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Southern Nuclear Operating Company
the southern electric system

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Docket No. 50-364

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

Joseph M. Farley Nuclear Plant - Unit 2
Steam Generator Tube Support Plate Voltage-Based Repair Criteria

Ladies and Gentlemen:

On August 3, 1995, Generic Letter 95-05, Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking, was issued. Farley Unit 2 currently has an interim tube repair criteria that expires at the end of the current operating cycle. As a result, Southern Nuclear is submitting a Technical Specification amendment for Unit 2 reflecting the guidance contained in the Generic Letter. Although some clarifications to the Generic Letter requirements are proposed, the Technical Specification requested continues to fulfill all safety requirements for a voltage-based steam generator tube repair criteria. NRC approval of the Farley voltage-based criteria is requested by September 1, 1996, based on a Unit 2 outage start date of October 12, 1996.

The safety analyses to support this amendment have been previously docketed. These analyses include:

1. WCAP-12871, Revision 2, J. M. Farley Units 1 and 2 Steam Generator Tube Plugging Criteria for ODSCC at Tube Support Plates, February 1992;
2. EPRI Report TR-100407, Revision 1, PWR Steam Generator Tube Repair Limits - Technical Support Document of Outside Diameter Stress Corrosion Cracking at Tube Support Plates; and
3. Southern Nuclear to NRC letter dated December 9, 1993, and associated Technical Specification amendment and NRC safety evaluation dated April 5, 1994.

Additional analyses exist in Generic Letter 95-05.

Attachment 1 contains responses to the Generic Letter. Attachment 2 contains the proposed changed Technical Specification pages in support of the voltage-based plugging criteria. Attachment 3 provides a significant hazards evaluation for the proposed voltage-based repair criteria in accordance with 10 CFR 50.92.

Southern Nuclear Operating Company has performed an assessment of the impact of the proposed revision to the Technical Specifications on the environment and has determined that there is no impact on the environment. The proposed revision does not affect the types or amounts of any radiological or non-radiological effluents that may be released offsite. No increase in individual or

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cumulative occupational radiation exposures will result from this revision. Additionally, the revision does not involve the use of any resources not previously considered in the Final Environmental Statement related to the operation of Farley Nuclear Plant.

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

DM Morey

Dave Morey

REM/NRCARCLT.DOC

Attachments

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

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 29th DAY OF March 1996

Martha Gayle Dow
Notary Public

My Commission Expires: November 1, 1997

Attachment 1

Responses to Generic Letter 95-05 Guidance

Responses to Generic Letter 95-05 Guidance

Southern Nuclear (SNC) will implement the requested actions of Generic Letter 95-05 with the following comments:

(1) The applicability requirements discussed in Section 1 of Attachment 1 of Generic Letter 95-05 will be implemented.

1.b.1 - Concerning the deformation or collapse of steam generator tubes following a loss of coolant accident plus a safe shutdown earthquake event, a Farley specific analysis was docketed under WCAP-12871, Revision 2 dated February 1992. As a result of this analysis, no tubes will be excluded from using the voltage repair criteria.

(2) The inspection guidance discussed in Section 3 of Attachment 1 of the Generic Letter will be implemented in accordance with the Appendix A guidelines last submitted to the NRC by letter dated February 23, 1994 and with the following responses:

3.b- SNC will utilize a motorized rotating coil probe, e.g., pancake or +Point, instead of specifying a rotating pancake coil. This wording change is made to ensure that the +Point probe can be used as an alternative to the rotating pancake coil.

3.b.1 - SNC will inspect all bobbin flaw indications with voltages greater than 2.0 volts with a motorized rotating coil probe.

3.b.2 - SNC will inspect all intersections where copper signals interfere with the detection of flaws with a motorized rotating coil probe.

3.b.3 - SNC will inspect all intersections with dent signals greater than 5 volts with a motorized rotating coil probe.

3.b.4 - SNC will inspect all intersections with large mixed residuals that could be expected to mask a 1.0 volt bobbin flaw signal with a motorized rotating coil probe.

3.c.2 - The ±10% limit on new probe variability will be implemented using the guidance included in the Nuclear Energy Institute to NRC letter dated January 23, 1996, concerning "New Probe Variability for Use in the ODSCC Alternate Repair Criteria," as discussed in the NRC to the Nuclear Energy Institute letter dated February 9, 1996.

