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U. S. Nuclear Regulatory Commission
ATTN.: Document Control Desk
Washington, D. C. 20555

Joseph M. Farley Nuclear Plant - Unit 1
Technical Specification Changes Associated With
Steam Generator Tube Support Plate Interim Repair Criteria

Gentlemen:

By letter dated February 26, 1991, Alabama Power Company submitted a steam generator tube support plate alternate plugging criteria. As a result of delays in approving the full voltage alternate repair criteria, a reduced voltage interim plugging criteria (IPC) was approved for Unit 1. The Unit 1 criteria were approved for one cycle of use on October 8, 1992. The one cycle interim plugging criteria is only valid until the next refueling outage currently scheduled for March 1994. Thus, Southern Nuclear is requesting a new interim plugging criteria for the next Unit 1 operating cycle. NRC approval of the Farley interim plugging criteria is requested by March 1, 1994.

In early June 1993, the NRC placed draft NUREG-1477 dated June 1, 1993, "Voltage Based Interim Plugging Criteria for Steam Generator Tubes - Task Group Report," in the public document room for comment. On the basis of recent discussions with the NRC Staff, a final version of NUREG-1477 is not anticipated until the spring of 1994. Draft NUREG-1477 dated June 1, 1993 contains many new recommendations for obtaining approval of an interim plugging criteria. In addition, it modifies several NRC positions approved in the first two interim plugging criteria. Southern Nuclear has reviewed the draft NUREG and has incorporated the appropriate recommendations for Plant Farley into its proposed interim plugging criteria.

Since approval of the first interim plugging criteria, the interim plugging criteria has become an integral part of the steam generator degradation management program at Farley. When coupled with laser welded sleeving and aggressive chemistry control, the interim plugging criteria provides Southern Nuclear with significant flexibility over the 40% through-wall criteria in managing our repair efforts in the Farley steam generators. However, with each outage, this flexibility is decreased. Tubes with indications less than the original 1.0 volt interim plugging criteria may grow to exceed the interim plugging criteria repair limit, thus requiring plugging or repair. In addition, tubes which have had sleeves installed at the tube sheet or lower tube support plates can no longer be inspected using the interim plugging criteria since the interim plugging criteria has only been approved for standard,

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i.e., .720", probes. As a result, the flexibility in managing steam generator repair efforts is being significantly reduced. Consequently, Farley is proposing an increased voltage interim plugging criteria and requesting concurrence for use of smaller diameter probes in tubes which have been sleeved.

This interim plugging criteria requests an increased repair voltage of 2.0 volts. Although this is an increased voltage, the safety analysis contained in Attachment 1 demonstrates the extremely conservative assumptions included in the development of the criteria. Use of a 2.0 volt repair criterion is predicted to result in allowing a significant number of tubes to remain in service which would have received unnecessary plugging/repair under a 1.0 volt interim plugging criteria.

In addition to requesting use of a 2.0 volt criterion for leaving tubes in service, Southern Nuclear is also proposing reductions in Farley's technical specification limits for specific activity of dose equivalent I^{131} and for transient dose equivalent I^{131} reactor coolant specific activity by a factor of four. The reduction in the limits for the allowable specific activity levels, coupled with the NRC predicted primary-to-secondary leakage (per draft NUREG-1477 dated June 1, 1993) in the event of a steam line break, will ensure that the site boundary dose levels do not exceed the appropriate Part 100 limits. This reduction in specific activity is identical to that approved for the Unit 2 interim plugging criteria.

Southern Nuclear is also requesting concurrence in the use of smaller diameter bobbin probes, e.g., 680 versus 720, for inspection of intersections which can not be accessed using the standard 720 probe. Use of the smaller probe would allow use of the interim plugging criteria at tube support plate intersections above installed sleeves at tube support plates or the tube sheet.

