October 11, 1996

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

SUBJECT: ISSUANCE OF AMENDMENT - JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2 (TAC NO. M95146)

Dear Mr. Morey:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 115 to Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant, Unit 2. The amendment changes the Technical Specifications (TS) in response to your submittal dated March 29, 1996, as supplemented by letters dated June 27, August 29, August 30 and September 16, 1996.

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

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

Sincerely,

(original signed by)
Jacob I. Zimmerman, Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-364

Enclosure:
1. Amendment No. 115 to NPF-8
2. Safety Evaluation

DFO1/1

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

October 11, 1996

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Vice President - Farley Project
Southern Nuclear Operating
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cc w/encls: See next page

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

ALABAMA POWER COMPANY

DOCKET NO. 50-364

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 115
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 March 29, 1996, as supplemented by letters dated June 27, August 29, August 30 and September 16, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-8 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 115, 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, 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: October 11, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 115

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 4-12a	3/4 4-12a
3/4 4-13	3/4 4-13
3/4 4-13a	3/4 4-13a
-------	3/4 4-13b
B 3/4 4-3	B 3/4 4-3
B 3/4 4-3a	B 3/4 4-3a
B 3/4 4-3b	B 3/4 4-3b*
-------	B 3/4 4-3c*

*overflow page - no changes

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

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

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

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

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

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

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

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

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

6. Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be repaired (i.e., sleeved) or removed from service by plugging and is greater than or equal to 40% of the nominal tube wall thickness. This definition does not apply for tubes that meet the F*/L*## criteria. For a tube that has been sleeved with a mechanical joint sleeve, through wall penetration of greater than or equal to 31% of sleeve nominal wall thickness in the sleeve requires the tube to be removed from service by plugging. For a tube that has been sleeved with a welded joint sleeve, through wall penetration greater than or equal to 37% of sleeve nominal wall thickness in the sleeve between the weld joints requires the tube to be removed from service by plugging. This definition does not apply to tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to 4.4.6.4.a.16 for the repair limit applicable to these intersections.

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

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

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

L* Criteria is applicable to Cycle 11 only.

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

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

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

   b. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit [2.0 volts], will be repaired or plugged except as noted in 4.4.6.4.a.16.c below.

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

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

d. If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in 4.4.6.4.a.16.a, 4.4.6.4.a.16.b, and 4.4.6.4.a.16.c. The mid-cycle repair limits are determined from the following equations:

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

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

where:

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

Implementation of these mid-cycle repair limits should follow the same approach as in TS 4.4.6.4.a.16.a, 4.4.6.4.a.16.b, and 4.4.6.4.a.16.c.

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

4.4.6.5 Reports

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

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

1. Number and extent of tubes and sleeves inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged or repaired.

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

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

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

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

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

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

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

L* Criteria is applicable to Cycle 11 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 = 150 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operational leakage of this magnitude can be readily detected by existing Farley Unit 2 radiation monitors. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged or repaired.

The voltage-based repair limits of 4.4.6.4.a.16 implement the guidance in GL 95-05 and are applicable only to Westinghouse-designed steam generators (SGs) with outside diameter stress corrosion cracking (ODSCC) located at the tube-to-tube support plate intersections. The voltage-based repair limits are not applicable to other forms of SG tube degradation nor are they applicable to ODSCC that occurs at other locations within the SG. Additionally, the repair criteria apply only to indications where the degradation mechanism is dominantly axial ODSCC with no significant cracks extending outside the thickness of the support plate. Refer to GL 95-05 for additional description of the degradation morphology.

Implementation of 4.4.6.4.a.16 requires a derivation of the voltage structural limit from the burst versus voltage empirical correlation and then the subsequent derivation of the voltage repair limit from the structural limit (which is then implemented by this surveillance).

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

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

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

The mid-cycle equation in 4.4.6.4.a.16.d should only be used during unplanned inspections in which eddy current data is acquired for indications at the tube support plates.

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

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or repair will be required for all tubes with imperfections exceeding 40% of the tube nominal wall thickness. If a sleeved tube is found to have through wall penetration of greater than or equal to 31% for the mechanical sleeve and 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.

