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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 117 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 7, 1994, as supplemented May 31, 1995, Southern Nuclear Operating Company (the licensee) submitted a request for changes to the Joseph M. Farley Nuclear Plant, Unit 1 (Farley Unit 1) Technical Specifications (TS). The requested amendment revises, in part, TS 4.4.6.2, 4.4.6.4, 4.4.6.5, 3.4.7.2, and 3.4.9 for the Farley Unit 1, Cycle 14 operation to permit the use of a voltage-based steam generator tube repair criteria for defects confined within the thickness of the tube support plate. The May 31, 1995, letter provided clarifying information that did not change the scope of the December 7, 1994, application and the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

The staff previously approved similar requests from the licensee to apply the voltage-based tube repair criteria at Farley Unit 1. Implementation of the voltage-based tube repair criteria for Farley Unit 1, Cycle 12 operation was approved as documented in Amendment No. 95 to Facility Operating License No. NPF-2 issued on October 8, 1992.

Similarly, implementation of the voltage-based tube repair criteria for Cycle 13 operation was approved by Amendment No. 106 dated April 5, 1994. The staff concluded that the tube repair limits and leakage limits would ensure adequate structural and leakage integrity for indications accepted for continued service under the voltage-based repair criteria at Farley Unit 1 consistent with applicable regulatory requirements, for operating Cycles 12 and 13.

This evaluation addresses comparable tube repair criteria for operating Cycle 14; however, in this amendment, the licensee has proposed to increase the voltage limits from 2.0/3.6 volts to 2.0/5.6 volts. Voltage limits of 2.0/5.6 volts were approved for Farley Unit 2 in Amendment No. 106 dated April 7, 1995. The NRC staff has published several conclusions regarding voltage-based repair criteria in draft NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes" and in Generic Letter 95-05 (GL 95-05) titled, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes," dated August 3, 1995. Although GL 95-05 has been issued since the submittal of the licensee's original application dated December 7, 1994 (which requested a permanent TS change), the staff believes that sufficient time was not available for the staff to develop and grant a generic TS change and support a restart date of mid-October for Unit 1. Therefore, this evaluation is case-specific and based on one cycle of operation as provided in the May 31, 1995, proposed TS change, which supplemented the licensee's original submittal.

The licensee's current proposal is applicable to Cycle 14 operation and is similar to the licensee's previous proposals that were approved. Furthermore, the licensee's submittal is consistent with GL 95-05, except as noted below.

3.0 PROPOSED INTERIM TUBE REPAIR CRITERIA

The Joseph M. Farley Nuclear Plant, Unit 1, TS 4.4.6.2, 4.4.6.4, 4.4.6.5, 3.4.7.2., and 3.4.9 and Bases 3/4.4.6, 3/4.4.7,and 3/4.4.9 are revised by this amendment request to specify the voltage-based tube repair criteria for ODSCC confined to within the thickness of the tube support plate. Modifications have been made to the previously approved (Cycles 12 and 13) TS pertaining to the implementation of the voltage-based tube repair criteria to make the currently proposed TS similar to those in GL 95-05. The TS changes for Cycle 14 implementation of the voltage-based tube repair criteria include, in part:

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

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

c. Changing the Cycle 13 repair limits for tube support plate intersections with indications of ODSCC from 2.0 and 3.6 volts to the following for Cycle 14:

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

2. Degradation attributed to ODSCC within the bounds of the tube support plate with bobbin voltage greater than 2.0 volts will be repaired or plugged except as noted in c.3 below.

3. Indications of potential degradation attributed to ODSCC within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts but less than or equal to 5.6 volts may remain in service if a rotating pancake coil inspection does not detect degradation. Indications of ODSCC degradation with a bobbin voltage greater than 5.6 volts will be plugged or repaired.

d. Adding the following reporting requirements:

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

1. If the estimated leakage, based on the actual measured end-of-cycle voltage distribution, would have exceeded the leak limit (for the postulated main steam line break utilizing licensing basis assumptions) during the previous operating cycle.