Since Southern Nuclear's last interim repair criteria submittal, additional data have been provided to the NRC Staff in other alternate/interim repair criteria submittals. These data points underscore the conservatism contained in a 2.0 volt interim plugging criteria for Farley Nuclear Plant. Furthermore, two tubes were pulled from Farley Unit 1 during its last outage. Although these two tubes had voltages of 3.3 and 3.2 volts, they exhibited no leakage under steam line break differential pressures. In addition, these same two tubes burst at differential pressures of 6,600 psig and 8,100 psig, respectively. These burst pressures are over twice the differential pressure expected in a steam line break.

Attachment 1 contains the safety analysis to support the amendment request. Attachment 2 contains the proposed interim voltage repair criteria including a sampling program. Attachment 3 contains the proposed changed technical specification pages in support of the interim plugging criteria. Attachment 4 provides a significant hazards evaluation for the proposed interim repair criteria in accordance with 10 CFR 50.92. Southern Nuclear has determined the proposed license amendment will not significantly affect the quality of the environment.

Revised eddy current guidelines are being developed for use during the Unit 1 outage. These guidelines, formerly referred to as Appendix A, provide probe specifications, calibration requirements, specific acquisition and analysis criteria, and flaw recording guidelines to be used in the inspection of steam generators. These guidelines will ensure that field bobbin indication voltage measurements will be obtained in a manner which results in voltage calls which are consistent with those used in development of the criteria. These guidelines will be provided to the NRC Staff on their completion.

A copy of these proposed changes is being sent to Dr. D. E. Williamson, the Alabama State Designee, in accordance with 10 CFR 50.91(b)(1).

If there are any questions, please advise.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

D. Morey

Dave Morey
Vice President
Farley Project

REM:maf-IPCUNIT1.R1

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 9th DAY OF December 1993

Martha Gayle Dow

Notary Public

My Commission Expires: November 1, 1997

Attachments

cc: Mr. S. D. Ebneter
Mr. T. A. Reed
Mr. B. L. Siegel
Mr. T. M. Ross
Dr. D. E. Williamson

Attachment 1

Safety Analysis

In Support of the

Technical Specification Changes Associated With

Steam Generator Tube Support Plate

Interim Repair Criteria

Joseph M. Farley Nuclear Plant - Unit 1
Technical Specification Changes Associated With
Steam Generator Tube Support Plate Interim Repair Criteria

Safety Analysis

During the last three Unit 1 and Unit 2 steam generator inspections at Farley Nuclear Plant, bobbin indications at tube support plates with bobbin voltage greater than 1 volt were repaired with an allowance for intersections with bobbin voltages less than 3.6 volts to remain in service if an RPC inspection did not detect a flaw. After two full operating cycles under these criteria, no significant primary-to-secondary leakage has been detected in either unit. This lack of primary-to-secondary leakage is directly attributable to the extensive examinations required for the implementation of an interim plugging criteria. The examinations required to implement an interim plugging criteria greatly exceed the requirements associated with the 40% plugging limit currently contained in the Farley Technical Specifications.

I. INTERIM PLUGGING CRITERIA

Southern Nuclear is proposing a 2.0 volt interim plugging criteria for the Unit 1 thirteenth cycle. The proposed interim plugging criteria consists of five facets:

A. a voltage repair criteria with associated inspection requirements and data analysis guidelines;
B. determination of an end-of-cycle voltage distribution;
C. calculation of leakage in the event of a steam line break;
D. a steam line break primary-to-secondary leakage limit; and
E. a reduced primary-to-secondary leakage limit.

Of these five facets, four have been reviewed and approved by the NRC during the approval process for the Farley Unit 2 interim plugging criteria for the Fall '93 outage. In fact, the calculation of leakage in the event of a steam line break and the determination of the steam line break primary-to-secondary leakage limit are based on guidance contained in draft NUREG-1477 and NUREG-0800. The determination of the end-of-cycle voltage distribution is the same methodology used since the first interim plugging criteria was approved. The reduced primary-to-secondary leakage limit has been implemented on Unit 1 as a 140 gpd limit based on circumferential cracking in the area at the top of the tube sheet. Therefore, the only portion of the revised interim plugging criteria which has not been previously reviewed is the justification of the 2.0 volt repair criteria.

Each of the facets is discussed in more detail below.