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

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

Unit 2 Cycle 11 Specific L* Criterion

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

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

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 115 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 March 29, 1996, as supplemented by letters dated June 27, August 29, August 30 and September 16, 1996, the Southern Nuclear Operating Company, Inc., et al. (the licensee), submitted a request for changes to the Joseph M. Farley Nuclear Plant, Unit 2, Technical Specifications (TS). The amendment requested staff approval of permanent voltage-based alternative repair criteria for steam generator tubes in the TS. The alternative repair criteria allow steam generator tubes, having outside diameter stress corrosion cracking (ODSCC) that is predominately axially oriented and that is confined within the tube support plates, to remain in service if the tube inspection and associated results satisfy the guidance in Generic Letter (GL) 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," issued on August 3, 1995. By letters dated June 27, August 29, August 30 and September 16, 1996, the licensee submitted additional information to clarify the changes to the proposed repair criteria, which did not change the scope of the March 29, 1996, application and the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

GL 95-05 outlined acceptance criteria and limitations for licensees considering implementation of voltage-based alternate repair criteria in plant technical specifications. The traditional regulatory framework for monitoring steam generator tube integrity establishes requirements for a minimum wall thickness in accordance with Regulatory Guide 1.121. The plant TS require that any tubes having degradation which exceeds 40 percent through wall will be repaired. The 40 percent limit is conservative for highly localized flaws such as pits, short cracks and in particular ODSCC that occurs at the tube support plates.

The voltage-based alternate repair criteria do not set limits on the depth of ODSCC to ensure tube integrity; instead it relies on correlating the eddy current voltage amplitude from a bobbin coil probe with the specific measurement of tube burst pressure and leak rate. This approach takes no credit for the tube support plates in preventing and/or reducing the likelihood of a tube from bursting and/or leaking during postulated accident conditions. It assumes that the degradation affecting the steam generator tubes at the tube support plate elevation is in the free span region of the tubes.

GL 95-05 specifies, in part, that: (1) the repair criteria is only applicable to axially oriented ODSCC located within the bounds of the tube support plates; (2) licensees should perform an evaluation to confirm that the steam generator tubes will retain adequate structural and leakage integrity until the next scheduled inspection; (3) licensees should adhere to specific inspection criteria to ensure consistency in methods between inspections; (4) tubes should be periodically removed from the steam generators to verify the morphology of the degradation and provide additional data for structural and leakage integrity evaluations; (5) the operational leakage limit should be reduced; (6) specific reporting requirements should be followed; and (7) the licensees' proposed TS should follow the model TS in Attachment 2 of the GL.

3.0 EVALUATION

Farley Unit 2 uses three Westinghouse model 51 steam generators. The tubes were fabricated using mill annealed alloy 600 material. Each steam generator has 3,388 tubes.

The staff approved three applications for the interim alternate repair criteria in the Farley Unit 2 TS in License Amendment No. 87 that was issued on April 1, 1992; License Amendment No. 94 on October 20, 1993; and License Amendment No. 106 on April 7, 1995. Each of the interim criteria is approved for specific operating cycles. The existing interim criteria will expire at the end of the current operating cycle in October 1996. The permanent alternate repair criteria will replace the interim criteria and will eliminate the need for periodic license amendments for the tube repair criteria addressed by GL 95-05.

The licensee has committed to follow the guidance in GL 95-05 for its proposed permanent alternate repair criteria. In addition, the licensee has proposed to incorporate verbatim the model TS in GL 95-05 into the Farley Unit 2 TS. Clarifications regarding the use of GL 95-05 are discussed below.

Section 1.b.1 of Attachment 1 to GL 95-05 specifies that the repair criteria do not apply to tube-to-tube support plate intersections where the tube with degradation may potentially collapse or deform as a result of the combined postulated loss-of-coolant accident and safe shutdown earthquake loadings. Licensees should perform or reference an analysis that identifies which intersections are to be excluded. The licensee submitted an analysis, WCAP-12871, Revision 2, as a part of its application for the interim alternate repair criteria on May 28, 1993. As a result of the licensee's analysis, no tubes need to be excluded from application of the voltage-based repair criteria. The staff agreed with the licensee's assessment as documented in License Amendment No. 94.

Section 2.b of Attachment 1 to GL 95-05 specifies criteria for an acceptable evaluation to confirm that the tubes will maintain adequate structural and leakage integrity until the next scheduled inspection. The evaluation includes a conditional burst probability calculation and a total leak rate calculation from the affected steam generators during a postulated main steam line break. The licensee stated that it will follow the methodology described in the Westinghouse report, "SLB Leak Rate and Tube Burst Probability Analysis Methods for ODSCC at TSP Intersections," WCAP-14277. The report prescribes deterministic and probabilistic methods. The staff has approved the probabilistic method in WCAP-14277 for calculating the leak rate and tube burst probability because it is consistent with the acceptance criteria outlined in GL 95-05. The licensee uses the probabilistic methodology to evaluate structural and leakage integrity of the tubes. The NRR staff will periodically verify the results of these calculations and assess the effectiveness of the methodologies to ensure that they are consistent with Section 2.b of Attachment 1 to GL 95-05.