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

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

4. If the calculated conditional burst probability exceeds 1.0×10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.

e. Permanently reducing the limits on primary-to-secondary leakage through all steam generators to 420 gallons per day and 140 gallons per day through any one steam generator.

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

1. The requested actions of GL 95-05 will be followed with the following exceptions: (1) use of the probe wear standard, and (2) limiting new probe variability.

These exceptions are discussed in Section 4.1 of this evaluation. In addition, the licensee has proposed not to include the mid-cycle equation for determining the voltage limits in the event of a forced outage not attributable to ODSCC at the tube support plates.

2. Calculation of the conditional probability of burst and total leak rate during a main steam line break (MSLB) will follow the methodology described in WCAP-14277, "SLB Leak Rate and Tube Burst Probability Analysis Methods for ODSCC at TSP Intersections," dated January 1995. As discussed in WCAP-14277, these methods are intended to be in accord with the draft generic letter on voltage-based tube repair criteria. The methods as specified in the draft generic letter are unchanged in GL 95-05.

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

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

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

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

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

8. All bobbin flaw indications with voltages greater than 2.0 volts will be inspected with a rotating pancake coil probe.

4.0 EVALUATION

4.1 Inspection Issues

The licensee has committed to incorporate the inspection guidance of GL 95-05 into their inspection program with the exception of the limits on new probe variability and the probe wear re-inspection requirements. For the limits on new probe variability, the licensee proposes to implement such limits when probes are available and certified to meet the limits in the Generic Letter. For the re-inspection for probes that do not meet the probe wear re-inspection requirements, the licensee proposes to use the same practices used during the last Farley Unit 1 steam generator inspection as discussed in a letter dated February 23, 1994.

Section 3.c.2 of GL 95-05 specifies that the voltage response for the 40-percent to 100-percent through-wall holes of new bobbin coils calibrated on the 20-percent through-wall holes should not differ from the nominal voltage by more than ±10 percent. The licensee indicated that bobbin coil probes with the voltage response tolerances specified in GL 95-05 will not be available until approximately 6 months after the NRC issues the generic letter. The scheduled date for the inspection of the Farley Unit 1 steam generators is less than 2 months following the release of GL 95-05. The availability of the appropriate bobbin coil probes will be limited at the time of the inspection. Due to the difficulty in obtaining bobbin coil probes with the response characteristics specified in Section 3.c.2 of GL 95-05, the licensee's decision not to inspect with such probes in the Cycle 14 refueling outage is acceptable to the NRC staff.

With respect to the use of alternate procedures (i.e., those which differ from GL 95-05 for re-inspecting tubes that fail to meet the probe wear criterion), the staff has concluded that alternate probe wear methods may be used on a continuing basis provided an assessment is performed demonstrating that (1) they provide equivalent detection and sizing capability on a statistically significant basis when compared to the guidance in GL 95-05, and (2) they are consistent with current methods for determining the end-of-cycle (EOC) voltage distributions which are used in the tube integrity analyses. These assessments, along with the statistical criteria for demonstrating that the techniques are equivalent, should be provided to the NRC for review and approval. With respect to this cycle-specific application, however, the NRC staff has concluded that the methods that have been previously employed for re-inspecting tubes when a probe fails to meet the probe wear criterion are acceptable.

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

4.2 Structural Integrity

4.2.1 Deterministic Structural Integrity Assessment

The licensee's tube repair limits are based on a correlation between the burst pressure and the bobbin voltage of pulled tube and model boiler data. This correlation is similar to that used in approving the voltage limits in the licensee's previous submittals and those used in GL 95-05. The staff finds the licensee's proposed voltage limits acceptable given the current burst pressure/bobbin voltage database, the licensee's growth rates, and the non-destructive examination uncertainty estimates.

