Revised **Responses to Generic Letter 95-05 Guidance**

Southern Nuclear Operating Company (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:~~

The inspection criteria discussed in Section 3 of Attachment 1 of the Generic Letter will be implemented with the following responses/clarifications. In addition, the inspection guidance will be implemented in accordance with the Appendix A guidelines last submitted to the NRC by letter dated February 23, 1994.

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. Any indication found with the motorized rotating coil will result in repair of the tube.

3.b.3 - All intersections with dent signals greater than 5 volts will be inspected with a motorized rotating coil probe. Any indications found at such intersections with the motorized rotating coil probe will result in repair of the tube. If circumferential cracking or primary water stress corrosion cracking indications are detected, the motorized rotating coil probe sampling plan may be expanded to include dents less than 5 volts.

~~SNC will inspect all intersections with dent signals greater than 5 volts with a motorized rotating coil probe. If circumferential cracking or primary water stress corrosion cracking is detected at the tube support plates intersections, a sampling~~

~~plan will be implemented in accordance with the PWR Steam Generator Tube Examination Guidelines, Revision 4. If indications are found at dents with voltages near 5 volts, the flaw will be characterized. If the flaw exceeds the structural requirement of Regulatory Guide 1.121, the sampling plan will be expanded to intersections with dents less than 5 volts. If the flaw is evaluated as not significant, the sampling plan will not be expanded.~~

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 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. Furthermore, SNC will verify that both the primary and mix frequencies will meet the ±10% variability requirement.

3.c.3 - The limits on probe wear will be implemented using the guidance included in Nuclear Energy 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; NEI to NRC letter dated February 23, 1996; and NRC to NEI letter dated March 18, 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 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.4 - Data analysts will be trained and qualified in the use of the analyst's guidelines and procedures. At Farley Nuclear Plant, a minimal number of analysts are used for determination of voltage. The use of a small number of analysts is intended to minimize the effect of analyst variability on determination of growth rate, resulting in as accurate a prediction for the next operating cycle as possible. We believe this results in a more accurate growth rate determination; however, it is time consuming and can result in difficulty in performing the calculations prior to returning the steam generators to service.

3.c.5 - Quantitative noise criteria have historically been applied and will be incorporated in the Farley Nuclear Plant Data Acquisition procedures. This enables noise levels due to electrical noise, tube noise, calibration standard noise, etc., to be addressed at the initial point of inspection which has minimized the need for re-inspection. Probes are typically replaced prior to 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 re-inspected. In addition, the Farley Nuclear Plant Analysis procedures allow the analyst to require re-inspection due to noise on a "qualitative" basis.

3.c.6 - Data analysts will review the mixed residuals on the standard itself and take actions as necessary to minimize these residuals.

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.

(2) Calculations of the main steam line break leakage will be per the guidance of Sections 2.b and 2.c 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 probabilistic methodology described in WCAP-14277, Revision 1, 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. A leakage limit for Farley Unit 1 and 2 of 20 gpm is justified in Attachment 4.

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. It is SNC's intent (and desire) to always perform the calculations on the projected EOC distributions. In the event that the growth rate determinations cannot be completed prior to returning the steam generators to service, the calculations will be based on the actual EOC distributions as allowed in Section 2.c. However, even if the calculation made prior to returning the steam generators to service is based on the actual measured voltage distribution, the calculation based on the projected EOC voltage distribution will be provided to the NRC in the 90 day report following the outage.

(3) 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, Revision 1, 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 database. The database proposed by NEI letter dated September 18, 1996, will be used for the Unit 1 outage. Following this outage, the database used will be based on the NRC/industry protocol.~~

Southern Nuclear will use the database forwarded to the NRC Staff by Duquesne Light Company letter dated March 27, 1996, for the upcoming Farley Unit 1 inspection/evaluation. This is the same database that was used on the Farley Unit 2 inspection/evaluation. Southern Nuclear also requests that the NRC Staff provide a projected schedule of review of the NEI database to allow utilities to make plans for upcoming outages.

(4) 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 guidelines contained in EPRI topical report "PWR Primary-to-Secondary Leak Guidelines," EPRI TR-104788, May 1995.

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

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

ENCLOSURE 3

Revised Technical Specification Page

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 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 is contained in GL 95-05.

The mid-cycle equation in 4.4.6.4.a.11.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 in 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: