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

January 8, 2002

NRC INFORMATION NOTICE 2002-02: RECENT EXPERIENCE WITH PLUGGED STEAM GENERATOR TUBES

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 (NRC) is issuing this information notice to inform addressees about findings from recent inspections and examinations of steam generator tubes at Three Mile Island Unit 1 (TMI-1). 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, suggestions contained in this information notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

On October 8, 2001, TMI-1 was shut down for a refueling outage. While inspecting the tubes in the B Steam Generator, the licensee (AmerGen Energy Company, LLC) identified signs of wear near the upper tubesheet on the outer surface of four tubes on the periphery of the tube bundle. These wear indications did not appear to have been present during the prior steam generator tube inspections, which were performed approximately 2 years earlier. Given the pattern and location of the wear signs, the licensee suspected that a neighboring plugged tube had caused the wear. The licensee removed the upper plug in the suspected tube and performed a video inspection. The video inspection revealed that the tube was severed near the secondary face of the upper tubesheet and in physical contact with the drilled hole where the tube passed through the 15^{th} support plate. Neighboring tubes were not in contact with the drilled holes at the point where they passed through the 15^{th} support plate.

The licensee removed the lower portion of the fractured surface of the severed tube to assess the root cause. The preliminary laboratory investigation of the severed tube indicated signs of high cycle fatigue, ductile failure, and outside diameter initiated intergranular attack (IGA).
In addition, the tube diameter was greater than the nominal tube diameter, indicating that the severed tube had swollen.

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The licensee determined that the most likely cause of failure was fatigue caused by flow-induced vibration of the swollen and restrained tube. The licensee attributed the swelling to water leaking into the tube around the plugs while the unit was shut down, followed by subsequent expansion of that water during plant startup. As the plant was heated, the water in the tube expanded at a rate greater than that which could escape past the tube plugs, thereby resulting in a pressure buildup and subsequent swelling of the tube. The effect is referred to as the “diode effect.” The swelling caused the tube to become clamped at the drilled hole in the 15th tube support plate and at the upper tubesheet. Under the high-flow conditions that existed in the area of the upper tubesheet, this clamping made the restrained tube more susceptible to fatigue caused by flow-induced vibration. The licensee also concluded that the IGA on the outside of the tube may have made it more susceptible to this phenomenon.

To address the concern that other plugged tubes might impact active tubes in future cycles, the licensee embarked on a tube deplugging and inspection program. This program indicated that some plugged tubes contained water, some were swollen, and some exhibited evidence of tube degradation that was not present when they were originally plugged. The tubes that contained water or were swollen were located throughout the tube bundle, not just on the periphery as was the severed tube. The program also revealed that water or swelling only occurred in tubes that were, or had been in the past, plugged with an Alloy 600 mechanical plug. In 1991, the licensee began plugging tubes with a newer type of mechanical plug made of Alloy 690. Some of the tubes that contained water or were swollen were plugged with an Alloy 690 mechanical plug, but these were replacements for original Alloy 600 mechanical plugs. No tubes with originally installed Alloy 690 mechanical plugs in the both upper and lower tubesheet showed any signs of water or swelling.

In addition to the above findings, the tube deplugging and inspection program indicated that one additional tube had circumferentially severed and one plugged tube had failed axially. The circumferential sever occurred at the 15th tube support plate. Apparently, the tube was captured within the tube support plate as there was no wear on any adjacent tubes. The axial failure occurred in the freespan between the 15th tube support plate and the upper tubesheet. Adjacent tubes were not impacted. Both tubes were swollen and were, or had been in the past, plugged with an alloy 600 mechanical plug.

To prevent plugged tubes that might sever in the future from impacting other tubes, the licensee planned to either stabilize each plugged tube or surround it with stabilized tubes to “cage” the non-stabilized plugged tube. This stabilizing would generally be achieved by inserting cables into the tubes from the upper tubesheet through at least the 14th tube support plate. Most of the tubes that exhibited swelling would be stabilized the full length of the tube.