A. Voltage Repair Criteria

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," was submitted to the NRC on August 31, 1993. This report contains an updated database of

pulled tubes and model boiler specimens. Figure 3-2 of TR-100407 provides an updated burst curve for 7/8 inch tubes.

The appropriate burst curve has been developed using a linear regression of burst pressure versus log bobbin voltage. On the basis of the regression, a lower 95% prediction interval has been derived. The lower 95% prediction interval curve has been further adjusted downward by a factor equal to a lower bound estimate of tubing mechanical properties at 650°F divided by the mechanical properties of the tubes tested at room temperature.

On the basis of this curve, the following burst voltages can be determined.

Differential Pressure	Best Fit Curve	95% Lower Prediction Interval Adjusted for Mechanical Properties
2560 psi (ΔSLB)	>170 volts	32 volts
3660 psi (1.43 x ΔSLB)	65 volts	9.6 volts

As can be seen, a free span flaw with a voltage of 32 volts would be expected to withstand the differential pressure of a steam line break based on a lower 95% prediction interval. Adding a safety margin of 1.43 to the differential pressure results in a structural limit voltage of 9.6 volts for burst.

It should be noted that concerning the burst curve development, draft NUREG-1477 states, "The regression fit of the burst pressure data as a function of voltage is valid. The available burst pressure data indicates an empirical relationship between burst pressure and voltage."

The 9.6 volt structural limit is a beginning-of-cycle structural limit. Applying allowances for voltage growth and NDE uncertainty, as illustrated below, results in a voltage repair limit of 5.6 volts.

Structural Limit	9.6 volts	
- Allowance for NDE uncertainty	-1.2 volts	20% of repair limit
- Allowance for crack growth	-2.8 volts	50% of repair limit
Repair Limit	5.6 volts	

The allowance for NDE uncertainty and the allowance for crack growth are based on earlier Farley specific submittals. However, the 50% allowance for crack growth greatly exceeds the average growth rates found at Unit 1 since 1988.

Previous submittals have based the structural repair limit on a pressure of 3 x normal operating differential pressure (3 x ΔNOP), i.e., 4,380 psi. However, this approach is unnecessarily conservative and greatly exceeds the requirements of Regulatory Guide 1.121. Regulatory Guide 1.121 states, "The margin of safety against tube rupture under normal operating conditions should be not less than 3 at any tube location where defects have been detected." Since the interim plugging criteria pertains only to flaws within the bounds of the tube support plates, the safety factor of 3 against tube rupture pertains to those flaws within the bounds of

the tube support plate. However, during normal operations these flaws are contained by the tube support plates, thus precluding tube burst and negating a repair limit based on 3 x ΔNOP. On this basis, the structural repair limit developed above is based on maintaining the Regulatory Guide 1.121 burst margin for postulated accidents.

When compared to a 5.6 volt repair limit based on guidance from Regulatory Guide 1.121, a 2.0 volt interim plugging criteria provides a large margin to tube burst. The 5.6 volt repair limit already has significant conservatisms built into it. Use of the 2.0 volt interim plugging criteria at Farley Nuclear Plant is conservative based on many factors, among them:

1. A safety margin of 1.43 as prescribed in Regulatory Guide 1.121.
2. The burst curve is based on a lower 95% prediction interval. The prediction interval curve has been further adjusted downward by a factor equal to a lower bound estimate of tubing mechanical properties at 650°F divided by the mechanical properties of tubes tested at room temperature.
3. The voltages used in the burst curve are pre-pull voltages for pulled tubes. As a result, any damage done to the tube during the pulling process is not taken into account in developing the curve. As a result, actual burst voltages for the pulled tubes in the steam generator should exceed those included in the data base.
4. No credit is taken for the tube support plate constraining the tubes in a steam line break. Even under the assumption of clean, open crevices, all 7 plates would not be expected to displace to the extent to uncover flaws. Earlier submittals (WCAP-12871) have shown that tube support plate corrosion and crevice deposits would prevent significant displacement of all tube support plates in a steam line break event.
5. Many crevices in the Farley steam generators are packed, restricting any leakage which would occur from through wall flaws. No credit is taken for the packed crevices.
6. Out of 124 tests, the lowest voltage for which leakage has been detected in 7/8 inch tubing for steam line break differential pressures is 2.81 volts.
7. Out of 124 tests, the lowest voltage for which leakage exceeded 1 liter/hour in 7/8 inch tubing for steam line break differential pressures is 6.52 volts.
8. Inspection requirements for implementation of the interim plugging criteria are significantly more stringent than those in the current Technical Specifications. Current Technical Specifications allow for a 3% sample size. The interim plugging criteria requires a 100% inspection of the tube support plate intersections.
9. Farley Unit 1 has N16 monitors located on the main steam line from each steam generator in order to detect primary-to-secondary leakage from any steam generator. The N16 monitors are used to trend leak rates upon an increase in leakage with plant response times reflecting the rate of change in leakage.