Section 2.b.3(2) of Attachment 1 to GL 95-05 recommends that licensees use the latest NRC-approved industry database (e.g., burst pressure, probability of leakage, and conditional leak rate database) in their tube integrity evaluations that should include calculation of tube repair limits, conditional burst probability, and total leakage under postulated accident conditions. The licensee stated that the latest NRC-approved database, using the NRC-approved data exclusion criteria, will be applied to the tube integrity evaluations. For the upcoming operating cycle, the licensee committed to use the database forwarded to the NRC by Duquesne Light Company letter for Beaver Valley Unit 1, dated March 27, 1996. The staff finds that the database submitted by Duquesne Light Company is acceptable for the tube integrity calculations for the upcoming operating cycle.

For the long-term, Nuclear Energy Institute (NEI) is developing a protocol for updating the steam generator degradation database. The staff will review the adequacy of the updating process and the associated database. Pending the implementation of an NRC-approved process for updating a generic industry database for steam generator tube degradation, the licensee will provide, as specified in GL 95-05, the database it intends to use prior to each refueling outage. The database will include the data from tubes that have been pulled and tested up to 2 months before the outage. The staff finds that the licensee's use of the database satisfies Section 2.b.3(2) of Attachment 1 to GL 95-05.

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

Section 3.b of Attachment 1 to GL 95-05 specifies guidance for tube inspection using the rotating pancake coil. The licensee stated that it will use a motorized rotating coil probe, e.g., the +Point coil, in addition to the rotating pancake coil. The licensee provided this clarification to ensure that the +Point coil can be used as an alternative to the rotating pancake coil. The staff finds that the use of the +Point coil is consistent with Note 1 on page 3 of GL 95-05 and is acceptable.

Sections 3.c.2 and 3.c.3 of Attachment 1 to GL 95-05 specify guidance in regard to probe wear and variability, which the licensee will follow. In addition, the licensee will follow the guidance in the two letters from NEI to NRC dated January 23 and February 23, 1996, and the two letters from NRC to NEI dated February 9 and March 18, 1996. The licensee will verify that both the primary and mix frequencies of the probe will satisfy the ± 10 percent variability requirement. The staff finds the licensee's program for probe wear and variability is consistent with GL 95-05 and the staff position.

Section 3.c.5 of Attachment 1 to GL 95-05 specifies quantitative noise criteria (e.g., electrical noise, tube noise, calibration standard noise) for the probe. The licensee stated that quantitative noise criteria have historically been applied and will be incorporated in the data acquisition procedures. This allows noise levels to be addressed at the beginning of inspection. Probes are replaced before exceeding the noise criteria. If, upon measurement, the probe in use fails to meet the criteria, tubes tested with that probe since the last satisfactory measurement are reinspected. In addition, the Farley analysis procedures allow the analyst to require reinspection due to noise on a qualitative basis. The staff finds the licensee's use of quantitative noise criteria consistent with GL 95-05.

The proposed amendment revised TS 3/4.4.6, "Steam Generators" and associated Bases section as part of implementing the voltage-based repair criteria for steam generator tubes. Specifically, the licensee changed the following Surveillance Requirements sections in the TS: TS 4.4.6.2.1, "Steam Generator Tube Sample Selection and Inspection;" TS 4.4.6.4.a, "Acceptance Criteria;" and TS 4.4.6.5, "Reports." The changes incorporate the methodology of calculating the upper voltage repair limit and mid-cycle repair limits. The licensee also changed TS Bases Section 3/4.4.6, "Steam Generator," consistent with these changes as stated above. After its review, the staff concludes that the proposed TS changes satisfy the model technical specifications for the voltage-based repair criteria as specified in GL 95-05 and, therefore, are acceptable.

4.0 STAFF CONCLUSION

On the basis of the information submitted, the staff concludes that the proposed permanent alternate repair criteria for the steam generator tubes in Farley Unit 2 are consistent with GL 95-05 and are acceptable. The staff also 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. The licensee may incorporate the proposed permanent alternate repair criteria into the Farley Unit 2 TS.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes the surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (61 FR 25711 dated May 22, 1996). 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.

7.0 CONCLUSION

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

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Date: October 11, 1996