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

June 21, 1996

NRC INFORMATION NOTICE 96-38: RESULTS OF STEAM GENERATOR TUBE EXAMINATIONS

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 promulgate information about steam generator tube examinations. 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

Improved techniques and equipment are constantly developed to detect flaws in steam generator tubes. In addition, as nuclear power plants get older, different degradation mechanisms of steam generator tubes occur. This information notice discusses recent experiences by licensees involving these new techniques and equipment and different degradation mechanisms.

Recent steam generator tube examinations have revealed degradation at a number of locations, such as in dented areas, the expansion transition region, the freespan region, and in the tubesheet crevice. The types of degradation observed in these locations are discussed below. In addition to identifying several degradation mechanisms, these examinations raised a number of technical issues with respect to classifying inspection results, periodicity of examinations, and expanding the initial inspection scope.

Axial and circumferential indications at dented tube support plates were identified at a number of plants, including Sequoyah Nuclear Plant Unit 1, Diablo Canyon Nuclear Power Plant Unit 1, and Salem Generating Station Unit 1. These indications are associated with minor dents (i.e., dents that can be inspected with a standard size probe). These dented regions were examined with Cecco probes or rotating probes with plus-point coils or pancake coils (or both). On the basis of the examinations, the axial indications appear to have initiated from the inside diameter of the tube, and the circumferential indications appear to have initiated from the outside diameter of the tube. However, at Diablo Canyon Unit 1, several circumferential indications have initiated from the inside diameter of the tube (as evidenced by destructive examination).

Some plants that have Combustion Engineering and Westinghouse-designed steam generators also reported circumferential indications at the expansion transition region. Among these are Sequoyah Nuclear Plant Unit 1, Diablo Canyon Unit 1, Salem Unit 1, Arkansas Nuclear One Unit 2, Braidwood Unit 1, Byron Unit 1, and Callaway Unit 1. At particular plants, from tens to thousands of indications were reported.

The circumferential indications at the expansion transition have occurred at roll expansions, kinetic/explosive expansions, and hydraulic expansions. For example, circumferential indications have been reported in mechanically roll-expanded tubes at Farley Unit 1, Westinghouse explosively expanded (i.e., WEXTEX) tubes at Sequoyah Unit 1, Salem Unit 1, and Diablo Canyon Unit 1, Combustion Engineering explosively expanded tubes (i.e., EXPLANSION tubes) at Arkansas Nuclear One Unit 2, and in hydraulically expanded tubes at Callaway Unit 1. The majority of these indications were seen at the hot-leg expansion transition; however, circumferential indications were reported at the cold-leg expansion transition at Arkansas Nuclear One Unit 2. The circumferential cracks detected at these plants were all in Alloy 600 mill-annealed tubes.

Freespan degradation has been reported at a few plants. Freespan degradation is degradation observed above the sludge pile region at the top of the tubesheet and is not located at any support structure (e.g., tube support plates including eggcrates, anti-vibration bars, and batwings). Historically, moderate amounts of freespan degradation had been observed at McGuire Units 1 and 2 and at Palo Verde Units 1, 2, and 3. During the fall outages, Arkansas Nuclear One Unit 2, Farley Unit 1, and Point Beach Unit 1 reported freespan tube degradation. In addition, Oconee Units 1, 2, and 3 reported freespan axial indications attributed to intergranular attack.

A few plants have tubes which are only partially expanded in the tubesheet. As a result, there is a crevice between the tube and the tubesheet for the portion of the tube in the tubesheet that is not expanded. Corrosion products can accumulate in this crevice and can lead to tube degradation. Historically, tubesheet crevice region defects have been observed with the bobbin coil and repaired, accordingly; however, many of the indications detected during outages this fall were not found with the conventional bobbin coil probe. As a result, extensive examinations using alternate techniques were performed (e.g., rotating pancake coil examinations). Extensive tube repairs were performed, such as sleeving at Zion Unit 1 and tube rerolling at Point Beach Unit 1.

Discussion

Steam generators with mill-annealed Alloy 600 steam generator tubes are susceptible to such degradation as stress corrosion cracking. Degradation has been observed in the hot legs and cold legs of the steam generator tubes, in the expanded portion of the tube, at the expansion transition, in the tube-to-tubesheet crevice, in the sludge pile, in the freespan, and at tube support structures such as the tube support plate, batwings, anti-vibration bars, and vertical straps. The severity of the degradation and the number of tubes affected tend to be plant specific since these depend on many factors such as temperature, operating time, water chemistry history, and tube mechanical properties, including microstructure. Inspections have illustrated the importance of comprehensive steam generator tube examinations using appropriate techniques to ensure tube integrity even if a specific type of degradation has not been observed at a given location in the past. Previous inspection findings do not ensure that a location/tube is not susceptible to a particular mechanism. For example, before the inspections at Callaway Unit 1, no circumferential cracking had occurred domestically at tubes which had been hydraulically expanded within the tubesheet. The inspections at Callaway demonstrate that continually assessing the condition of all portions of the steam generator tube can ensure that new forms of degradation are detected.