3.c.3 - The limits on probe wear will be implemented using the guidance included in the Nuclear Energy Institute to NRC letter dated January 23, 1996, concerning "Eddy Current Probe Replacement Criteria for Use in ODSCC Alternate Repair Criteria," as discussed in the NRC to the Nuclear Energy Institute letter dated February 9, 1996. The following summarizes this guidance as agreed to by the NRC Staff:

- For all tubes identified with indications above 1.5 volts (i.e., 75% of the 2 volt repair limit for 7/8 inch tubes is 1.5 volts) since the last successful probe wear check (< 15% wear), the whole tube, i.e., all hot-leg tube support plate

intersections to the lowest cold-leg TSP intersection with known ODSCC, will be re-inspected with an acceptable probe (<15% probe wear) and all eddy current data from the acceptable probe will be evaluated. If a large indication (greater than approximately 1 volt for 7/8 inch tubes) is detected which was previously missed with the failed probe, an assessment of the significance will be performed during the outage. This assessment, along with the description of actions taken, will be provided to the NRC in the 90-day report.

The inspection described above will be modified slightly for tubes which would require a double entry to inspect the entire tube. For low row tubes in which the U-bend radius precludes passing a full size bobbin coil over the U-bend or for tubes with sleeves which preclude passing a full size bobbin through the sleeve, the portion of the tube with the indication above 75% of the repair limit will be re-inspected. The second entry for inspection of the remainder of the tube is not required provided there is not an indication above 75% of the repair limit.

- Actions will be taken to minimize the potential for tubes to be inspected with probes that fail the probe wear check. This includes replacing a probe immediately upon finding that it fails the probe wear check.
- If a probe fails prior to performing a probe wear check, it will be assumed that the probe failed the probe wear check and the probe wear criteria approved by the NRC Staff will be followed.
- The effects of probe wear will be explicitly assessed as a potential contributing factor if significant differences between the actual and end-of-cycle projections exist in the 90-day report.
- The 90-day report will address if a non-proportionate number of new indications have been detected in tubes which were inspected in the previous outage with a probe that failed the probe wear check.

3.c.5 - Quantitative noise criteria have been and will continue to be used in data collection. Data analysts will use qualitative guidelines in the evaluation of the data. However, it is expected that these criteria will be evolving over future inspections and, as a result, are subject to change.

3.c.8 - Data analysts will be trained on the potential for primary water stress corrosion cracking to occur at tube support plate intersections. The discovery of PWSCC at tube support plate intersections will be reported to the NRC Staff prior to startup.

(3) Calculations of the main steam line break leakage will be per the guidance of Section 2.b of Attachment 1 of Generic Letter 95-05 with the following responses:

2.b - Calculations performed in support of the voltage-based repair criteria will follow the methodology described in WCAP-14277, SLB Leak Rate and Tube Burst Probability Analysis Methods for ODSCC at TSP Intersections, January 1995.

2.b.2(1) - No distribution cutoff will be applied to the voltage measurement variability distribution.

2.b.4 - In order to preclude the possible need for rapid turn around of a Technical Specification amendment for reactor coolant system specific iodine activity, the Farley Technical Specification will remain at 0.5 μCi/gram. The current leakage limit for Farley Unit 2 is 11.4 gpm based on the April 7, 1995, NRC letter approving the 2 volt interim repair criteria for Unit 2.

2.c - Reference is made to the use of an RPC probe. SNC will utilize a motorized rotating coil probe, e.g., pancake or +Point, instead of specifying a rotating pancake coil.

(4) Calculation of the conditional burst probability will be per the guidance of Section 2.a of Attachment 1 of Generic Letter 95-05 with the following responses:

2.a - Calculations performed in support of the voltage-based repair criteria will follow the methodology described in WCAP-14277, SLB Leak Rate and Tube Burst Probability Analysis Methods for ODSCC at TSP Intersections, January 1995.

2.a.2 - The upper voltage repair limit will be determined 2 months prior to the outage using the most recently approved NRC data base.

(5) Farley leakage monitoring measures provide guidance on trending and response to rapidly increasing leaks. Guidance is provided not only for the absolute leakage measured, but also on the rate of change of the leak rate. Timely detection of leaks is ensured by the N-16 monitors on both units. Farley has also implemented the current industry guidelines.

(6) Tube pull guidance of Section 4 of Attachment 1 of Generic Letter 95-05 will be followed.

(7) Results will be reported per the guidance of Section 6 of Attachment 1 of Generic Letter 95-05.