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

August 28, 2003

NRC INFORMATION NOTICE 2003-13: STEAM GENERATOR TUBE DEGRADATION AT DIABLO CANYON

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 findings from a recent steam generator tube inspection at the Diablo Canyon Power Plant, Unit 2 (DCPP-2). The NRC anticipates that recipients will review the information for applicability to their facilities and consider taking actions, as appropriate, to avoid similar problems. However, no specific action or written response is required.

Description of Circumstances

DCPP-2 has four Westinghouse model 51 steam generators (SGs), with 7/8 inch outside diameter (OD), mill-annealed Alloy 600 tubing and drilled hole carbon steel tube support plates. The model 51 steam generator has 45 rows of tubes, with row 1 having the smallest bend radii in the U-bend area.

During Operating Cycle 11, a small steam generator tube leak (less than or equal to approximately 6.5 gallons per day) was present at DCPP-2. During the 2003 refueling outage, Pacific Gas & Electric (PG&E), the licensee for DCPP-2, performed SG secondary side pressure tests to locate the source of the SG leakage. Several potentially leaking SG tubes were identified and subsequent eddy current testing identified two contributing degradation modes: circumferential primary water stress corrosion cracking (PWSCC) in the U-bend region and axial outside diameter stress corrosion cracking (ODSCC) at the tube-to-tube support plate intersections.

Circumferential Indications

The licensee inspected the U-bend region of all of the active tubes in all four SGs with a rotating eddy current probe after finding leaking circumferential flaws in the U-bend region of a row 5 tube and identifying other circumferential indications in the U-bend region of other high row (above row 3) tubes during the initial examinations. In all, 12 tubes in rows 3 through 10 were identified with circumferential indications in the U-bend region. The indications were short, about 0.25 inch long. They were on the tube flank and originated from the inside of the

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tube. PG&E concluded that PWSCC was the most probable cause of cracking based on the susceptibility of mill-annealed Alloy 600 tubing, the residual stresses from bending the tubes, and the exposure of the tubes to primary coolant. It was not practical to remove one of these flawed tubes for destructive analysis because of the tubes' location in the SG. All tubes with circumferential indications in the U-bend region were pressure-tested in situ and met the structural integrity performance criteria. The estimated potential for accident induced leakage from these tubes when combined with estimated potential for accident induced leakage from other tubes in the SG did not exceed the accident leakage integrity performance criteria.

Axial ODSCC

All of the axial ODSCC indications identified in the potentially leaking tubes had been detected during the previous SG tube inspection. The tubes were left in service in accordance with the voltage-based alternate repair criteria (ARC) discussed in Generic Letter (GL) 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," and the plant's technical specifications. During subsequent bobbin coil inspections of the SG tubes during the 2003 refueling outage, the licensee detected an unexpected number of large bobbin voltage indications (i.e., greater than 3 volts), including one indication which had a substantially larger voltage than any other indication. This indication measured 21.5 volts. The licensee investigated the cause of the 21.5 volt indication and the cause of the unexpected number of high voltage (>3 volts) indications.

The licensee found that the tube with the 21.5 volt indication had been previously identified with a 2 volt indication during the 2001 refueling outage. The tube was left in service at that time since the bobbin voltage (2 volts) met the plant's technical specification repair criteria for remaining in service. PG&E had also inspected the indication with a rotating eddy current probe during the 2001 outage and the 2003 outage. During the root cause investigation in 2003, PG&E used the rotating probe eddy current data to estimate the length and depth of the flaw for both outages. The licensee estimated that the flaw was nearly through wall at the time of the 2001 refueling outage and that the flaw fully penetrated the tube wall during the ensuing operating cycle. This through wall penetration occurred over a significant portion of the crack length. Industry data shows that flaw voltage response increases sharply upon initial through wall penetration and with subsequent lengthwise growth of the through wall component. Thus, the licensee concluded that the large increase in voltage observed for this indication over the operating cycle was the result of the crack penetrating entirely through wall over a significant length rather than being the result of a significant increase in growth rate of the physical dimensions of the flaw.