The recent inspections also indicate the importance of comprehensively examining all portions of the steam generator tubes using techniques and equipment capable of reliably detecting degradation to which the steam generator tubes may potentially be susceptible. This experience calls into question the effectiveness of the bobbin coil for detecting circumferential indications or for detecting indications where significant interfering signals exist (e.g., expansion transition locations, dented locations, and locations with excessive deposits), as discussed in NRC Information Notice 94-88, "Inservice Inspection Deficiencies Result in Severely Degraded Steam Generator Tubes." In addition, this experience further indicates that a generically qualified technique may need to be supplemented to account for the testing conditions at a specific plant. Furthermore, optimizing such test variables as probe design and frequencies for the type of degradation observed at the plant such as inside-diameter initiated indications versus outside-diameter initiated indications, and controlling such test variables as cable length and capacitance within the range for which the technique was qualified can be important in ensuring the reliable detection of degradation.

Several large indications were detected during the most recent examinations of steam generator tubes. As a result, several licensees took additional measures to ensure that all tubes were capable of withstanding the pressure loadings specified in Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes." These additional measures (in situ pressure testing and removing tubes for destructive examination) were performed even though many of these indications were repaired. Although methods other than removing tubes for destructive examination exist for evaluating tube integrity, tube removal has the advantage of assessing inspection reliability, developing additional confidence in the ability to size indications, determining the root cause of the degradation, and possibly identifying corrective actions. Assessment of the inspection findings after every inspection assures that all tubes are capable of performing their intended safety function for the planned operating interval. In some instances, these assessments have led to mid-cycle inspections.

When degraded tubes are left in service (i.e., for degradation mechanisms for which qualified sizing techniques exist), assessment of the acceptable operating interval typically involves a detailed knowledge of the growth rate of the degradation, the scope of the examination, and the capabilities of the inspection technique.

For degradation mechanisms for which there is no qualified depth sizing technique, a tube with an indication typically has been considered defective. In these instances, demonstrating that the largest indications detected during an inspection were capable of withstanding specified pressure loadings (through such techniques such in-situ pressure testing or burst and leakage testing or both) can provide assurance that tubes currently without indications will also be capable of withstanding specified pressure loadings at the end of the next inspection interval, if the interval is of comparable duration and operating parameters (e.g., water chemistry and hot leg temperature) to the previous inspection interval.

Although only steam generators that contain tubes made from mill-annealed Alloy 600 are discussed above, the information may have applicability to all PWRs. 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.

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Information Notice No.	Subject	Date of Issuance	Issued to
96-37	Inaccurate Reactor Water Level Indication and Inadvertent Draindown During Shutdown	06/18/96	All pressurized water reactor facilities holding an operating license or a construction permit
96-36	Degradation of Cooling Water Systems Due to Icing	06/12/96	All holders of OLs or CPs for nuclear power reactors
96-35	Failure of Safety Systems on Self-Shielded Irradiators Because of Inadequate Maintenance and Training	06/11/96	All U.S. Nuclear Regulatory Commission irradiator licensees and vendors
96-34	Hydrogen Gas Ignition during Closure Welding of a VSC-24 Multi-Assembly Sealed Basket	05/31/96	All holders of OLs or CPs for nuclear power reactors
96-33	Erroneous Data From Defective Thermocouple Results in a Fire	05/24/96	All material and fuel cycle licensees that monitor temperature with thermocouples
96-32	Implementation of 10 CFR 50.55a(g)(6)(ii)(A), "Augmented Examination of Reactor Vessel"	06/05/96	All holders of OLs or CPs for nuclear power reactors
96-31	Cross-Tied Safety Injection Accumulators	05/22/96	All holders of OLs or CPs for pressurized water reactors
96-30	Inaccuracy of Diagnostic Equipment for Motor-Operated Butterfly Valves	05/21/96	All holders of OLs or CPs for nuclear power reactors
96-29	Requirements in 10 CFR Part 21 for Reporting and Evaluating Software Errors	05/20/96	All holders of OLs or CPs for nuclear power reactors

OL = Operating License
CP = Construction Permit

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*See previous concurrence

Tech Editor reviewed and concurred on 3/26/96

Jim Conran of CRGR reviewed and approved on May 3,1996

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For degradation mechanisms for which there is no qualified depth sizing technique, a tube with an indication typically has been considered defective for determining tube repair options, classifying inspection results, and determining sample expansion criteria (regardless of probe type). In these instances, demonstrating that the largest indications detected during an inspection were capable of withstanding specified pressure loadings (through such techniques such in-situ pressure testing or burst and leakage testing or both) can provide assurance that tubes at the end of the next inspection interval will also be capable of withstanding specified pressure loadings, if the interval is of comparable duration and operating parameters (e.g., water chemistry and hot leg temperature) to the previous inspection interval,

An assessment of inspection findings may also indicate that the time between inspections can be lengthened. Typically, technical specifications state the frequency at which steam generator tubes are normally to be examined (e.g., 12 to 24 calendar months); however, these specifications also typically state when the frequency of inspection may be relaxed. For expanding the inspection interval beyond the specified interval (e.g., 24- or 40-calendar-month limit in the Standard Technical Specifications), Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-month Fuel Cycle," states, in part, that the 25-percent extension provision of Technical Specification 4.0.2 does not apply for extending the frequency for performing inservice inspections of the steam generator tubes.

Although primarily plants from two vendors and only steam generators that contain tubes made from mill-annealed Alloy 600 are discussed above, the information may have applicability to all PWRs. 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.

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