<u>Background</u>

TMI-1 uses once-through steam generators, in which the primary reactor coolant enters the top of the steam generator, passes through the upper tubesheet, flows down through approximately 15,500 tubes, passes through the lower tubesheet, and exits the steam generator at the bottom. Secondary-side feedwater enters the steam generator in the center, flows downward in the annular region between the tube bundle and the steam generator shell, turns upward and flows around the outside of the tubes where it is converted to steam. The steam continues upward and then turns downward in the annular region between the tube bundle and the steam generator shell and exits in the center. Secondary-side flow is characterized by counterflow conditions in the tube bundle and crossflow conditions near the upper and lower tubesheets. The highest flowrates on the secondary side of the tubes occur at the periphery of the upper tubesheet as the steam flow changes from counterflow to crossflow and turns downward in the annular region. Similar flowrates occur near the lower tubesheet, although the flow is mostly liquid in this area.

The tubes are attached to an upper and lower tubesheet and supported by 15 tube support plates, which have holes that permit the passage of the tubes. The holes in the tube support plates are broached with three points of contact, with the exception of the holes in the periphery of the 15th tube support plate which are round drilled holes. In accordance with the NRC's requirements, usually in Technical Specifications, the licensee inspects the tubes during plant outages to assess the structural and leakage integrity of the tubes. Tubes with degradation above specified limits are plugged or repaired using methods approved by the NRC.

Plugging involves removing a tube from service by installing plugs in both ends of the tube to prevent the flow of primary coolant through it. The tube plugs are typically either mechanically expanded against the tube wall or welded to the tube. Once a tube is removed from service, it no longer requires inspection because it is no longer part of the reactor coolant pressure boundary.

The tube that severed at TMI-1 and caused wearing of adjacent tubes was Tube 130, located in the B Steam Generator at Row 66 (B66-130). This tube was plugged with an Alloy 600 mechanical plug in 1986 as a result of IGA near the fifth tube support plate. At the time it was originally plugged, there was no observable degradation at the point where the tube eventually severed. The original plug was replaced in 1997 with the newer Alloy 690 mechanical plug as part of a program to replace many of the Alloy 600 plugs in the upper tubesheet.

Eddy current testing of the four adjacent tubes affected by severed tube B66-130 indicated that the tube walls had worn through 37 percent of the wall thickness for the least affected tube, to 92 percent of the wall thickness for the most affected tube. The overall length of the wear scars ranged from approximately 2.8 inches to 8.3 inches. Typically the bobbin coil probe, which is the eddy current probe most relied upon by industry to detect wear indications, is expected to detect flaws of this range. However, the licensee determined that those analysts reviewing bobbin coil data did not detect the wear indication on the least affected tube. Only those analysts reviewing rotating pancake coil data were able to identify the wear on the least affected tube at TMI-1.

The licensee also pressure-tested the four affected tubes to determine the margin to failure. Specifically, the licensee increased pressure until either the tube failed or pressure reached three times the differential pressure encountered during normal operating conditions. The most affected tube failed at a point near the design differential pressure for a main steamline break. One other tube failed above the main steamline break design pressure, but very near three times the normal operating differential pressure. These two tubes challenged the steam generator tube structural performance criteria discussed by the Nuclear Energy Institute in

NEI 97-06, "Steam Generator Program Guidelines." The remaining two tubes had no leakage when held at three times normal operating differential pressure for 2 minutes.

Discussion

Licensees have plugged tubes using a variety of different plug types and materials since steam generators were placed in service. The industry has experienced leaking plugs for at least 20 years, and probably back to the initial use of mechanical plugs. If the phenomenon at TMI-1 were widespread, numerous instances of freespan wear scars adjacent to plugged tubes should have been observed throughout the years. Although this is not believed to be the case, indicating that the phenomenon is not widespread, the potential for plugged tubes to affect the integrity of adjacent tubes may increase with time. If isolated occurrences of this phenomenon were to occur, they might be risk-significant.

