UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

August 6, 1990

NRC INFORMATION NOTICE NO. 90-49: STRESS CORROSION CRACKING IN PWR STEAM GENERATOR TUBES

Addressees:

All holders of operating licenses or construction permits for pressurized-water reactors (PWRs).

Purpose:

This information notice is intended to inform licensees of recent problems involving stress corrosion cracking (SCC) in PWR steam generator (SG) tubes. In particular, this information notice is intended to alert licensees to recent findings at Millstone Unit 2 and to recent problems in SCC detection during inservice inspections. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

1. Circumferential Cracking at Millstone Unit 2

In October 1989, the licensee for Millstone Unit 2 conducted a mid-cycle inspection of the SG tubing using eddy current testing (ECT). Circumferential SCC had been observed in previous inspections to be affecting the outer diameter (OD) surface (that is, the secondary side) of the tubes at the expansion transition at the top of the tubesheet. The mid-cycle inspection followed a previous inspection during the February 1989 refueling outage and was performed, in part, out of concern for the relatively high rate of SCC growth observed during the previous inspection and to ensure that SCC did not excessively degrade the integrity of the tubes. Just before the mid-cycle outage, leakage from the primary side to the secondary side was less than 5 gallons per day (gpd). The plant's Technical Specifications limit for such leakage is 144 gpd.

The ECT inspections at Millstone 2 were conducted with a rotating pancake coil (RPC) probe to ensure optimal sensitivity to circumferential cracks. Tubes found with circumferential crack indications were also inspected by ultrasonic testing (UT) to obtain additional information regarding the length and depth of the cracks. In addition, two tubes with circumferential crack indications were removed and examined.

9007310195

D18

The ECT/RPC inspections revealed 104 tubes with circumferential cracks at the expansion transition. The macrocracks, as defined by ECT/RPC, consisted of several discontinuous microcracks that were separated by small ligaments of sound material. The discontinuous nature of the array of microcracks was confirmed by the UT and examination of the removed tube specimens. As measured by UT, the macrocracks ranged in circumference from 84 degrees to 329 degrees and ranged in depth up to 100-percent throughwall.

All tubes with crack indications were staked and plugged. In addition, the licensee evaluated the residual strength of the cracked tubes to assess their capability to sustain normal operating and postulated accident loadings before their removal from service. This structural evaluation considered the profiles for each crack obtained from the UT examination. This evaluation revealed one cracked tube which failed to meet the ASME Code, Section III, NB-3225 and Appendix F stress limits for postulated accident conditions. (Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," states that margins should be consistent with the stress limits in Section III of the code.) Based on these findings, the staff concludes that the integrity of the subject tube was not ensured under postulated accident conditions.

The staff has recently identified service induced, circumferential SCC, such as at Millstone Unit 2, to be a source of significant degradation to tubes in PWR steam generators. Such cracking is particularly noteworthy because it is generally not detectable with conventional bobbin probes used routinely for inservice inspection. Such cracking is generally only detectable through the use of specialized probes, such as the RPC probe.

Most circumferential cracking has been observed at tube expansion transitions at or near the top of the tubesheet. In addition to Millstone Unit 2, circumferential cracking at the expansion transition has recently been identified at one other Combustion Engineering (CE) plant (Maine Yankee), at three plants with Westinghouse Model 51 steam generators (North Anna Unit 1, Trojan Unit 1, and Sequoyah Unit 1), and at one plant with Westinghouse Model D steam generators (McGuire Unit 1). Tubes in the affected CE and Westinghouse Model 51 steam generators were explosively expanded against the tubesheet. Tubes in the McGuire Model D steam generators were expanded against the tubesheet by mechanical rolling. In addition to being found at the expansion transition location, widespread circumferential SCC has been observed at drilled-hole support plate locations at Palisades (CE steam generators). Isolated instances of circumferential SCC have been reported at the uppermost support plate of a pre-replacement Westinghouse Model 44 steam generator of Indian Point Unit 3 and at a row 1 U-bend of a Model 51 steam generator at Zion Unit 1. The circumferential SCC at Palisades and Indian Point Unit 3 appears to be associated with significant denting at the support plates.