10. Farley Nuclear Plant has used the same eddy current inspection vendors since development began for the alternate plugging criteria. This reduces inconsistencies in the bobbin coil inspection practices.
11. As a result of use of a consistent calibration methodology and the use of the same eddy current vendors, growth rate determinations at Farley are probably the most accurate to date.
12. Bobbin voltages are called by two independent eddy current inspection vendors. The highest called voltage is used in implementation of the interim plugging criteria.

As a result, Southern Nuclear believes that with a 2.0 volt interim plugging criteria, sufficient margin exists to ensure tube burst will not occur in a steam line break.

Structural Limit	Full Repair Limit	Requested IPC
9.6 volts	5.6 volts	2.0 volts

B. End-of-Cycle Voltage Distribution

Draft NUREG-1477 raises two issues concerning the determination of the end-of-cycle voltage distribution. The draft NUREG states:

- The method for estimating the distribution of voltage changes during the next operating cycle is appropriate for purposes of accounting for crack growth during the cycle in assessing burst and leakage integrity, assuming that crack growth and voltage growth rates are within the bounds of previous experience. However, crack growth rates may vary with time and may not necessarily be bounded by previous experience. Therefore, licensees should describe, as part of their interim plugging criteria submittals, actions being taken to mitigate the corrosive environment in the tube support plate crevices and to ensure that future growth rates and crack morphologies will be within expected bounds.

- The method for estimating the combined distribution of the variability of the voltage response to a given flaw and the variability of the voltage measurements among different analysts is appropriate to account for NDE measurement uncertainty in assessing burst and leakage integrity. This finding is subject to additional information being provided to the Staff to justify the use of a 20-percent cutoff to the distribution of voltage measurement variability. Alternatively, no cutoff should be applied to the combined distribution of voltage variability.

In response to these NUREG comments, discussions of Farley growth rates, action taken at Farley to mitigate the corrosive environment in the tube support plate crevices, and a justification of the use of a 20% cutoff to the distribution of voltage measurement uncertainty are provided below.

B.1. Farley Unit 1 Growth Rates

On the basis of the last Farley Unit 1 steam generator inspection, the average voltage growth rate for all flaws in all Unit 1 steam generators was 0.22 volts/cycle. This corresponds to an average percentage growth rate of 26% for all flaws. If only the flaws with a beginning-of-cycle voltage greater than 0.75 volts are analyzed, the average voltage growth rate is 0.23 volts/cycle, or 21% per cycle.

The largest growth rate found in Unit 1 for the last outage was 1.90 volts. This growth rate is significantly less than the 2.8 volts assumed in determining the voltage structural limit of 5.6 volts in the above discussion. In fact, using the bounding Unit 1 growth rate and an allowance for NDE uncertainty results in an expected end-of-cycle voltage of 4.3 volts, less than the structural limit based on 3 x ΔNOP of 4.5 volts, as illustrated below:

+2.0 volts	Proposed Interim Plugging Criteria
+1.9 volts	Bounding Unit 1 Growth Rate
+0.4 volts	20% Allowance for NDE Uncertainty
+4.3 volts	Maximum Expected Voltage Based on Bounding Assumptions

It should be noted that the 1992 bobbin voltage analyses together with the reanalysis of the 1991 data for the 1992 indications were performed such that a single analyst evaluated growth data on a given tube. This process results in consistent data interpretation to improve the accuracy of the growth rates.