To confirm the nature of the degradation occurring at the tube support plate elevations, tubes are periodically removed from the steam generators for destructive analysis. Tube pulls confirm that the nature of the degradation being observed at the tube support plate elevations is predominantly axially oriented ODSCC and also provide data for assessing the reliability of the inspection methods and for supplementing existing databases (e.g., burst pressure, probability of leakage, and leak rate). GL 95-05 contains guidance that states utilities on an ongoing basis (follow-up) should pull an additional tube specimen with the objective of retrieving as many intersections as practical (minimum of two intersections). Furthermore, this tube pull should be obtained at the refueling outage following accumulation of 34 effective full power months of operation or at a maximum interval of three refueling outages, whichever is shorter, following the previous tube pull. The licensee's last tube pulls were in the fall of 1992 for Farley Unit 1. The staff has concluded that the licensee's commitment to obtain an additional pulled tube specimen with an objective of retrieving three intersections and obtaining a minimum of two intersections is consistent with the guidance contained in GL 95-05 and is therefore acceptable. Furthermore, the staff has concluded that the licensee's commitment to provide the metallurgical results from these pulled tube specimens within 120 days is acceptable for this cycle-specific application.

4.2.2 Probabilistic Structural Integrity Assessment

A probabilistic analysis for the potential for steam generator tube ruptures, given an MSLB, has been performed for the previous applications of this tube repair criteria. Additional guidance on this analysis is contained in GL 95-05. The licensee has committed to perform this calculation per the guidance in the Generic Letter that will most likely result in a higher conditional probability of burst than would have been obtained using the previous methodology because it includes parametric uncertainty. The results of the probabilistic analysis will be compared to a threshold value of 1×10^{-2} per the guidance in the Generic Letter. This threshold value will provide assurance that the probability of burst is acceptable considering the assumptions of the calculation and the results of the staff's generic risk assessment for steam generators contained in NUREG-0844, "NRC Integrated

Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity." Failure to meet the threshold value indicates that ODSCC, confined to within the thickness of the tube support plate, could contribute a significant fraction to the overall conditional probability of tube rupture from all forms of degradation that was assumed and evaluated as acceptable in NUREG-0844. The licensee has committed to calculate the conditional probability of burst per the guidance of GL 95-05. The licensee referenced WCAP-14277, "SLB Leak Rate and Tube Burst Probability Analysis Methods for ODSCC at TSP Intersections," dated January 1995, as a document containing the details of the methodology for calculating the conditional probability of burst given a MSLB. The staff finds the licensee's proposal to perform the calculation in accordance with the guidance in the Generic Letter to be acceptable for this outage-specific application. As noted above, the NRC staff expects this calculation to result in a higher probability of burst than would have been calculated previously because it includes parametric uncertainty. The staff notes that all applicable data should be included in the burst pressure database when performing this calculation.

4.3 Leakage Integrity

4.3.1 Normal Operational Leakage

Consistent with prior amendments approving the use of the voltage-based repair criteria at Farley Unit 1, the licensee will continue to limit the amount of operating leakage through any one steam generator to 140 gallons per day (gpd) and will limit the amount of operating leakage through all steam generators to 420 gpd. This requirement will be made permanent with this amendment.

4.3.2 Accident Leakage

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

The calculational methodology being proposed by the licensee for Farley Unit 1 for determining the amount of primary-to-secondary leakage under postulated accident conditions has previously been reviewed and approved by the NRC staff in the Amendment No. 106 Safety Evaluation related to Farley Unit 2, dated April 7, 1995. The staff finds this methodology acceptable for Farley Unit 1. The staff notes that all applicable data should be included in the probability of leakage and conditional leak rate databases when performing this calculation. The staff has concluded that the licensee's proposal to perform the calculation using a methodology that follows the guidance in GL 95-05 to be acceptable.

