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

February 12, 1996

NRC INFORMATION NOTICE 96-09: DAMAGE IN FOREIGN STEAM GENERATOR INTERNALS

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 alert addressees to recent findings of damage to steam generator internals, namely support plates and wrapper, at foreign PWR facilities. 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

In April 1995 during a routine eddy current inspection of the steam generator tubing at a foreign facility, anomalous support plate signals were observed at the uppermost support plate. The steam generators are similar but not identical to Westinghouse model 51 steam generators. The support plates are of the drilled hole type and fabricated from carbon steel. Video camera inspections were conducted to investigate the anomalous signals and revealed that a significant portion of the support plate had wasted away. Pieces of the affected region of the support plate were found resting on the next lower support plate.

Subsequent investigation has identified chemical cleaning performed in 1992 as the cause of the support plate damage. Review of previous eddy current data shows that the anomalous support plate signals were present in inspections dating back to 1993 when the first inservice inspection following chemical cleaning was performed. Support plate signals obtained immediately prior to the chemical cleaning were normal. The foreign regulatory authority believes that pipes used to direct the chemical solution into the steam generators were installed incorrectly, too close to the upper support plate. This caused an excessively high impingement velocity of the cleaning solution against the support plate which is believed to have been sufficient to render ineffective

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the corrosion inhibitor in the cleaning solution. U.S. industry representatives stated during recent meetings with the NRC staff that chemical cleanings which have been performed in the U.S. involve different cleaning agents and inhibitors than that used at the foreign facility and involve less risk for producing similar damage.

The support plate damage at the foreign facility effectively eliminated lateral support to tubes within the affected region. Lateral support provides vibrational stability and the ability to sustain earthquake and loss-of-coolant-accident loadings. Accordingly, all tubes found not to be supported at the uppermost support plate were plugged.

Based on this experience, the foreign utility carefully examined the support plate eddy current signals at other PWR facilities. At one of these units, with steam generators similar but not identical in design to Westinghouse model 51 steam generators, eddy current signals indicative of support plate ligament cracks were found at the uppermost support plate. The support plates are of the drilled hole type and are fabricated from carbon steel. Subsequent visual inspection confirmed the presence of ligament cracks near the periphery of the support plate. Part of the support plate periphery was observed to be entirely broken away in the vicinity of a radial seismic support. The steam generators at this facility have not been chemically cleaned. Review of past eddy current results indicates that the indications of ligament cracks date back at least 9 years. It is not clear whether the ligament cracks were present prior to initial service or whether the cracks may have developed shortly thereafter. The cause of these cracks is under investigation by the foreign utility and steam generator manufacturer. Tubes whose lateral support was potentially affected by these cracks have been plugged. Press reports indicate that similar indications of support plate ligament cracks have recently been found at other facilities in the same country with similar steam generators.

Visual inspections conducted in June 1994 at a foreign PWR facility revealed the bottom of the wrapper had dropped down by 20 millimeters in one steam generator and by 5 millimeters in another steam generator. The steam generators are similar but not identical to Westinghouse model 51 steam generators. The visual inspections were performed through handholes located above the tubesheet. Further investigation revealed that wrapper welds at each of six vertical supports in the first steam generator and at three of six vertical supports in the second steam generator had failed, allowing the downward displacement of the wrapper. The cause of this occurrence is under investigation by the foreign utility and the steam generator manufacturer. Their preliminary assessment is that unanticipated axial restraint against differential thermal expansion between the wrapper and steam generator pressure vessel shell led to significant loading of the wrapper vertical supports. This unanticipated restraint between the wrapper and shell may have been due to differential thermal expansion between support plate number 7 and the shell, preventing relative axial motion between the wrapper and shell at this elevation, during transients involving the auxiliary feedwater. Poor quality of the wrapper welds at the vertical support may also have been a contributing factor.

Implications of a complete fall of the wrapper have been assessed by the foreign utility to include the potential for loss of feedwater, damage to the largest radius tube u-bends, loose parts, and tube rupture. Accordingly, the foreign utility has implemented temporary repairs to stabilize and monitor the wrappers pending further investigation regarding long-term resolution of this matter.

Discussion

As illustrated by the foreign experience, support plate signal anomalies during eddy current testing of the steam generator tubes may be indicative of support plate damage or ligament cracking. The signal anomalies at the foreign units were present for several years before they were first identified by the data analysts. The Electric Power Research Institute (EPRI) has initiated an effort, in response to the foreign experience, to develop a qualified procedure for detecting support plate ligament cracks.

The steam generator tube support plates function to support the tubes against lateral displacement and vibration and to minimize bending moments in the tubes during accidents. Damage and/or cracking of the support plates can impair the ability of the support plates to perform this function and, thus, may potentially impair tube integrity. In addition, the staff has recently approved a 3 volt plugging criteria for two U.S. plants based, in part, upon evidence from inspections using EPRI preliminary procedures that the tube support plates are capable of locally constraining the tubes against tube rupture.

Known instances of support plate cracking/damage in the U.S. have generally involved support plates with significant denting. The potential for support plate cracks has tended not to be of significant concern in recent years since the steam generators most affected by denting have been replaced and, in addition, the industry has been successful in controlling denting progression in operating steam generators. The foreign experience serves to highlight that there are other mechanisms which may lead to support plate damage and/or cracking.

Based on the information available to the NRC staff, it is not yet known whether steam generators in the U.S. are vulnerable to the type of wrapper damage observed at the foreign unit.

The staff will continue to monitor information on support plate and wrapper damage as it becomes available from foreign authorities.

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.

Dennis M. Crutchfield

Dennis M. Crutchfield, Director
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LIST OF RECENTLY ISSUED
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Information Notice No.	Subject	Date of Issuance	Issued to
96-08	Thermally Induced Pressure Locking of a High Pressure Coolant Injection Gate Valve	02/05/96	All holders of OLs or CPs for nuclear power reactors
96-07	Slow Five Percent Scram Insertion Times Caused By Viton Diaphragms in Scram Solenoid Pilot Valves	01/26/96	All holders of OLs or CPs for boiling water reactors
96-06	Design and Testing Deficiencies of Tornado Dampers at Nuclear Power Plants	01/25/96	All holders of OLs or CPs for nuclear power reactors
96-05	Partial Bypass of Shutdown Cooling Flow from the Reactor Vessel	01/18/96	All holders of OLs or CPs for boiling water reactors
96-04	Incident Reporting Requirements for Radiography Licensees	01/10/96	All radiography licensees and manufacturers of radiography equipment
96-03	Main Steam Safety Valve Setpoint Variation as a Result of Thermal Effects	01/05/96	All holders of OLs or CPs for nuclear power reactors

OL = Operating License
CP = Construction Permit

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Reviewed by Tech Editor prior to receipt by PECB
Coordination of foreign information was handled by MCullingford, NRR
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