Unit 1 growth rates have been tracked since 1985. Over the last five cycles, the average percent growth has been 45%, 59%, 36%, 33%, and 26%. The trend toward lower growth rates is a demonstration of the effectiveness of actions taken to mitigate the corrosive environment.

B.2. Chemistry Actions Taken to Mitigate the Corrosive Environment in Steam Generator Tube Support Plate Crevices

Vast improvements in knowledge and technology relative to steam generator chemistry have occurred in the last few years that have had great impact on control of steam generator tube degradation. Farley Nuclear Plant has been aggressive in implementation of new programs and technologies in the chemistry program to minimize corrosion in the steam generators, especially at the critical tube/tube support plate crevice region. These changes in the chemistry program have proven effective in transition from a caustic to a near neutral crevice environment in the Farley steam generators. The low eddy current voltage growth rates measured during the most recent outages on Unit 1 illustrate the benefit of maintaining a neutral crevice environment. Farley Nuclear Plant has implemented a three part chemistry program that includes:

a. Control of steam generator cleanliness;
b. Control of cation/anion ratio; and
c. Addition of boric acid to mitigate tube/tube support plate crevice corrosion.

The major elements of the program are discussed below.

a. Clean steam generators are maintained by avoiding build-up of sludge and minimizing the ingress of impurities which accelerate corrosion in the tube support plate crevices. The volume of sludge in the Farley Unit 1 steam generators has shown a steady decrease since the implementation of morpholine as a secondary pH control agent. Farley has recently transitioned from morpholine to ethanolamine as a pH control agent because it provides a higher liquid phase pH and should result in lower iron transport to the steam generators. Farley has also increased the secondary pH by increasing hydrazine concentrations in feedwater according to the EPRI Interim PWR Secondary Water Chemistry Recommendations for intergranular attack/stress corrosion cracking control. The increased hydrazine concentration ensures that a reducing environment exists in steam generators.

   Farley has also decreased the amount of impurity ingress by improving make-up water quality. A reverse osmosis ultra pure make-up water system was added in 1991 that significantly improved the quality of make-up water to the steam generators. Farley also utilizes, on an as-needed basis, mid-cycle flushes for sodium removal and performs sludge lancing, pressure pulse cleaning, and steam generator fill and drain activities during outages to aid in the removal of sludge and impurities.

b. In addition to maintaining clean conditions in the steam generators, balancing the molar ratio of impurities that do enter the steam generators is extremely important. The second element of the Farley steam generator chemistry program to reduce crevice attack is control of the cation/anion balance to avoid caustic conditions. A ratio control program has been established to ensure that proper cation/anion balance is achieved. This has been accomplished by lowering the sodium ingress and achieving a proper cation/anion balance from the existing ultra pure make-up water system. Farley performs hideout return measurements for each refueling outage to determine existing crevice chemistry using MULTEQ and molar ratio evaluations. Chemistry experts at Westinghouse, NWT, and EPRI also review and evaluate the hideout return data to ensure the steam generators are properly characterized. The results from these evaluations are then compared to NDE examinations to determine any correlations.

c. The third program element is addition of boric acid to mitigate denting and intergranular attack/stress corrosion cracking in the tube/tube support plate crevices. The addition of boric acid provides some buffering effect in the crevice region, and appears to have contributed to reduced plugging/sleeving rates. On the basis of the above, Farley continuously adds boric acid to the steam generators.

Other inhibitors, such as titanates, are currently being evaluated by Southern Nuclear for possible implementation at Farley in the near future. Southern Nuclear will continue to aggressively monitor and trend chemistry data to ensure program success.

B.3. Justification of the Use of a 20% Cutoff to the Distribution of Voltage Measurement Uncertainty

The 20% cutoff value is based on Farley specific guidelines for resolution of voltage calls by two or more independent analysts. Analysts are provided extensive guidance, probably the most detailed guidance in the industry, for making conservative bobbin voltage calls for detected flaws. If the difference in voltage calls between the independent analysts exceed 20% and one or more of the voltage calls exceed the voltage threshold for interim plugging criteria implementation, a guideline calls for resolution by the lead analyst. As a result of these multiple analyst reviews and lead analyst resolution, it is reasonable to use the 20% cutoff for voltage uncertainty.

