applicable to the Cycle 13 only. Therefore, based on the above, the staff finds that the proposed changes are acceptable.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of Alabama official was notified of the proposed issuance of the amendment. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 (59 FR 2879). 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.

8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: April 5, 1994

AMENDMENT NO. 106 TO FACILITY OPERATING LICENSE NO. NPF-2 - FARLEY, UNIT 1

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