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

August 3, 2004

NRC INFORMATION NOTICE 2004-16: TUBE LEAKAGE DUE TO A FABRICATION FLAW IN A REPLACEMENT STEAM GENERATOR

Addressees:

All holders of operating licenses for pressurized-water reactors (PWRs), except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor.

Purpose:

The U.S. Nuclear Regulatory Commission is issuing this information notice to inform addressees about recent operating experience with replacement steam generators. In particular, the potential for tubes to be damaged during fabrication and packaging. The NRC anticipates that recipients will review the information for applicability to their facilities and consider taking actions, as appropriate, to avoid similar issues. However, no specific action or written response is required.

Description of Circumstances:

Both steam generators at Palo Verde Nuclear Generating Station (PVNGS), Unit 2, were replaced in December 2003. These new steam generators incorporate many of the design enhancements present in other replacement steam generators (e.g., Alloy 690 tubes, stainless steel tube supports). The tube bundle in the new steam generators is consistent with the original Combustion Engineering design with a horizontal run between the hot and cold leg rather than the more typical U-shape.

When the plant was started up following the replacement of the steam generators in December 2003, the licensee observed a small primary-to-secondary leak measuring approximately 0.6 gallons per day (gpd) (2.3 liters per day (lpd)). Over the following 2 months, the leak rate varied between 0.4 and 0.7 gpd (1.5 and 2.6 lpd) until February 19, 2004, when the leak rate increased from approximately 0.7 to 11 gpd (2.6 to 41 lpd) in a 38-minute timeframe. Although the leak rate did not exceed the technical specification limit, the plant was shut down to identify the source of the leak.

While the plant was shut down, the secondary side of the steam generator was pressurized to 600 pounds per square inch (psi) (4137 kilopascal (kPa)) to assist in the identification of the leaking tube or tubes. During this pressure test, leakage was easily observed coming from a peripheral tube. This tube was subsequently inspected with both a bobbin and a rotating probe. These inspections did not reveal evidence of inservice degradation. These inspections did confirm the presence of a dent near a vertical support in the middle of the horizontal run of the tube that was detected in the preservice examination. This dent signal was considered anomalous because it differed from a typical dent signal in that it exhibited some flawlike characteristics (i.e., it had a vertical component). A comparison of the preservice bobbin and

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rotating probe inspection data to the data obtained during the outage revealed no significant differences in the dent signal. Although the dent signal was anomalous, there was no distinct indication of material volume loss.

Since the eddy current inspections of the affected tube did not provide conclusive evidence of a through wall flaw, additional testing was performed. This testing included primary and secondary side visual inspections and an in situ pressure test. The visual inspections confirmed the presence of a dent which did not appear to be due to the fabrication of the support structure since the dent was not located directly next to a support strap and did not appear to be the result of impact or leverage. During an in situ pressure test of the entire tube, 0.08 gpm (0.3 lpm) leakage was observed at the differential pressure associated with postulated accident conditions (e.g., a main steam line break), and the tube did not burst at three times the differential pressure associated with normal operating conditions. These tests confirmed the tube had adequate structural integrity. In addition, the leakage from this tube was well below the allowable leakage under postulated accident conditions. Following the in situ pressure test, the leaking tube was plugged and stabilized.

In response to the findings regarding the leaking tube, the rotating probe data for all dent signals obtained during the preservice examination were reviewed to ascertain whether similar anomalous dent signals existed. In addition, rotating probe inspections were performed at dents whose voltages exceeded a specific voltage (e.g., 2 to 5 volts) if these dents had not been inspected with a rotating probe during the preservice inspection. Based on these efforts, one additional tube was identified with an anomalous signal, but was not conclusively similar to the other indication with respect to the vertical presentation of the eddy current signal. This tube was plugged during the preservice examination because the dent obstructed the passage of the normal-sized bobbin probe and there was a concern regarding the future inspectability of this location.

Upon identification of the leaking tube, additional efforts were made to determine the root cause of the leak. These efforts included reviewing steam generator manufacturing records and developing mock-up specimens to simulate the anomalous eddy current signal in the leaking tube.