C. Calculation of Predicted Steam Line Break Leakage

Steam line break leakage as a function of end-of-cycle voltage will be determined in accordance with draft NUREG-1477 dated June 1, 1993, Section 3.3, "Recommended Method for Calculating Potential Leakage."

D. Steam Line Break Primary-to-Secondary Leakage Limit

Predicted steam line break leakage will be limited to 22.8 gpm. The limit of 22.8 gpm is derived as the product of 5.7 gpm x 4 where:

•5.7 gpm is the leakage limit based on a Standard Review Plan analysis submitted to the NRC by Southern Nuclear's June 4, 1992 letter; and

•4 is the multiplication factor on the allowable increase in leak rate justified by reducing the limits for specific activity of the primary coolant contained in Farley Technical Specification 3.4.9. by a factor of 4. The specific activity dose equivalent I^{131} will be reduced from 1 μCi/gm to 0.25 μCi/gm, and the 100% transient dose equivalent I^{131} will be reduced from 60 μCi/gm to 15 μCi/gm.

This methodology is described in draft NUREG-1477 dated June 1, 1993.

E. Normal Operating Primary-to-Secondary Leakage Limit

The normal operating primary-to-secondary leakage limit contained in the Unit 1 technical specifications has been reduced to 140 gpd as the result of non-tube support plate related corrosion. This limit is more conservative than the 150 gpd limit previously associated with the interim plugging criteria.

II. ADDITIONAL STEAM GENERATOR TUBE PULLS

A total of 5 steam generator tube intersections from Farley Unit 1 and 13 from Farley Unit 2 have been pulled and evaluated since 1985. Included in the alternate repair criteria databases are 3 intersections from Farley Unit 1 and 6 intersections from Farley Unit 2. Two intersections from Unit 1 with relatively high voltages, i.e., 3.3 and 3.2 volts, were removed in 1992 and destructively examined to support Farley interim plugging criteria/alternate repair criteria applications. The existing tube pull results substantially confirm that the dominant flaw feature affecting the integrity at Farley is outside diameter stress corrosion cracking. Therefore, Farley does not intend to pull additional tubes for the next Unit 1 cycle.

III. USE OF SMALLER BOBBIN PROBES

Repair of tubes by sleeving reduces the tube inside diameter resulting in reducing the probe size (diameter) needed to inspect tube sections above or between the sleeves. For the few instances in which a portion of the tube can not be inspected using the normal 0.720" probe, satisfactory inspection can be accomplished using a smaller probe whose diameter will pass through the sleeve and whose design allows it to be centered in the unsleeved portion of the tubes. For 7/8" tubes with 3/4", 0.040" thickness sleeves, the nominal 0.720" probe is replaced with a 0.640" probe with centering devices which provide stability in the 0.775" ID of the unrepaired portion of the tube.

To overcome the reduced fill factor associated with smaller probes, the gain factor applied to the data is increased to provide a calibration equivalent to that obtained from the nominal probe size with the ASME standards. Specifically, the 20% holes are set to yield 4.0 volts at the prime inspection frequency and 2.75 volts in the tube support plate suppression mix. The wear standard is employed to confirm satisfactory centering of the smaller probes.

Recent inspections at one plant and at Farley Unit 1 provided experience with the use of reduced diameter probes centered for 7/8" tubing. Forty-six tube support plate intersections from the first plant were examined with both 0.720" and either 0.560" or 0.580" diameter probes. The normalized amplitudes obtained were compared to determine whether non-conservative results might be obtained due to the reduced sensitivity associated with lower fill factor probes. In all cases, the signal observed with the 0.720" probe was detected with the smaller probe. Only 4 of 46 signals had less than 100% of the 0.720" probe amplitude:

0.720" Voltage	0.580"/0.560" Voltage	Percentage
1.44	1.43	99%
1.35	1.24	92%
1.13	1.03	91%
1.47	1.37	93%

Forty-two of forty-six inspections using the smaller probe resulted in equal or higher voltage calls for the smaller probes.

