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

January 19, 1994

NRC INFORMATION NOTICE 94-05: POTENTIAL FAILURE OF STEAM GENERATOR TUBES WITH KINETICALLY WELDED SLEEVES

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 (IN) to alert addressees to the potential failure of steam generator tubes sleeved with kinetically (explosively) welded sleeves supplied by B&W Nuclear Service Company (BWNS). 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.

Background

For certain types of defects, a steam generator tube may be repaired using an approved sleeving method as an alternative to plugging the tube and removing it from service. In this process, the sleeve is positioned inside the steam generator tube so that it bridges the defect. The sleeve is then joined to the parent tube on both sides of the defect to serve as a new primary coolant interface and allow the tube to be returned to service. The sleeve can be joined to the tube wall by a mechanical seal or a weld. In the case of the kinetic welding process used by BWNS, an explosive charge expands a narrow band of the sleeve, fusing the outer sleeve wall to the inner tube wall. The process leaves residual stresses in the parent tube in the vicinity of the seal or weld which necessitates a post-weld heat treatment to relieve the stresses. The heat treatment is necessary in the parent tube because the tube is constructed of nickel Alloy 600 which is susceptible to stress corrosion cracking. Heat treatment is not necessary for the repair sleeve because it is made of nickel Alloy 690 which is more resistant to stress corrosion cracking.

Description of Circumstances of the McGuire 1 Incident

On August 22, 1993, operators at Unit 1 of the William B. McGuire Nuclear Station (McGuire) shut down the reactor because of a primary-to-secondary leak of about 760 liters [200 gallons] a day in steam generator A. This amount of leakage was within the technical specification limits but exceeded the administrative limit. Duke Power Company, the licensee, determined that a tube containing a BWNS kinetically welded sleeve was the source of the leak.

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The licensee removed the tube and the sleeve and found a circumferential crack in the parent tube just above the upper weld that joined the tube and the sleeve. The sleeve had been installed in 1991 and had been reinspected in April 1993 with no indications of cracking. The licensee used a rotating eddy current probe to examine the tube in situ and found an indication of a 120-degree to 180-degree circumferential defect. Destructive examination of the tube found a through-wall circumferential crack extending 270 degrees around the tube. The remaining 90 degrees of the tube was cracked approximately 50 percent through the wall. The crack had initiated from the inner surface (primary coolant side) and was characteristic of primary water side stress corrosion cracking (PWSCC). This type of stress corrosion cracking is a well-known failure mechanism in steam generator tubes. The mechanism is discussed in greater detail in the NRC information notices referenced at the end of this notice.

Another tube that had been sleeved at the same time as the leaking tube was also removed because an eddy current indication was found in the same area as in the failed tube. The indication was not a defect but was found to be the result of a variation in the surface geometry. Metallographic examination of the sleeve and parent tube also showed no signs of cracking. A review of process records showed that both tubes had received the same stress relief temperature and time. Hardness measurements confirmed that both tubes had been stress relieved after the kinetic welding.

Physical and chemical tests performed on the two tubes that were removed showed significant differences in the yield strength, carbon content, microstructure, and PWSCC susceptibility of the tube material. Tests sponsored jointly by Studsvik Power, the Swedish State Power Board, and AB Sandvick Steel show that a strong correlation exists between the carbon content and yield strength of the material and its susceptibility to PWSCC. Results of accelerated corrosion tests indicate that the time to cracking for reverse U-bend tubes is shorter for tubes constructed of materials with elevated yield strengths and carbon content. Based on a yield strength of 51.7 MPa (72.5 ksi) and a 0.05-percent carbon content, the material heat of the leaking tube would be ranked as one of the most susceptible to PWSCC in the plant. Metallographic examination of the tube material confirmed that it had a susceptible microstructure.

The BWNS kinetic sleeve stress-relieving process was originally qualified for a range of material corrosion susceptibilities. The stress relief temperatures were selected for what was believed to be the worst-case material. As evidenced by the destructive examination of the tube that leaked, material properties of steam generator tubes can be significantly different than the properties listed in a Certified Material Test Report (CMTR). Based on the CMTR, the yield strength of the leaking tube should have been 44.1 MPa (64 ksi), whereas the actual yield strength was 51.7 MPa (72.5 ksi). As a result of this and other industry data, BWNS will be evaluating the appropriateness of using alternate stress relief cycles for tubing of higher yield strengths.

