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

December 7, 1992

NRC INFORMATION NOTICE 92-80: OPERATION WITH STEAM GENERATOR TUBES SERIOUSLY DEGRADED

Addressees

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

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to inform licensees of recent findings from steam generator (SG) tube inspections and investigations at Arkansas Nuclear One Unit 2 (ANO-2). The Arkansas Power and Light Company, the licensee for ANO-2, found three tubes to be degraded to the point where they no longer retained adequate structural margins to sustain the full range of normal operating, transient, and postulated accident conditions without rupture. 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

In 1978, the NRC licensed ANO-2 for operation. ANO-2 is a two-loop PWR designed by Combustion Engineering, Incorporated (CE). On March 9, 1992, the licensee shut down ANO-2 upon detecting a primary-to-secondary leak of 0.95 liters per minute [0.25 gallons per minute]; half of the technical specification limit. The licensee conducted an eddy current inspection of the SG tubes using a motorized rotating pancake coil (MRPC) probe and found the source of the leak to be a circumferential crack in a tube at the hot leg expansion transition location, which is near the top of the tubesheet. The licensee reviewed the eddy current test data from the previous refueling outage inspection in 1991 and found that this tube had exhibited a bobbin coil indication at that time. Two independent data analysts had missed this indication. The licensee found six other bobbin coil indications that the data analysts had also missed. The licensee reports that, if these indications had been correctly analyzed, the licensee would have evaluated them further. It is common industry practice to perform supplemental MRPC inspections (and sometimes pulled tube examinations) to better characterize low amplitude, ambiguous, or distorted bobbin coil indications. In 1991, the licensee did not perform MRPC inspections at the expansion transition locations of these steam generators.

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Because of the finding of the circumferential crack, the licensee conducted a 100 percent MRPC inspection of the expansion transition locations on the hot leg side of both steam generators and a 20-percent MRPC inspection of the expansion transition locations on the cold leg side of one steam generator. The licensee found indications (generally circumferential) on the hot leg side of 488 tubes. The licensee found no indications in the tubes on the cold leg side. Tubes with MRPC indications were also inspected with a bobbin probe; however, this probe did not detect most of the MRPC indications. The licensee sleeved 448 of the tubes with indications and plugged and stabilized the remaining 40 tubes. The licensee pulled a number of tubes, including three tubes with circumferential indications in the expansion transition location, to measure the extent of damage and to determine the degradation mechanism.

On August 7, 1992, the licensee reported to the NRC the preliminary results from the examination of the pulled tube segments. On three of the tubes, the examination found circumferentially oriented intergranular stress corrosion cracking (IGSCC) beginning on the outer diameter of the tubes. The cracks extended 360 degrees around each of the tubes and had average depths ranging between 88 to 94 percent of the tube wall thickness. The licensee reported that all three tubes failed to satisfy the structural margin criteria of NRC Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes." Thus, at the time of the March 1992 shutdown, these tubes retained inadequate structural margins to ensure their integrity for the full range of normal operating, transient, and postulated accident conditions.

Discussion

The licensee attributed the missed bobbin indications from the 1991 inspection to (1) a lack of training for the eddy current data analysts in guidelines for tube damage mechanisms specific to ANO-2 and similar sites; (2) the lack of a performance demonstration test of the data analysts using actual site data; and (3) the inherent difficulties in analyzing signals at the expansion transition locations caused by interference from the tubesheet, the expansion transition geometry and deposits on the SG tubes. The indications may also have been missed if the entire tubesheet entry signal was not screened for distortions that could indicate flaws at the expansion transition locations. Examining the entire tubesheet entry signal is important because the precise location of the expansion transition in relation to the top of the tubesheet, and thus the location of the cracks, varies among tubes.

Although the above may have been contributing factors, the NRC staff believes that the failure to prevent the excessive loss of tube structural margin at ANO-2 resulted primarily from using test probes inappropriate for tube locations susceptible to circumferential cracking. Based on experience at other CE plants, the tube expansion transition locations in CE steam generators are known to be susceptible to circumferential IGSCC. Bobbin probes are generally insensitive to circumferential IGSCC unless the cracks are large enough to have a significant axial component or crack opening. Before this condition could occur, the structural margins of the tube would already be substantially reduced.