Known experience suggests that the most likely consequence is an axial failure of the plugged tube unless a circumferential flaw or high crossflow velocities are present. Although the long-term potential for axial failures to cause a plugged tube to sever is unknown, such tubes are probably a lesser contributor to risk than plugged tubes that contain circumferential flaws resulting from corrosion or fatigue. Tubes with detected circumferential flaws are normally stabilized when plugged; however, undetected circumferential flaws, typically less than 40 percent of the wall thickness, will sometimes be present in tubes that are plugged for other causes. In addition, the potential for the development of corrosion-related circumferential flaws in plugged tubes is not clearly known, since the cooler tube temperatures may reduce the potential for corrosion, while higher axial loads in plugged tubes may work to increase the potential for circumferential stress corrosion cracking.

Nonetheless, the events at TMI-1 illustrate an effect in which a plugged tube can sever and affect adjacent tubes. For two of the tubes adjacent to the severed tube, the extent of wear was such that the structural performance criteria were challenged. The degradation of the active tubes appeared to occur during one cycle. The three circumstances that apparently contributed to the severed tube include tube swelling, flow-induced vibration, and tube degradation from IGA. The results at TMI-1 indicate the importance of either evaluating plugged tubes or stabilizing plugged tubes to ensure that they do not compromise the integrity of adjacent active tubes, i.e., the reactor coolant pressure boundary.

High cycle fatigue was the dominant, and perhaps sufficient, cause of the severed tube at TMI-1. Fatigue was also involved in the severed tube in the original steam generators at the R. E. Ginna Nuclear Power Plant in 1982. In addition, all known circumferential failures of active tubes have also involved high cycle fatigue (e.g., Oconee 2, Rancho Seco, North Anna 1, Mahomet). These high cycle fatigue failures involved flow-induced vibration associated with high crossflow velocities, although repeated impacts from a loose part may also have contributed to severing the plugged tube at Ginna. Nominally, plugged and active tubes are not expected to experience fatigue under flow-induced vibration; however, off-nominal conditions may enhance the potential for excessive vibration and, thus, fatigue. Such off-nominal conditions could include swelling of tubes, denting, localized flow peaking effects, or adjacent loose parts.

The licensee did not determine if only a specific population of plugged tubes was susceptible to this phenomenon and stabilize or cage only these tubes, rather they elected to stabilize or cage all plugged tubes, as discussed above. Nonetheless, the licensee did attempt to address the flow velocities and stability ratios of concern, the extent to which tubes were clamped as a result of swelling, denting, and/or other phenomena, whether the plug type (e.g., welded, mechanical) played a role in determining which tubes swell, and the effects that degradation (e.g., an axial crack, a fishmouth rupture, a circumferential crack, etc.) in a plugged tube may have on the flow induced vibration analysis.

In response to the eddy current findings at TMI-1, the NRC independently reviewed some of the bobbin coil data. For the tube least affected by the severed tube, the NRC determined that the differential channel showed a very small flaw signal and the absolute channel showed evidence of signal drift. The NRC concluded that the wear indication may have been missed because of the relatively small voltage and atypical signal behavior. The absolute drift signal rotated in the clockwise direction as channel frequencies were decreased whereas a typical flaw signal would be expected to rotate in the counter-clockwise direction.

The staff is continuing to evaluate the generic implications of the TMI-1 occurrence.

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).

/RA/

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2001-19	Improper Maintenance and Reassembly of Automatic Oil Bubblers	12/17/2001	All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.
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2001-08, Sup. 2	Update on Radiation Therapy Overexposures in Panama	11/20/2001	All medical licensees.
2001-16	Recent Foreign and Domestic Experience with Degradation of steam Generator Tubes and Internals	10/31/2001	All holders of operating licenses for pressurized-water reactors (PWR), except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor.

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