The licensee partly attributed the unexpected number of large voltage indications (other than the 21.5 volt indication) to the voltage growth rate distribution used in earlier projections. These earlier projections were based on the industry methodology outlined in the EPRI report "Steam Generator Tubing Outside Diameter Stress Corrosion Cracking at Tube Support Plates Database for Alternate Repair Limits." The methodology described in the EPRI report recognizes that voltage growth rates can be higher for indications with larger initial voltages. To determine the growth rate distribution, the growth rate data is binned based on this initial voltage. The industry methodology specifies that each bin (i.e., range of initial voltages) should have a certain number of growth rate data points. To satisfy this minimum number-of-data-points requirement, the licensee expanded the range of the highest growth rate bin to include

the growth rates for indications with lower initial voltages. This reduced the probability that a large growth rate would be selected for indications in the upper voltage bin and resulted in nonconservative projections of the bobbin voltage.

The licensee pulled two tubes with axial ODSCC indications for destructive examination: the tube with the largest axial ODSCC flaw voltage and a tube with a large voltage axial ODSCC flaw. The laboratory burst and leakage test results indicated that both tubes had adequate integrity.

Discussion

DCPP-2 identified two steam generator tube issues during the 2003 refueling outage: circumferential indications in the U-bend region out to row 10, and large, increases in bobbin voltage associated with axial ODSCC at the tube-to-tube support plate intersections.

Historically, industry practice has been to inspect the U-bend region of low row tubes for indications of circumferential cracking and to expand the inspection to higher row tubes based on the results. This practice was developed based on the understanding that the low row tubes have higher residual stress levels in the U-bend region due to the tighter bend radii than the U-bend region of higher row tubes. The higher residual stress level makes the U-bend region of low row tubes more susceptialble to cracking. Prior to the experience at Diable Canyon, operating experience from steam generator inspections has validated this approach. The experience from DCPP-2 suggests that the U-bend region of higher row tubes may have a similar susceptibility to cracking in the U-bend region as the U-bend region of lower row tubes.

PG&E's experience with axial ODSCC indications yields two insights. First, collection and further evaluation of rotating probe inspection data from axial ODSCC indications may help identify indications that could be prone to significant bobbin voltage growth. Second, when developing voltage-dependent growth rates through the use of generic industry guidance or other methodologies, it is important that the methodology result in conservative growth rates.

The previously described examples of SG tube degradation illustrates the need for maintaining robust steam generator inspection programs. An effective program should sample for degradation based on both operating experience and engineering assessments of potentially susceptible locations and should be able to conservatively predict degradation growth.

Related Generic Communications

The following documents describe other recent reactor operating experience with steam generator tubes:

IN 2003-05, “Failure to Detect Freespan Cracks in PWR Steam Generator Tubes,” dated June 5, 2003

IN 2002-02 and IN 2002-02 supplement 1, “Recent Experience With Plugged Steam Generator Tubes” dated January 8, 2002 and July, 17, 2002

IN 2002-21, “Axial Outside-Diameter Cracking Affecting Thermally Treated Alloy 600 Steam Generator Tubing” dated June 25, 2002

IN 2001-16, “Recent Foreign and Domestic Experience with Degradation of Steam Generator Tubes and Internals,” dated October 31, 2001

NRC Generic Letter 97-05, “Steam Generator Tube Inspection Techniques,” dated December 17, 1997

Inspection Report 50-323/03-09, “Diablo Canyon Power Plant - NRC Special Team Inspection Report” dated May 8, 2003 (Adams ML031290198)

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 one of the technical contacts listed below or the appropriate project manager in the NRC’s Office of Nuclear Reactor Regulation (NRR).

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2003-12	Problems Involved in Monitoring Dose to the Hands Resulting from the Handling of Radiopharmaceuticals	08/22/2003	All holders of 10 CFR Parts 32, 33, and 35 licenses.
2003-11	Leakage Found on Bottom-Mounted Instrumentation Nozzles	08/13/2003	All holders of operating license or construction permits for nuclear power reactors, except those that have permanently ceased operations and have certified that fuel has been permanently removed from the reactor.
2003-10	Criticality Monitoring System Degradation at BWX Technologies, Inc., Nuclear Products Division, Lynchburg, VA	08/04/2003	All U.S. Nuclear Regulatory Commission (NRC) licensees authorized to possess a critical mass of special nuclear material.
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