Circumferential cracks can be detected satisfactorily only by specialized probes such as the MRPC. This has has been discussed extensively in industry literature and reported by the NRC. In Information Notice 90-49, "Stress Corrosion Cracking In PWR Steam Generator Tubes," the NRC alerted licensees to problems with such cracks at Millstone Unit 2 and at Maine Yankee. During the ANO-2 SG tube inspections in 1991, the licensee did not perform MRPC inspections of the expansion transition locations.

Effective inspection programs to ensure early detection of SG tube degradation can be achieved through the use of data acquisition equipment and procedures and data analysis procedures appropriate for the detection of potential flaw types, including circumferential IGSCC. Training and performance demonstration testing of the data analysts for all potential flaw types are important parts of such programs. Further, by reviewing relevant industry experience, insights may be gained into the types of flaws that affect a particular SG as a function of the design or fabrication process. For example:

> Industry reports show that circumferential IGSCC indications have been observed at the tube expansion transition locations of SGs at four CE plants (including ANO-2) and at several Westinghouse plants. Tubes in the SGs at the four CE plants were explosively expanded against the tubesheet using the CE "explansion" process. Tubes in seven of the affected Westinghouse plants with Model 51 SGs were explosively expanded against the tubesheet using the Westinghouse WEXTEX process. Tubes in two of the affected Westinghouse plants with Model D SGs were expanded using a mechanical rolling process over the full depth of the tubesheet. Tubes in another affected Westinghouse plant with Model 51 SGs were expanded over only the lower 2-1/2 inches of the tubesheet thickness (partial depth expansion) using a mechanical rolling process.
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> Widespread circumferential IGSCC has also been observed at tube support plate intersections in Westinghouse Model 51 SGs at the North Anna Power Station Unit 1 and in the SGs that were replaced at the Palisades Nuclear Power Station. Axial stress caused by denting is believed to have caused the cracks in these steam generators. Denting at the top of the tubesheet crevices in Westinghouse Model 51 SGs at the Donald C. Cook Plant (Cook) Unit 1 appears to have caused circumferential SCC at these locations. The tubes in the SGs at Cook were partial depth expanded against the tubesheet. Isolated instances of circumferential cracks have also been observed at the upper "egg-crate" supports and U-bends in the CE SGs that were replaced at

the Millstone Nuclear Power Station Unit 2 and at a row 1 U-bend in the Westinghouse Model 51 SGs at the Zion Nuclear Power Station Unit 1.

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 Office of Nuclear Reactor Regulation (NRR) project manager.

Brian Grimes

Brian K. Grimes, Director
Division of Operating Reactor Support
Office of Nuclear Reactor Regulation

Technical contact: E. Murphy, NRR
(301) 504-2710

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LIST OF RECENTLY ISSUED
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Information Notice No.	Subject	Date of Issuance	Issued to
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92-78	Piston to Cylinder Liner Tin Smearing on Cooper-Bessemer KSV Diesel Engines	11/30/92	All holders of OLs or CPs for nuclear power reactors.
92-77	Questionable Selection and Review to Determine Suitability of Electropneumatic Relays for Certain Applications	11/17/92	All holders of OLs or CPs for nuclear power reactors.
92-76	Issuance of Supplement 1 to NUREG-1358, "Lessons Learned from the Special Inspection Program for Emergency Operating Procedures (Conducted October 1988 - September 1991)"	11/13/92	All holders of OLs or CPs for nuclear power reactors.
92-75	Unplanned Intakes of Airborne Radioactive Material by Individuals at Nuclear Power Plants	11/12/92	All holders of OLs or CPs for nuclear power reactors.
92-74	Power Oscillations at Washington Nuclear Power Unit 2	11/10/92	All holders of OLs or CPs for nuclear power reactors.
92-61, Supp. 1	Loss of High Head Safety Injection	11/06/92	All holders of OLs or CPs for nuclear power reactors.
92-73	Removal of A Fuel Element from A Research Reactor Core While Critical	11-04/92	All holders of OLs or CPs for nuclear power reactors.

OL = Operating License
CP = Construction Permit

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