April 5, 2001

Mr. William T. Cottle
President and Chief Executive Officer
STP Nuclear Operating Company
South Texas Project Electric
Generating Station
P. O. Box 289
Wadsworth, TX 77483

SUBJECT: PRELIMINARY RESULTS OF STEAM GENERATOR TUBE INSPECTIONS FOR SOUTH TEXAS PROJECT UNIT 2 END-OF-CYCLE 8 REFUELING OUTAGE

Dear Mr. Cottle:

During a series of conference calls on March 19, 26, and 27, 2001; South Texas Project Nuclear Operating Company (the licensee) briefed the U. S. Nuclear Regulatory Commission (NRC) staff on preliminary results of steam generator (SG) tube inspections during refueling outage 8, for South Texas Project (STP), Unit 2. As a result of those conversations, the licensee's staff and management agreed to meet with the NRC staff and management, to discuss the implications of the results of the inspections with regards to the STP, Unit 2, operations for Cycle 9. In this letter the NRC outlines the background and scope of the requested meeting.

In a telephone call on March 19, 2001, the licensee discussed with the NRC staff, preliminary results of SG tube inspections for refueling outage 8 at STP, Unit 2. The licensee reported that at the end-of-Cycle 8, the operating primary-to-secondary leak rates for SGs 2A, 2B, 2C, and 2D were 10.5, 7.5, 8, and 9 gallons per day (gpd), respectively. After shutting down for the refueling outage, the licensee performed a secondary side pressure test and identified about 100 leaking tubes. Based on eddy current inspection of the SG tubes, the licensee attributed the operating leakage to several large voltage eddy current indications located at the tube support plate (TSP) intersections which were characterized as outside diameter stress corrosion cracking (ODSCC). The licensee implemented the 1-volt alternate repair criteria for ODSCC indications in accordance with the NRC's Generic Letter (GL) 95-05. While, the NRC staff approved (on March 8, 2001) 3-volt repair criteria for three TSPs (C, F, and J); the licensee, as a result of the unexpected large growth rates and large number of ODSCC indications, conservatively modified the approved 3-volt repair criteria, by plugging tubes with ODSCC indications greater than 1.5 volts, instead of 3.0 volts.

The licensee indicated that, as of March 26, 2001, it completed in-situ pressure testing of 6 tubes with ODSCC indications (2 from SG 2B and 4 from SG 2D). Each of the 6 tubes leaked under main steamline break conditions. The test leak rates from SG 2B and SG 2D tubes totaled 1.27 gallons per minute (gpm) and 1.01 gpm, respectively. The licensee stated that all 6 tubes maintained a margin of 3 under the normal operating differential pressure for burst testing. Using the GL 95-05 methodology and actual end of Cycle 8 indications, the licensee calculated an accident-induced leak rate of 2.8 gpm in SG 2D, which is the limiting SG. The licensing basis allowable for the accident-induced leakage is 15.4 gpm.

Based on the licensee's above stated inspection results, the NRC staff has the following concerns regarding the integrity of the STP, Unit 2, SG tubes: (1) tubes in the STP, Unit 2, SGs have experienced unexpectedly high degradation growth rates; the NRC staff needs to understand how the licensee assessed the root cause of the high growth rates and the impact of the high growth rate trend on the ODSCC indications left in service for Cycle 9; (2) the scope of in-situ pressure test samples as related to the Electrical Power Research Institute report, “Steam Generator In Situ Pressure Test Guidelines: Revision 1,” April 1999; Appendix B (tables B-2, B-4, B-6, and B-8) of the guidelines, which recommend that all known leaking tubes be tested; and (3) at the end of Cycle 8 the operating leak rate for SG 2D of 9 gpd may correspond to the steamline break accident leakage of 2.8 gpm; if operating leakage occurs and escalates during Cycle 9 operation, the accident-induced leakage may exceed the licensing basis assumption of 15.4 gpm; it would also indicate that preventive plugging measures taken in response to the high growth rates were not effective. In view of potential operating leakage in Cycle 9, the NRC staff requests that the licensee discuss the need for operational leakage restrictions for Cycle 9 operation.

The NRC staff suggests that the meeting be held at the NRC headquarters during the week of April 16, 2001, to discuss the above issues. If there are any questions, please contact me at (301) 415-1476.

Sincerely,

/RA/

Mohan C. Thadani, Senior Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-499

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