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

May 19, 1997

NRC INFORMATION NOTICE 97-26: DEGRADATION IN SMALL-RADIUS U-BEND REGIONS OF STEAM GENERATOR TUBES

Addressees

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

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to disseminate information about recent degradation affecting small-radius (rows 1 and 2) U-bend regions of tubes in recirculating steam generators (SGs), in order to alert utilities to potential problems in ensuring the integrity of the small-radius U-bends, and to provide information about action taken by certain licensees to ensure adequate integrity. It is expected that recipients will review the information for applicability to their facilities and consider this information, as appropriate, in their SG inspection programs. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

Licensees that use Westinghouse-designed recirculating SGs have for many years identified indications in the U-bend regions of tubes with small radii. During the late 1970s and early 1980s, many units, as a preventative measure, plugged small-radius U-bend tubes to avoid potential forced outages due to leakage. However, some licensees subsequently unplugged these tubes and performed in situ stress relief to reduce the susceptibility for degradation. Also SG designs evolved over time and a number of different material conditions are represented in currently operating PWRs. These include mill-annealed alloy 600, mill-annealed alloy 600 in situ stress relieved, thermally treated alloy 600, and thermally treated alloy 690. The following discussion of experience at four plants represents recent operating experience regarding U-bend degradation that involved various tube material conditions.

During a 1996 inspection, Commonwealth Edison Company (ComEd) identified a total of 64 axially oriented and 2 circumferentially oriented indications in the U-bends of the row 1 SG tubes at Zion Unit 2. ComEd characterized the indications as primary water stress-corrosion cracking. The tubes at Zion Unit 2 were fabricated with mill-annealed alloy 600 material, and the U-bends had not been heat treated. As a result of the inspection findings, ComEd preventively plugged all the row 1 tubes at Zion Unit 2.

During a 1996 inspection, the Tennessee Valley Authority (TVA) identified axial indications in 17 small-radius U-bends of SG tubes at Sequoyah Unit 2 and characterized the degradation as primary water stress-corrosion cracking. The tubes were fabricated with mill-annealed alloy 600 material and the U-bends were in situ heat treated during the cycle 6 outage in 1994. TVA plugged the 17 row 1 tubes that contained the U-bend indications.

During 1992 and 1995 inspections, Pacific Gas & Electric Company (PG&E) identified circumferential indications having relatively small arc angles in the small-radius U-bends of SG tubes at Diablo Canyon Unit 1. The tubes were fabricated with mill-annealed alloy 600 material and the small-radius U-bends were in situ heat treated after the second refueling outage in 1988. PG&E plugged the degraded tubes.

During a 1996 inspection, ComEd identified a single axial indication in the U-bend of one of the SG tubes at Braidwood Unit 2. The Braidwood Unit 2 tubes were fabricated with thermally treated alloy 600 tubes and the U-bends in the first seven rows received additional thermal stress relief after bending during the manufacturing process. ComEd plugged the degraded tube.

A small number of axial indications originating on the outside diameter of the tubes have been reported in the small-radius U-bend regions of the SGs at Palo Verde 1, 2, and 3 and St. Lucie 1. These SGs were designed by Combustion Engineering.

Discussion

U-bend degradation has occurred in mill-annealed alloy 600 tubes irrespective of whether they have been heat treated. Tubes with thermally treated alloy 600 material are less susceptible to degradation than mill-annealed alloy 600 tubes. However, thermally treated alloy 600 tubes have also begun to experience U-bend degradation. None of the degraded thermally treated alloy 600 tubes have been removed from SGs for confirmation of the degradation mechanism. Reports of U-bend degradation have been based on eddy current inspection results. The susceptibility to cracking in small-radius U-bends and the findings of recent field inspections have emphasized the importance of inspection of this area of SGs with techniques capable of accurately detecting U-bend degradation.

U-bend degradation can potentially impair tube integrity if not effectively managed. Concerns in this regard stem from limitations of eddy current testing to detect and size U-bend cracks, the potential for some U-bend cracks to have relatively long lengths, and the potential for high crack growth rates for some of these cracks. The industry standard bobbin coil has proven unreliable for detecting U-bend cracks and, in addition, is not qualified for this application under the Electric Power Research Institute (EPRI) technique qualification protocol. The industry has developed special probes for these inspections. The industry has qualified a rotating pancake coil and a Plus Point coil for detecting indications in small-radius U-bends, in accordance with enhanced qualification criteria developed by EPRI.

