UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555-0001

October 6, 2010

NRC INFORMATION NOTICE 2010-21: CRACK-LIKE INDICATION IN THE U-BEND REGION OF A THERMALLY TREATED ALLOY 600 STEAM GENERATOR TUBE

ADDRESSEES

All holders of an operating license or construction permit for a nuclear power reactor issued under Title 10 of the *Code of Federal Regulations*, Part 50, "Domestic Licensing of Production and Utilization Facilities," except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice (IN) to inform addressees of recent operating experience with the detection of a crack-like indication in the U-bend region of a thermally treated Alloy 600 steam generator tube. The NRC expects that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. Suggestions contained in this IN are not NRC requirements; therefore, no specific action or written response is required.

DESCRIPTION OF CIRCUMSTANCES

Vogtle Electric Generating Plant (Vogtle), Unit 1, has four Westinghouse Model F steam generators. Each steam generator has 5,626 tubes fabricated from thermally treated Alloy 600. The tubes have a nominal outside diameter of 0.688 inch and a nominal wall thickness of 0.040 inch. Following the bending of the tubes, the U-bend region of the tubes installed in the first 10 rows of the steam generator was thermally stress-relieved at approximately 1,320 degrees Fahrenheit (715 degrees Celsius) for 2 hours to relieve the residual stresses induced by bending. The radius of a Row 1 U-bend is 2.20 inches. The tubes are arranged in a square pattern, and the centers of the adjacent tubes are 0.98 inch apart.

When Vogtle, Unit 1 was shut down for the 2009 outage, the unit had operated for approximately 19.8 effective full-power years. The hot leg temperature is approximately 618 degrees Fahrenheit (326 degrees Celsius).

During the cycle before the 2009 refueling outage, no primary-to-secondary leakage was observed at Vogtle, Unit 1; however, while the unit was being shut down, there were a few radiation monitor alarms, indicating the presence of activity on the secondary side of the unit. The leak rate was too small to measure. With the static pressure from the water on the secondary side of the steam generator acting on the tubes, there was no visible evidence of leakage coming from any of the tubes.

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For this refueling outage, the licensee planned to use an eddy current rotating probe equipped with a +Point™ coil to inspect the U-bend region of approximately 50 percent of the tubes in Rows 1 and 2 in each of the four steam generators. During these rotating probe examinations, the licensee found an axially oriented crack-like indication approximately 3 inches above the uppermost hot-leg tube support plate (i.e., the seventh support plate) in the tube in Row 1, Column 20 (R1C20), in Steam Generator 3. The crack-like indication was close to the apex of the tube on the extrados and was attributed to primary water stress-corrosion cracking. The voltage amplitude of the indication, as measured with the midrange +Point™ coil, operated at 300 kilohertz (kHz) was 3.09 volts. The axial length of the indication was 0.54 inch, its maximum depth was 100 percent through-wall, and it exceeded 80 percent through-wall over 0.40 inch of its length.

As a result of detecting this indication, the rotating probe examinations in the U-bend region were expanded to include 100 percent of the Row 1 and 2 tubes in each of the four steam generators and 20 percent of Row 3 tubes in Steam Generator 3. No other crack-like indications were found in the U-bend region during these examinations. The licensee plugged the tube in R1C20 in Steam Generator 3.

The U-bend region of the tube in R1C20 in Steam Generator 3 was inspected in prior outages. The U-bend region was inspected with a bobbin coil in 1986 (baseline inspection), 1991, 1993, 1997, and 2000. A rotating probe examination of the U-bend region of this tube was also performed in 1997, 2003, and 2009. The practice of inspecting the U-bend region of tubes in low rows (e.g., Rows 1 and 2) with only a rotating probe (and not with a bobbin probe) is a fairly common practice today, given the difficulties in performing quality inspections of tight radius U-bends with a bobbin probe.

The prior inspections of this tube indicated the presence of a manufacturing indication, referred to as a Blairsville bump (because the bump was most likely introduced during bending of the tube at a facility in Blairsville, PA). This bump is located at the start of the bent region of the tube (i.e., the start of the U-bend region). During the review of the bobbin coil data obtained in 2000, one of the analysts reviewing the data (typically two analysts review all eddy current data) identified a nonquantifiable indication at the location where the crack-like indication was eventually discovered. This indication was eventually dismissed by the resolution analyst (an analyst who oversees the review of the primary and secondary data analysts) because the 1997 rotating probe examination indicated that no flaws were present at this location, the bobbin coil data indicated that the signal had not changed since the 1986 inspection, and there was a general absence of any cracking in tubes fabricated from thermally treated Alloy 600 tubing at the time of the inspection. During the review of the 2003 rotating probe data, an axial indication was reported by one of the analysts at the location where the crack-like indication was eventually discovered. This indication was also dismissed by the resolution analyst because the indication from the rotating probe did not change appreciably from 1997 (1.75 volts as measured from the 300-kHz channel) to 2003 (1.83 volts as measured from the 300-kHz channel).