At Farley Unit 1, an additional nineteen intersections were similarly compared using 0.720" and 0.640" probes. In the Farley test, three of nineteen signals observed with the 0.640" probe exhibited less than 100% of the 0.720" probe voltage amplitudes:

0.720" Voltage	0.640" Voltage	Percentage
0.59	0.51	86%
0.53	0.20	38%
0.90	0.26	29%

Of the three signals in which lower voltages were called, all of the 0.720" voltages were less than 1.0 volt. Consequently, no changes in the tubes repair requirements would have resulted from using a 0.640" probe for the inspection. Of the six intersections with 0.720" probe voltages at least 0.99 volts or greater, all were called with higher voltages using the 0.640" probe. In summary, at Farley Unit 1, all nineteen specimens produced measurable signals with both the 0.720" and 0.640" probes; and, in none of the specimens was a signal greater than 1.0 volt with the 0.720" probe undersized with the 0.640" probe.

By letter ET-NRC-93-3863 dated April 14, 1993, Westinghouse forwarded WCAP-13692 to the NRC Staff. This WCAP provides documentation which is the basis for the discussion on the use of smaller probes.

Based on the testing discussed above, Southern Nuclear believes the use of reduced diameter bobbin probes with centering devices for inspection of intersections which are inaccessible with the standard diameter probe is conservative and appropriate. In fact, based on the testing discussed above, use of the smaller diameter probe will result in more conservative calls, i.e., higher voltages, for significant flaws, i.e., greater than 1 volt, than use of the standard diameter probe and none of the test cases would have resulted in leaving a tube in service which should have been repaired.

Conclusion

Use of a 2.0 volt interim plugging criteria at Farley Unit 1 provides sufficient margin to ensure tube burst will not occur during a steam line break. Burst during normal operations is precluded by the constraint of the tube support plate. Furthermore, leakage at steam line break differential pressures has not been observed at the 2.0 volt level. However, even if

leakage did occur, usage of the NRC leak rate model ensures that overly conservative leak rates will be predicted. This will ensure that the doses at the site boundary in the event of a steam line break will be bounded by those currently assumed in the Farley Technical Specifications. Additionally, use of the smaller diameter bobbin probe, when calibrated properly, has been shown to not result in missing or undercalling voltages for significant flaws.

Consequently, use of a 2.0 volt interim plugging criteria, coupled with the use of smaller diameter probes on an as-needed basis, will ensure that steam generator tube ruptures at tube support plates will not occur during a steam line break and that the doses at the site boundary will not exceed those currently assumed in the Farley Technical Specifications.

Attachment 2

Voltage Repair Criteria

for the

Steam Generator Tube Support Plate

Interim Repair Criteria

Southern Nuclear Operating Company Proposal:

Voltage Repair Criteria

1. A bobbin inspection of 100% of the hot and cold leg steam generator tube support plate intersections will be performed.

2. Flaws 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.

3. Flaws 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. Flaw indications 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 probe (RPC) inspection does not detect a flaw. Flaw indications with a bobbin voltage greater than 5.6 volts will be plugged or repaired.

5. All flaw indications with bobbin voltages greater than 1.5 volts and less than 2.0 volts will be inspected by RPC.

6. A sample RPC inspection 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 volts. 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.

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

End-of-Cycle (EOC) Voltage Distribution

Acceptable methods for determining the end-of-cycle voltage distribution are as follows:

1. The methodology described in WCAP-12871, Revision 2. This involves sampling the cumulative probability distribution of NDE uncertainty and the voltage growth rate using Monte Carlo techniques and applying the results to the beginning-of-cycle (BOC) voltage distribution.

2. A simplified approach may be used as an alternative (to the WCAP-12871, Revision 2 approach) provided it allows for a conservative treatment of the tails of the cumulative probability distributions of NDE uncertainty and the voltage growth rate to the 100% cumulative probability values.