There continues to be an absence of pulled tube information to confirm that the detection threshold for these cracks is better than 40 or 50-percent through wall. In addition, available inspection techniques are not capable of reliably sizing crack depths and, for this reason, it has been industry's practice to "plug on detection" U-bend indications that are found.

Information available on crack growth rates being experienced in the field is very limited by virtue of the inability to perform reliable crack depth measurements and the resulting need to "plug on detection." However, U-bend cracks have led to leakage as early as the first cycle of operation and, thus, crack growth rates may potentially be high for some cracks. Given the relatively high detection thresholds, the relatively long operating cycles, and the potentially high growth rates, the depth of cracks may be in excess of 50-percent through wall when they are first detected.

In view of these concerns, effective management of the degradation of SG tubes is important to ensure that adequate tube integrity is being maintained in accordance with 10 CFR Part 50, Appendices A and B. One such approach being implemented by a number of licensees involves the use of tube integrity assessments to ensure that inspection sensitivity to U-bend cracks and the frequency and scope of inspection are sufficient to ensure that U-bend flaws are being detected and removed from service before tube integrity is impaired.

For example, ComEd performed in situ pressure tests at Zion Unit 2 on four tubes having the longest axial U-bend indications and on two tubes with circumferential U-bend indications using pressure loading consistent with the margins recommended in Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes." The two tubes having circumferential indications satisfied RG 1.121 margins without leaking. Three of the four tubes having axial indications leaked at a pressure of main steamline break conditions but did not burst under a pressure loading of three-times-normal operating pressure. Because of the limitations of the test equipment, the pressure in the fourth tube did not reach the three-times-normal operating pressure criterion of RG 1.121. For this tube, ComEd performed analyses to show that the tube would not burst under a pressure loading of three-times-normal operating pressure. These analyses are based on eddy current test measurements. Since these measurements may have large uncertainties, ComEd conservatively assumed that the cracks were 100-percent through wall. On the basis of the leakage measurements at main steamline break pressures, ComEd was able to demonstrate that accident leakage would satisfy the requirements of 10 CFR Part 100.

For U-bend indications at Sequoyah Unit 2, TVA did not perform in situ pressure testing; instead, it performed bounding analyses to show that the three tubes having the largest U-bend cracking satisfied RG 1.121 criteria. However, it should be noted that in situ pressure testing provides more definitive assurance of structural and leakage integrity than analyses.

For axial indications in the small-radius U-bend regions of the SGs at Palo Verde 1, 2, and 3 and St. Lucie 1, the licensees plugged the tubes.

As shown by the examples discussed above, the integrity of the small-radius U-bend regions can be more fully ensured by efforts that include performing inspections of rows 1 and 2 U-bends using qualified eddy current techniques; performing in situ pressure testing, as necessary, to assess the condition of defective tubes; taking appropriate corrective actions, including plugging defective tubes; and assessing the appropriate operating intervals until the next SG tube inspection.

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 contacts list below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

Seymour Weiss for

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Information Notice No.	Subject	Date of Issuance	Issued to
87-10, Sup. 1	Potential for Water Hammer During Restart of Residual Heat Removal Pumps	05/15/97	All holders of OLs or CPs for boiling-water reactors
97-25	Dynamic Range Uncertainties in the Reactor Vessel Level Instrumentation	05/09/97	All holders of OLs or CPs for Westinghouse pressurized-water reactors
97-24	Failure of Packing Nuts on One-Inch Uranium Hexafluoride Cylinder Valves	05/08/97	All U.S. Nuclear Regulatory Commission licensees and certificatees authorized to handle uranium hexafluoride in 30- and 48-inch diameter cylinders
97-23	Evaluation and Reporting of Fires and Unplanned Chemical Reactor Events at Fuel Cycle Facilities	05/07/97	All fuel cycle conversion, enrichment, and fabrication facilities
97-22	Failure of Welded-Steel Moment-Resisting Frames During the Northridge Earthquake	04/25/97	All holders of OLs or CPs for nuclear power reactors
97-21	Availability of Alternate AC Power Source Designed for Station Blackout Event	04/18/97	All holders of OLs for nuclear power reactors
97-20	Identification of Certain Uranium Hexafluoride Cylinders that do not comply with ANSI N14.1 Fabrication Standards	04/17/97	All holders of OLs for nuclear power

OL = Operating License
CP = Construction Permit

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OFC	Contacts	BC/EMCB	BC/PECB:DRPM	AD/DRPM
NAME	JTsao* EBenner*	JStrosnider*	AChaffee*	MSlosson SHW for
DATE	04/08/97 04/08/97	04/16/97	05/08/97	05/13/97

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