During the review of the manufacturing records of the steam generator, it was determined that one tube was scrapped during the fabrication of the replacement steam generators since it was damaged (or pierced) by a packing screw. Screws are used in the packing crate in which the tubes are shipped from the tubing manufacturer to the steam generator fabrication facility. The affected portion of this tube was sent back to the tubing manufacturer and corrective actions were taken; however, at the time of the discovery of this damaged tube, all of the tubes in one of the Unit 2 steam generators were installed and the other steam generator was in the process of being fabricated.

To simulate the anomalous dent signal in the leaking tube, a series of dents was fabricated in a mock-up facility. The simulation included impact dents from a nail, a screw, and a drill bit. A wood screw, similar to that used in the tube manufacturer’s crate, was driven through a piece of wood and into the sample tube. Eddy current testing was performed on these specimens and the damage caused by the wood screw yielded a similar anomalous signal to that found in the leaking tube at Palo Verde.

During its formal root cause evaluation, the licensee for PVNGS Unit 2 confirmed that the tube packing crate used wood spacers and cross brace materials that were assembled using common screws as the tubes were loaded into the crate. The design of this packing material placed the screws in close proximity to specific locations on some tubes, and the location, shape, and size of the deformation in the leaking tube are consistent with damage that would occur if a screw penetrated completely through the packing material and came in contact with the tube.

As a result of the findings, the licensee took many corrective actions, including performing inspection of selected tubes, plugging and stabilizing the leaking tube, adding additional quality control inspectors at the steam generator fabrication facility (since replacement steam generators for Unit 1 are being fabricated at this facility), modifying the receipt inspections performed (including procedural changes) on the tubes at the fabrication facility, evaluating/ modifying the packing procedure/design, identifying the tubes that were shipped in package locations where packing screw damage was possible, and initiating additional mock-up testing to improve the capability to identify and characterize volumetric flaws located within a dent (e.g., puncture-type defects).

After concluding that there was reasonable assurance of tube integrity, the licensee returned the plant to service. The primary-to-secondary leak rate following startup was near the detection threshold (i.e., less than 0.1 gpd (0.4 lpd)). In addition, following plant startup, six additional tubes were found at the fabrication facility during the unpacking of tubes for the Palo Verde Unit 1 replacement steam generators that had been pierced by a packing crate screw. These tubes were not installed in any of the steam generators being fabricated.

Discussion:

Steam generators have been replaced at many U.S. plants, and a number of other plants plan to replace in the next several years.

The finding of tubes damaged during the fabrication of the Palo Verde replacement steam generators illustrates the importance of monitoring the fabrication process. This includes the packing procedures for the tubes and the receipt inspections performed on these tubes once they arrive at the steam generator fabrication facility.

In addition, the findings at Palo Verde illustrate the importance of fully evaluating the implications of all abnormal conditions (i.e., conditions adverse to quality) identified during the fabrication process and communicating these results to all affected parties within an organization. In this instance, the personnel performing the preservice inspection at Palo Verde were not specifically notified of the identification of a tube that been damaged by a packing screw. As a result, they did not consider the potential for this type of flaw to exist in their review of the inspection data. By communicating non-conforming conditions observed during fabrication to the individuals responsible for the preservice examination, nondestructive examination techniques can be selected and the personnel trained to ensure potential defects are reliably detected and evaluated.

The findings at Palo Verde also illustrate the inspection challenges in finding flaws (such as from a screw) when they are located within a dent. These inspection challenges include determining the appropriate voltage threshold at which rotating probe examinations should be performed on a dent to detect flaws and determining the capability of the rotating probe to reliably identify flaws (e.g., holes) in a dent.

Lastly, the findings at Palo Verde indicate that the source of small amounts of primary-to-secondary leakage from volumetric defects can be determined through secondary side pressure tests.

This information notice does not require any 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 project manager in the NRC's Office of Nuclear Reactor Regulation (NRR).

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