2. Axial Cracking at Support Plates

Licensees have reported secondary side-initiated, axial SCC at several plants with Westinghouse Model 51 and Model D steam generators at support plate intersections exhibiting little or no denting. Recent difficulties experienced in the detection of such cracks were the subject of Westinghouse Customer Information Letter GEN-LTR-90-006, "Steam Generator Tube Outer Diameter Stress Corrosion Cracking at Tube Support Elevation-Eddy Current Detection Issue," which was issued on or about February 8, 1990, to all utilities with Westinghouse steam generators. Westinghouse reported that metallographic examinations of tubes removed from the field have revealed the presence of OD-initiated SCC at tube support plate (TSP) intersections that were not reported by personnel using a bobbin probe to perform field eddy current tests. For example, these examinations revealed one tube containing axial cracks within two 30-degree-wide bands on opposite sides of the tube, with the deepest crack penetrating to 62-percent throughwall. The field eddy current interpretation of the signal for this TSP location was "no detectable degradation" (NDD) using the plant voltage threshold criteria.

The EPRI "Steam Generator Examination Guidelines, Revision 2" contains Figure C-55 showing the qualitative relationship between bobbin probe signal amplitude and crack depth determined metallographically. The staff believes that the amplitude threshold criteria used at the plant in the above-mentioned example were taken from the actual data used to develop Figure C-55. The recent evidence cited by Westinghouse suggests that this data is not conservative for all plants.

EXHIBIT
revised
EDDY Currc
TESTING
1990

Industry meetings attended by representatives from a number of vendors providing SG inspection services, EPRI, and the Westinghouse Owners Group have been conducted to examine various proposals to detect and measure SCC at TSP locations. The minutes of the EPRI Guidelines Revision Committee meeting on February 13, 1990, note that general principles still apply, pending development of updated guidance, for the detection and measurement of SCC at support plates. Section 4.6.1 of the EPRI guidelines states that as a general rule, an "extremely conservative position" should be adopted for the resolution of distorted indications or undefined signals not covered by existing analysis guidelines. Specifically, tubes with these types of indications should be recommended for plugging unless other supporting data exists (tube pulls or NDE diagnostic data) that justifies their retention as active tubes.

Discussion:

The reliable detection and sizing of SCC during inservice inspections pose a significant challenge to current ECT technology and practice in view of the low signal-to-noise ratios frequently exhibited by such cracks. Experience

indicates that SCC is frequently not detected until it has penetrated beyond 40-percent throughwall. Fortunately, the vast majority of SCC flaws consist of short axial or circumferential crack segments. The staff believes that such flaws can be detected before they grow sufficiently large to degrade the structural margins of the tube to below the Regulatory Guide 1.121 criteria.

Tubes are generally inspected once per refueling cycle. Depending on the rate of crack growth and the number of tubes involved, this frequency may or may not be sufficient to ensure that all cracks are detected before they become sufficiently large to degrade structural margins to less than the Regulatory Guide 1.121 criteria. A structural assessment of the crack geometries found during an inspection, such as performed at Millstone Unit 2, provides a means for assessing whether the inspection frequency is sufficient to ensure adequate structural margins for all tubes between inspections.

The staff believes that the effectiveness of eddy current testing for detecting and sizing SCC can be enhanced through improved criteria for the qualification and performance demonstration of the eddy current data acquisition equipment (including probes), data analysis procedures, and data analysts. The staff and the industry, including the EPRI Steam Generator Reliability Project, are evaluating this issue. In the meantime, field experience indicates that careful attention to the potential for SCC at the tubesheet region, at tube expansion transitions, at support plates, and at tube U-bends is important. The recommendations in Section 4.6.1 of the EPRI steam generator examination guidelines indicate the importance of being alert to distorted and undefined signals at these locations and employing diagnostic measures as appropriate to establish the cause of these signals and to validate the plant data analysis procedures and criteria.

Finally, the findings at Millstone Unit 2 illustrate that cracks will not necessarily cause leakage approaching the Technical Specifications limit before the structural margin in the affected tube drops below the Regulatory Guide 1.121 criteria for postulated accident conditions. This point is further illustrated by the steam generator tube rupture (SGTR) event that occurred at McGuire Unit 1 on March 7, 1989, as a result of axially oriented SCC. Leakage before the SGTR event was about 15 gpd, which was small compared to the plant's Technical Specifications limit of 500 gpd. The McGuire event occurred under normal operating conditions. Thus, the McGuire event was preceded by a period during which the subject tube was vulnerable to rupture if challenged by a postulated accident.

This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact the technical contact listed below or the appropriate NRR project manager.

Charles E. Rossi, Director
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

Technical Contact: E. Murphy, NRR
(301) 492-0710

Attachment: List of Recently Issued NRC Information Notices
ENDEND