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

5.0 ASSESSMENT OF RADIOLOGICAL CONSEQUENCES

In support of the amendment request to apply a voltage-based repair limit for the Farley Unit 1 steam generator tube support plate intersections experiencing outside diameter stress corrosion cracking, the licensee stated that their assessment of the radiological dose consequences of a main steam line break accident was based upon an 11.4 gpm primary to secondary leak initiated by the accident. This leak rate is based on the results of an analysis submitted to the staff by the licensee in a letter dated June 4, 1992, that concluded the leak rate in the faulted steam generator should be 5.7 gpm. This leak rate has since been doubled to 11.4 gpm to account for a factor of two reduction in primary coolant activity. The licensee's conclusion as to the acceptability of the radiological doses also assumed an allowable activity level of dose equivalent ^{131}I of 0.5 $\mu Ci/g$ in the primary coolant and 0.1 $\mu Ci/g$ in the secondary coolant.

The staff has independently calculated the doses resulting from a main steamline break accident using the methodology in SRP 15.1.5, Appendix A. Two assessments were performed. One was based upon a pre-existing iodine spike activity level of 30 $\mu Ci/g$ of dose equivalent ^{131}I and the other was based upon an accident initiated iodine spike. The staff calculated doses for individuals located at the Exclusion Area Boundary (EAB) and at the Low-Population Zone (LPZ). The control room operator's thyroid dose was also calculated. The parameters which were utilized in the staff's assessment are presented in Table 1.

The staff's calculations showed that the thyroid doses for the EAB and LPZ are within the acceptance criteria presented in SRP 15.1.5, Appendix A of NUREG-0800. The control room operator thyroid dose would be less than the acceptance criteria presented in General Design Criterion 19. Since the calculated doses meet those acceptance criteria, the staff concluded that a leak rate of 11.4 gpm is an acceptable limit for the maximum primary to secondary leakage initiated by the steam line break accident.

6.0 SUMMARY OF EVALUATION

The licensee intends to follow the guidance of GL 95-05 on voltage-based tube repair criteria, except as noted above, for this cycle-specific application. As a result, the staff concludes that adequate structural and leakage integrity can be ensured, consistent with applicable regulatory requirements, for indications to which the voltage-based repair criteria will be applied during Cycle 14 at Farley Unit 1. The staff's approval of the proposed voltage-based repair criteria is based, in part, on the licensee's commitment to demonstrate that the conditional probability of burst and the primary-to-secondary leakage during a postulated MSLB will be acceptable.

7.0 STATE CONSULTATION

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

8.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes the surveillance requirement. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (60 FR 8754 dated February 15, 1995). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

9.0 CONCLUSION

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

Attachment:
Table 1

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Date: September 28, 1995

TABLE 1

INPUT PARAMETERS FOR FARLEY UNIT 1
EVALUATION OF MAIN STEAMLINE BREAK ACCIDENT

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

Pre-existing Spike Value (μCi/g)

^{131}I	=	23.1
^{132}I	=	8.3
^{133}I	=	37.0
^{134}I	=	5.5
^{135}I	=	20.3

2. Volume of primary coolant and secondary coolant.

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

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

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

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

Primary to secondary leak rate, maximum any SG (gpd)	140
Primary to secondary leak rate, total all SGs (gpd)	420

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

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

6. Iodine Partition Factor

Faulted SG	1
Intact SG	0.1
Primary to Secondary Leakage	1.0

INPUT PARAMETERS FOR FARLEY UNIT 1
EVALUATION OF MAIN STEAMLINE BREAK ACCIDENT
(continued)

7. Steam Released to the environment

Faulted SG (lbs/2 hours)	91,000 plus primary to secondary leakage
Intact SGs (lbs/2 hours)	479,000 plus primary to secondary leakage

8. Letdown Flow Rate (gpm) 60

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

	Ci/hr
^{131}I =	4
^{132}I =	9
^{133}I =	9.7
^{134}I =	14
^{135}I =	9.8

10. Atmospheric Dispersion Factors

EAB (0-2 hours)	6.5×10^{-4}
LPZ (0-8 hours)	1.0×10^{-4}