The tube in R1C20 in Steam Generator 3 was pressure-tested in situ in 2009 to confirm its integrity. During the test, no leakage was observed under simulated normal operating conditions, approximately 0.002 gallons per minute was observed under simulated steamline break conditions, and approximately 0.09 gallons per minute was observed at three times the normal operating differential pressure. The tube did not burst at three times the normal operating differential pressure, demonstrating that it retained adequate structural integrity. The leak rate from this tube under accident conditions, when combined with all other sources of accident-induced leakage, was within acceptance limits, demonstrating that the steam generator had satisfied the accident-induced leakage performance criteria.

BACKGROUND

Previous related generic communications include the following:

- NRC IN 2003-13, "Steam Generator Tube Degradation at Diablo Canyon," dated August 28, 2003 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML032410215)

- NRC IN 97-26, "Degradation in Small-Radius U-Bend Regions of Steam Generator Tubes," dated May 19, 1997 (ADAMS Accession No. ML031060007)

DISCUSSION

Cracking in the U-bend region of steam generator tubes has been observed for years. Many of these cracks have been located in the U-bends of tubes in Rows 1 and 2 as discussed in NRC IN 97-26; however, there have been some cracks in the U-bends of higher row tubes as discussed in NRC IN 2003-13. All of these cracks have been in tubes made with mill-annealed Alloy 600 tubing. (Note: Although NRC IN 97-26 indicates that an axial indication was discovered in the U-bend region of a thermally treated Alloy 600 tube at Braidwood Station, Unit 2, subsequent evaluation indicated that this indication was not a crack).

The findings at Vogtle, Unit 1 are particularly noteworthy because this is the first confirmed instance of cracking in the U-bend region of a thermally treated Alloy 600 tube. Although there was a manufacturing indication near the location where the crack developed, it is not known whether this condition is necessary to lead to the initiation of a crack in this region of the tubing. In addition, although the crack at Vogtle, Unit 1 occurred in the U-bend region of a Row 1 tube, it is not clear that this phenomenon would be primarily limited to low-row (i.e., Rows 1 and 2) tubes as it primarily was for units with mill-annealed Alloy 600 steam generator tubing. In the units with mill-annealed Alloy 600 tubing, the U-bends were typically not stress relieved after bending. As a result, the residual stresses; and therefore, the susceptibility to cracking, would normally be expected to be highest in the lowest row tube (i.e., the one with the shortest bend radius), provided there were no significant changes in the bending processes from one tube row to another. However, in the case of the units with thermally treated Alloy 600 tubing, the U-bends of the low-row tubes (i.e., Rows 1 through 8, 9, or 10, depending on the model) were stress relieved after bending. Assuming the stresses were fully relieved as a result of this process, the resistance to cracking should generally be the same if all other factors (such as material microstructure) are the same.

In 2003, the licensee concluded that there was no flaw at the location where the crack was eventually observed in 2009 in the tube in R1C20 in Steam Generator 3. Its conclusion was based on observing only a minor change in the eddy current signal (bobbin and rotating probe) between 1997, 2000, and 2003 and the general lack of cracking in units with thermally treated Alloy 600 tubing. Following the discovery of the crack in 2009, the NRC staff reviewed the 2003 and 2009 rotating probe eddy current data for the tube in R1C20. Although the NRC staff did not have all of the information available to the licensee, the NRC staff's review of the 2003 rotating probe data indicated the presence of a flaw-like signal. These results highlight the limitation of confirming flaw signals based on signals exhibiting change from one inspection to the next and the difficulties in detecting new or unexpected forms of degradation. Detecting primary water stress-corrosion cracking (i.e., cracking that initiates from the inside diameter of the tubing) may be enhanced by using smaller probes operated at high frequencies. Although the noise levels in the data obtained at higher and lower frequencies may be comparable, the signal level associated with a flaw on the inside diameter of the tube may be significantly greater at the higher frequencies. This will increase the signal-to-noise level for this type of flaw and improve the likelihood that it would be detected.

As discussed above, at least one of the eddy current data analysts during the 2000 and 2003 inspections identified a signal at the location where the crack was eventually observed in 2009. These signals were dismissed by a resolution analyst. To limit the potential for human error and to improve detection of flaws, some licensees require two independent resolution analysts to review the signals identified by the primary and secondary data analysts rather than assigning this review to a single resolution analyst. In addition, when dismissing indications based on a historical review of the data, two independent analysts reviewing the data may improve the detection of flaws or changing conditions.

CONTACT

This IN requires no specific action or written response. Please direct any questions about this matter to the technical contact listed below or to the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

/RA/ by TQuay for

Timothy J. McGinty, Director
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Note: NRC generic communications may be found on the NRC public Web site, http://www.nrc.gov, under Electronic Reading Room/Document Collections.

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