The destructive examination of the sleeved tube at McGuire indicates that the root cause of the parent tube leak was the high susceptibility of the parent

tube material to stress corrosion cracking. Since 1990, approximately 4500 sleeves manufactured by BWNS have been installed worldwide using the kinetic weld process. According to BWNS, the defective tube sleeve at McGuire is the first confirmed case of cracking in a sleeved tube that had received the post-weld stress relief treatment required in the process qualification. A kinetically sleeved tube cracked and leaked at the Trojan Nuclear Plant in 1992 but that tube had not received the required post-weld stress relief treatment.

Discussion

In September 1993, B&W Nuclear Technologies contacted affected domestic licensees to inform them that the destructive examination of the sleeved tube at McGuire indicated that the root cause of the leak in the parent tube was a high susceptibility of the parent tube material to stress corrosion cracking. Five domestic nuclear units have installed tube sleeves using the BWNS process and a number of others are licensed to install them.

After the incident at McGuire 1, B&W Nuclear Technologies made recommendations to licensees with BWNS kinetically welded sleeves concerning (1) the identification of highly susceptible parent tube material, (2) procedures for dealing with primary-to-secondary leakage, and (3) operator readiness to respond to a tube leak such as the one that occurred at McGuire. Duke Power Company implemented these recommendations at McGuire 1 and for preventive purposes plugged sleeved tubes that were not axially restrained (peripheral tubes not completely surrounded by other tubes).

This failure mechanism has the potential for introducing difficult-to-detect circumferential stress corrosion cracks in steam generator tubes which could lead to rapidly increasing primary-to-secondary leakage. The NRC staff has contacted all affected licensees concerning the implications of these findings and is continuing to monitor this issue.

Related Generic Communications

1. NRC IN 92-80, "Operation With Steam Generator Tubes Seriously Degraded," December 7, 1992.

2. NRC IN 91-43, "Recent Incidents Involving Rapid Increases in Primary-to-Secondary Leak Rate," July 5, 1991.

3. NRC IN 90-49, "Stress Corrosion Cracking in PWR Steam Generator Tubes," August 6, 1990.

4. NRC IN 88-99, "Detection and Monitoring of Sudden and/or Rapidly Increasing Primary-to-Secondary Leakage," December 20, 1988.

No specific action or written response is required by this information notice. If you have any questions about this matter, please contact the person listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

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

Technical contact: H. Conrad, NRR
(301) 504-2703

Attachment:
List of Recently Issued NRC Information Notices

LIST OF RECENTLY ISSUED NRC INFORMATION NOTICES

Information Notice No.	Subject	Date of Issuance	Issued to
94-04	Digital Integrated Circuit Sockets with Intermittent Contact	01/14/94	All NRC licensees except licensed operators.
94-03	Deficiencies Identified during Service Water System Operational Performance Inspections	01/11/94	All holders of OLs or CPs for nuclear power reactors.
94-02	Inoperability of General Electric Magne-Blast Breaker Because of Misalignment of Close-Latch Spring	01/07/94	All holders of OLs or CPs for nuclear power reactors.
94-01	Turbine Blade Failures Caused by Torsional Excitation from Electrical System Disturbance	01/07/94	All holders of OLs or CPs for nuclear power reactors.
93-101	Jet Pump Hold-Down Beam Failure	12/17/93	All holders of OLs or CPs for boiling-water reactors.
93-100	Reporting Requirements for Bankruptcy	12/22/93	All U.S. Nuclear Regulatory Commission licensees.
91-29, Supp. 2	Potential Deficiencies Found During Electrical Distribution System Functional Inspections	12/22/93	All holders of OLs or CPs for nuclear power reactors.
93-99	Undervoltage Relay and Thermal Overload Setpoint Problems	12/21/93	All holders of OLs and CPs for nuclear power reactors.
93-98	Motor Brakes on Valve Actuator Motors	12/20/93	All holders of OLs and CPs for nuclear power reactors.
93-97	Failures of Yokes Installed on Walworth Gate and Globe Valves	12/17/93	All holders of OLs or CPs for nuclear power reactors.

OL = Operating License
CP = Construction Permit