UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

September 20, 1995

NRC INFORMATION NOTICE 95-40: SUPPLEMENTAL INFORMATION TO GENERIC LETTER 95-03, "CIRCUMFERENTIAL CRACKING OF STEAM GENERATOR TUBES"

Addressees

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to provide additional information on steam generator tube examination results from Maine Yankee Atomic Power Station as previously discussed in Generic Letter (GL) 95-03, "Circumferential Cracking of Steam Generator Tubes." 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 are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

The staff issued GL 95-03, to obtain information necessary to assess compliance with requirements regarding steam generator tube integrity in light of the inspection findings at the Maine Yankee plant. In GL 95-03, the staff requested that utilities (1) evaluate recent operating experience with respect to the detection and sizing of circumferential indications, (2) develop a safety assessment justifying continued operation until the next scheduled steam generator tube inspections are performed, and (3) develop plans for the next inspections of steam generator tubes as they pertain to the detection of circumferential cracking. Since the issuance of GL 95-03, additional information pertaining to in situ pressure testing and destructive analysis for the tubes removed from the Maine Yankee plant has become available. In addition, the wrong title given to NUREG-0844 in GL 95-03 was erroneously indicated as, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes." The correct title is, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity."

Discussion

On July 15, 1994, Maine Yankee Atomic Power Company, the licensee for Maine Yankee, shut down the plant when the measured primary-to-secondary leak rate approached 189 liters [50 gallons] per day. After shutting down the plant, the licensee tested for leaks and found four leaking tubes. IN 94-88, "Inservice Inspection Deficiencies Result in Severely Degraded Steam Generator Tubes," discusses in situ pressure testing performed by the licensee in 1994, on tubes containing some of the largest indications, to assess their actual burst integrity. At that time, certain tubes could not be pressurized due to a combination of leakage and pump capacity limitations, and the staff had not reached a conclusion regarding the validity of the tests to simulate an actual pressure transient in the steam generators.

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In 1995, the licensee performed additional steam generator inspections. Seven tubes were subjected to in situ pressure testing, three of which were from the sample subjected to in situ pressure testing in 1994 and

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four of which were tubes containing some of the largest indications identified at the end of the 1994-to-1995 operating interval. The testing indicated that the tubes were capable of withstanding pressure loadings in excess of the loads for which failure would be predicted on the basis of the size estimates with the standard pancake coil. Furthermore, the pressures to which the tubes were subjected were greater than design-basis loads. NRC Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," indicates that tubes should be able to withstand "3 times operating pressure" and "1.4 times main steam line break maximum pressure" without bursting. At Maine Yankee, 3 times operating pressure is approximately equal to 34.47 MPa [5000 psi] and 1.4 times main steam line break maximum pressure equals 27.97 MPa [4057 psi]. All tested tubes at Maine Yankee were subjected to at least 39.30 MPa [5700 psi] hydrostatic pressure. Three tubes exhibited no defect leakage and no tubes burst. The staff has concluded that these tests adequately bound main steam line break loads on steam generator tubes.

As stated in GL 95-03, three tubes were removed from the Maine Yankee steam generators for destructive examination: two tubes with marginal plus-point coil responses (sized by the eddy current analysts as probably less than 40 percent through-wall depth) and one with an intermediate response (sized by the eddy current analysts as probably greater than 40 percent through-wall depth). Before the tubes were removed, they were examined with several nondestructive methods, such as ultrasonic, fluorescent penetrant, and eddy current techniques to confirm the nature of the indications. The eddy current methods included examination with a standard rotating pancake coil, a plus-point coil, and a high-frequency pancake coil. The indications were sized with various techniques. The size estimates for the high-frequency pancake coil and the plus-point coil were obtained after calibration of the probes on electric discharge-machined (EDM) notches contained within a standard. With the high-frequency pancake coil, the most sensitive of the coils to the degradation at Maine Yankee, the indications on the pulled tubes were sized with maximum through-wall depths of 36, 32, and 44 percent, and average depths of 30, 21, and 27 percent, respectively. The average depth estimates obtained from the eddy current examination are calculated from the maximum depth and the circumferential extent by assuming that the maximum depth is the depth of the degradation over the entire measured circumferential arc length and averaging this estimate over the entire tube circumference. The corresponding destructive examination results for these tubes indicated that the maximum depths were 45, 37, and 57 percent, with average depths of 24, 23, and 26 percent, respectively. The destructive examination of these tubes indicated that numerous small cracks had initiated at various locations about the circumference and at various elevations (axial locations) within a 1.27 mm [0.05 inch] band in the "expansion" transition region of the tubes, noncorroded ligaments existed between some of the cracks. The cracks initiated at the inner diameter of the tubes. The licensee compared the sizing of several of the larger indications that were inspected with both a standard pancake coil and the high-frequency pancake coil. The high-frequency pancake coil is, in general, more sensitive than the standard pancake coil to cracks initiating at the inner diameter. The results of this comparison indicated that the maximum and average depths estimated by the high-frequency pancake coil were consistently lower than the maximum and average depths estimated with the standard pancake coil even though the length (i.e., circumferential extent) estimates were longer with the high-frequency coil.

The smaller depth estimates obtained with the high-frequency coil suggest that many of the indications may not have been as structurally significant as the standard pancake coil suggested and as was reported in IN 94-88. Furthermore, the destructive examination indicated that the cracks were not coplanar, but rather of short circumferential length and staggered over a short axial region. There were, in fact, ligaments of material between the cracks. Due to the nature of this cracking (i.e., the spacing between the cracks), the ligaments of sound material could not be distinguished by the nondestructive examination (i.e., standard and high-frequency pancake coil and plus-point coil) data; however, the nondestructive examination data are conservative in that the tubes are most likely more structurally sound than estimated by the eddy current examination. The observed segmented character of these cracks is consistent with the results of fluorescent penetrant examination results at Maine Yankee and with the morphology of circumferential cracks observed on specimens of tubes pulled from other plants.

This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

/s/'d by DMCrutchfield

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